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Analysis of Three Mile Island – Unit 2 Accident

NUCLEAR
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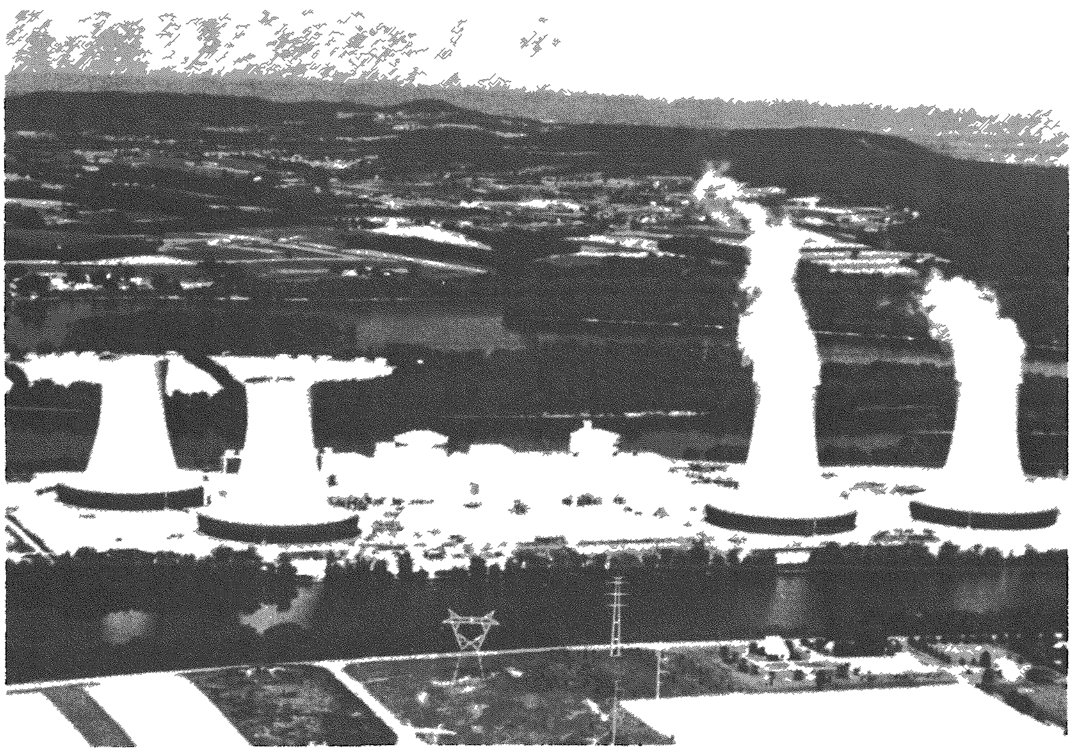

Prepared by the staff of the Nuclear Safety Analysis Center

DISCLAIMER

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LEGAL NOTICE

FOREWORD

The Three Mile Island Accident and its contributing circumstances have been reported in four major published studies. These are in chronological order: (1) the study by the Nuclear Safety Analysis Center report issued July, 1979 (NSAC-1 and supplements), (2) the study by the Nuclear Regulatory Commission issued August, 1979 (NUREG 0600), (3) the President's Commission Report (Kemeny Commission) issued October, 1979, and (4) the Rogovin Report, issued January, 1980. The effort at the Nuclear Safety Analysis Center has been continuous since May, 1979 -- an effort now totaling in excess of 12 man years of data-gathering and technical analysis. Many of the implied lessons from these studies have been treated in three documents issued by the NRC; NUREG 0578 on Short Term Lessons Learned, NUREG 0585, "Final Report of the TMI lessons learned Task Force, and NUREG 0660 "NRC Action Plan" (Draft 3, March 4, 1980). These documents list a large number of prescriptive remedies or actions, many of which have been ordered by NRC and are already implemented - or are in the process of implementation by utilities.

The conclusion of the Kemeny Commission, and the Rogovin Report, and the implication of the NRC Action Plan are that a very large number of things are in need of major changes in order to minimize the likelihood of future accidents. There is a seriously misleading implication in these reports that most things are done poorly most of the time. Taken literally this view can be counter-productive to improved safety in the future.

A more illuminating and factual view of the deficiencies discovered in the studies of the Three Mile Island Accident can be obtained by noting that the regulatory process and disciplined reactor operation has produced an extremely good record of both plant and public safety, when viewed objectively. <u>Even including</u> the Three Mile Island Accident the actual damage to the public is much smaller for the energy produced than most other available sources of energy.

The NSAC-1 analysis of the TMI accident, and of related but less serious events before and since, supports the view that most operations are conducted adequately to protect public health and safety most of the time.

Good practices in equipment, maintenance, management and operation are the rule rather than the exception. However, lapses from good practice do occur. Most lapses do not affect public safety, but have evident economic effects. For example, about 15% of all plants have outages substantially larger than normal refueling times in most years.

The basic remedy for the evident deficiency shown by the Three Mile Island accident can be expressed as follows: <u>It is essential that the good practices of the industry - in equipment,</u>

management and operation, which are generally observed by most utilities - be rigorously extended to all, and that the frequency of lapses from good practice be reduced. However, perfection is not necessary.

Systems are designed and operated to be accommodating or fail-safe to a considerable number of lapses and equipment malfunctions or failures. For example--about 20 factors contributed to the damaging results at Three Mile Island. If any one of these factors had been different--in a way which is common in other plants--there would have been no core damage and no release of radioactivity.

Practical means are available to reduce the frequency of lapses which may occur in the future. There are also practical means to anticipate or detect, and to remedy lapses from good practice in operation, design, or application, before they are likely to lead to severe damage to reactors, or to adverse environmental effects.

The main generic lessons learned and already being implemented by industry fall into the following broad categories:

1. Increased scope of continuous technical support for reactor operation and safety surveillance.

2. Added measures and criteria for selection and training of operators, including more comprehensive procedures for coping with plant upsets and accidents.

3. Organization and procedures for insuring systematic and rigorous learning from operating experience.

4. Equipment or design changes for improved ease of operation of existing plants.

5. Evaluations of long term improvements in reactor design.

6. Establishment of emergency response organizations and defined decision-making processes.

7. Establishment of practical goals for reactor safety.

These generic lessons are being treated primarily by suitable actions by the utilities, with supplemental and supporting actions by reactor and equipment suppliers, new institutions such as NSAC and INPO, and enabling actions by the regulatory agencies.

The basic lessons learned--when applied intensively, can help prevent the full range of accidents involving core damage, environmental effects, and their social and fiscal consequences.

A report summarizing "lessons learned" as they are already largely being implemented by the utility industry - and including practical goals for reactor safety, is being issued separately. This report was prepared by the Nuclear Safety Analysis Center (NSAC), operated for the utility industry by the Electric Power Research Institute (EPRI). Individuals from over 30 different organizations participated in this work, some throughout the entire period of preparation and some for shorter periods. These individuals and their affiliations are listed below.

The first edition of this report was published in July 1979, followed by a supplement in September 1979. The present revision combines the supplement with the main report, makes various editorial corrections and changes or adds to the report.

Notably, an executive summary has been added, in order more clearly to explain the system and the accident. A comparison of the NSAC Sequence of Events with those from four other sources has been made and is included as Appendix SOE COMP. A new appendix "Alternate Success Paths" (ASP) has been added, conveying in qualitative fashion the fact that, in spite of the seriousness of the accident, there were numerous barriers, in the form of actions which the operator could take, or actions which the available plant equipment would take or physical barriers which together make it very improbable that the accident could have proceeded to the point of a massive release of radioactivity to the environs. A third new appendix "Precursor Events" summarizes and briefly discusses the events at Oconee 3 and Davis-Besse 1 which could be regarded as precursors to TMI-2. The graphical display of plant parameter variations during the accident (The Equipment and Systems Action Matrix, ESAM) has been re-drawn to provide more information.

Appendices which have been expanded or significantly updated include CI (Core Instrumentation and Analysis of Coolant Level), OTSG (Once Through Steam Generator), RM (List of Radiation Monitors and Their Locations) and ROUTES (Potential Routes of Radioactivity Transport). In the case of ROUTES some information on radiation exposure of population in the vicinity of TMI-2 has been added. However, the principal investigation of this subject is still underway in a study sponsored jointly by NSAC and the State of Pennsylvania.

Each element of the report was prepared by a team whose leader was an EPRI employee assigned to NSAC. Team members were EPRI employees, consultants, or employees on loan from different organizations or companies which had special expertise in relevant areas of design, analysis, operation, thermal-hydraulics, core behavior, and instrumentation. Data and extensive assistance and review of data for numerical accuracy were provided by the organizations responsible for the design or operation of TMI-2 (Babcock & Wilcox, Burns and Roe, General Public Utilities, and Metropolitan Edison), but these organizations did not participate in the writing of the report, and bear no responsibility for content or conclusions, either explicit or implicit.

The cooperation and substantial efforts of GPU and Metropolitan Edison personnel in promptly seeking out and supplying records, data, and supporting technical information is especially notable. This included providing hard copy of many plant records, tapes, manuals, procedures, logs, and the like as requested. These data were supplemented by an indexed microfilm library of about 75,000 frames, covering the instrumental records for the period of the accident.

During the preparation of the report about 65 persons were involved at EPRI. The total professional effort applied for the initial and revised editions of the report was about 12 man years.

Comments received on the initial edition of the report and supplement have been considered in the preparation of this revision, but the revised report itself has not been circulated for industry review prior to publication.

Both the initial version and this revision of the report focus on the observable effects of the accident which are supported by firm, recorded data, and the inferences which can be reasonably directly made or calculated from the known data. Some new data have become available since the initial version of the report, and further study has permitted drawing more conclusions than could be justified at the time of the issuance of that version.

Some differences in this document from the other primary sequences of events, narratives and analyses (those prepared by NRC, by G.P.U.-Met. Ed., and by B. & W.) appear to arise from the following:

a) Omission of some events from this version which do not appear relevant to causation, prolongation or consequences of the accident.

b) Primary reliance in this document on the reactimeter tape for reliable time scale.

c) Omission from this report of interpretations or inferences based entirely or largely on operator recollections and not directly confirmed by recorded data.

Some elements of the analyses have substantial ranges of uncertainty due to inherent limitations in the original records. Some of the factors with considerable inherent uncertainties include reactor water levels and system inventories versus time, maximum regional and local temperatures in the reactor core and vessel, and detailed damage assessments. In addition, it has been discovered that the data plotted in some of the graphs, particularly in Appendix TH, contain time discrepancies of the order of a few minutes or similar small pressure discrepancies, in a few cases. These discrepancies do not affect the conclusions of the report, but any reader who proposes to use the graphs as a source of detailed, precise information should first consult NSAC.

Aspects such as what the operators apparently thought and what information was easily available to them are considered to a limited extent in this revision.

Support for NSAC is being provided jointly by public and private sectors of the utility industry, including; American Public Power Association, Bonneville Power Administration, companies of the Edison Electric Institute, National Rural Electric Cooperatives Association, and the Tennessee Valley Authority. A total of 63 utilities have designated coordinators for their response to issues raised by the TMI accident and to work with NSAC as needed.

Edwin L. Zebroski

E. L. Zebroski
Director, Nuclear Safety Analysis Center

Participants in this report are listed below:

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George Thomas	Yankee Atomic
Dan Wilkinson	General Electric
Dan Williams	Arkansas Power
Keith Woodard	Pickard, Lowe, & Garrick
Larry Young	Arkansas Power
Ed Zebroski	Electric Power Research Institute

In addition, data and supporting information on equipment design and characteristics were received from the organizations responsible for design and operation of the plant: Babcock & Wilcox, Burns & Roe, General Public Utilities and Metropolitan Edison. For Example, GPU supplied microfilmed records and a computerized index. Significant periods of effort were provided by the following individuals:

Robert Bredder	Burns & Roe
John Brummer	Metropolitan Edison
Ron Gribble	Babcock & Wilcox
James Hagan	Burns & Roe
Ed Kreahling	Metropolitan Edison
James Lemon	Babcock & Wilcox
Robert Long	General Public Utilities
James Mallay	Babcock & Wilcox
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Ivan Porter	Metropolitan Edison
Joseph Steinbrunner	Babcock & Wilcox

Contents

ANALYSIS OF THREE MILE ISLAND ACCIDENT

Introduction

The Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Institute has analyzed the Three Mile Island-2 accident. Early results of this analysis were a brief narrative summary, issued in mid-May 1979 and an initial version of this report issued later in 1979 as noted in the Foreword. The present report is a revised version of the 1979 report, containing summaries, a highly detailed sequence of events, a comparison of that sequence of events with those from other sources, 25 appendices, references and a list of abbreviations and acronyms. A matrix of equipment and system actions is included as a folded insert.

The appendices serve either to describe plant features which are pertinent to the understanding of the sequence of events, or to indicate how certain inferences and conclusions in the report were reached.

This report embodies the principal results of that phase of NSAC's work which is devoted to learning and understanding what happened during the TMI-2 accident. Subsequent phases will concentrate on causes, lessons learned and generic remedial or preventive measures which may be appropriate. These phases are being reported separately.

EXECUTIVE SUMMARY OF TMI-2 ACCIDENT

Introduction

As a review of how a power plant of the TMI-2 type works, the following paragraphs build up pictorially and in words, a step at a time, a picture of such a plant and its functioning.

A nuclear power reactor has a uranium oxide core which produces heat energy that is converted to steam in a boiler. The steam turns a turbine and generator which makes electric current for distribution to the public. In a pressurized water reactor, the type installed at TMI-2, there are three cooling circuits. The primary circuit (green in Fig. 1) circulates water through the reactor core. The primary cooling water picks up the heat from the fission reaction and carries it out of the core to two steam generators (or boilers). These are 35 foot tall tanks in which the primary water passes through a large number of small-diameter tubes, transferring heat to water flowing in the secondary circuit (blue in Fig. 2), which is outside these tubes. Water in the secondary circuit boils to make steam which goes to the turbine-generator unit. The steam passes from the turbine to a condenser which is cooled in turn by water from the cooling towers (yellow in Fig. 3).

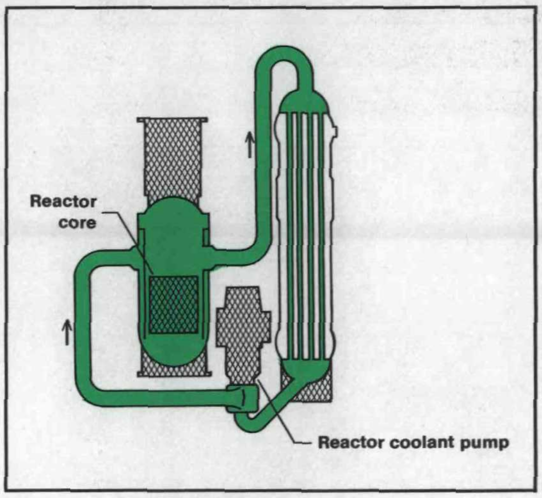


Figure 1.

The water in the primary loop is kept from boiling by keeping it under high pressure--normally about 2200 pounds per square inch. A large vessel connected to the primary loop called the pressurizer (Fig. 4) is used to maintain this pressure. The pressurizer is about half full of water, with a steam (red)

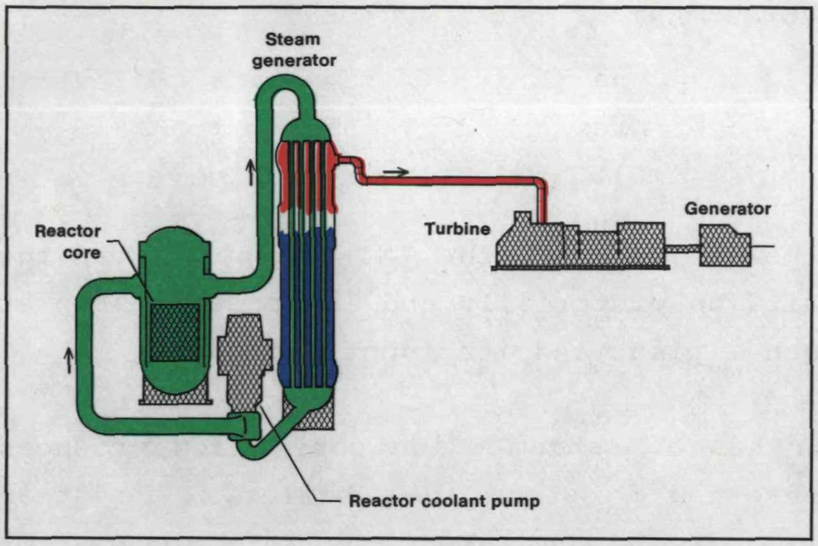


Figure 2.

cushion in the top half. As the water in the primary loop heats up or cools down, it expands or contracts by many hundreds of cubic feet. The steam cushion in the pressurizer takes up the slack, while maintaining pressure on the primary water.

The control system or the plant operators control the pressure exerted by the pressurizer by controlling the temperature of the pressurizer with electric heaters, and with a cooling water spray.

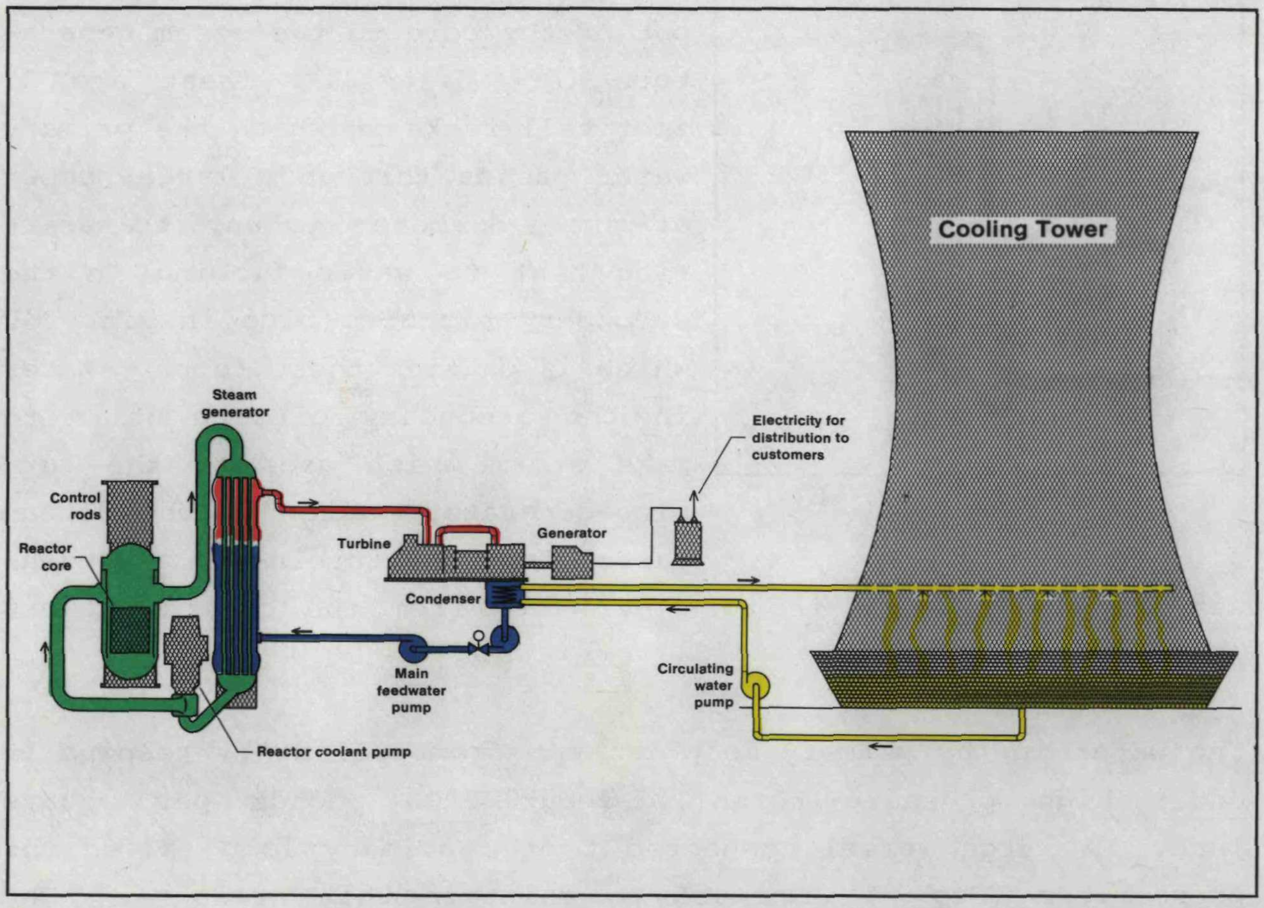


Figure 3.

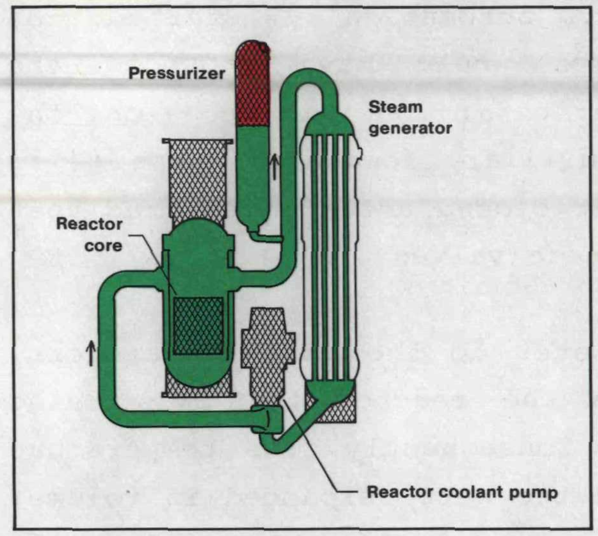


Figure 4.

A relief valves are provided on the pressurizer to prevent over-pressuring the system (see Figure 5). This valve is a power-operated relief valve. If this valve is opened to relieve the excess pressure, the steam or water flows to a drain tank. If the drain tank becomes overfilled, a rupture disk is provided on that tank to relieve pressure. Also, since valves sometimes stick, the relief valve has a backup, which is called a block valve. Additionally, two large safety valves provide protection against larger transients.

The Accident

On the morning of March 28 at 4:00 a.m., the Three Mile Island Nuclear Power Station Unit 2 reactor was operating at full power. A condensate pump in the blue circuit (1 in Fig. 6) tripped off-line--that is, stopped operating. Lacking feedwater supply, the two feedwater pumps in the blue circuit (2 in Fig. 6) also tripped (shut off automatically) within one second, since these pumps have a protection system which shuts them down if there is no water in the feedwater supply line. Stoppage of the two feedwater pumps activated a safety circuit which automatically shut down the turbine-generator. Three auxiliary feedwater pumps (3 in Fig. 6),

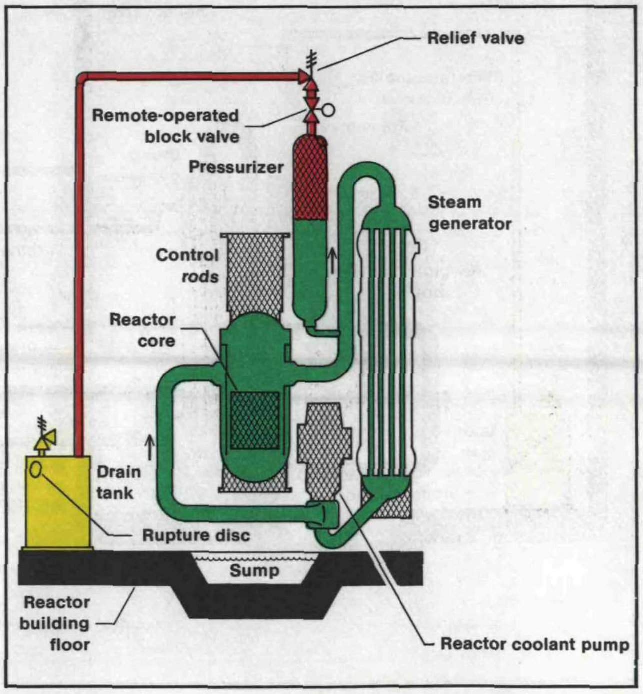


Figure 5.

which are provided as backups, automatically started, as designed. Normally, these pumps would have provided the feedwater needed by the steam generators. On this occasion, the block valves downsteam from the auxiliary feedwater pumps (4 in Fig. 6) had inadvertently been left closed after a required test operation which involved closing these valves.

As a result of the loss of feedwater to the steam generators, more heat was being produced in the reactor than was being removed by the steam generators. Consequently, the temperature of the primary water increased and the water expanded in volume; this caused a flow of primary water coolant into the pressurizer. This flow compressed the steam cushion in the pressurizer and increased the pressure in the primary system.

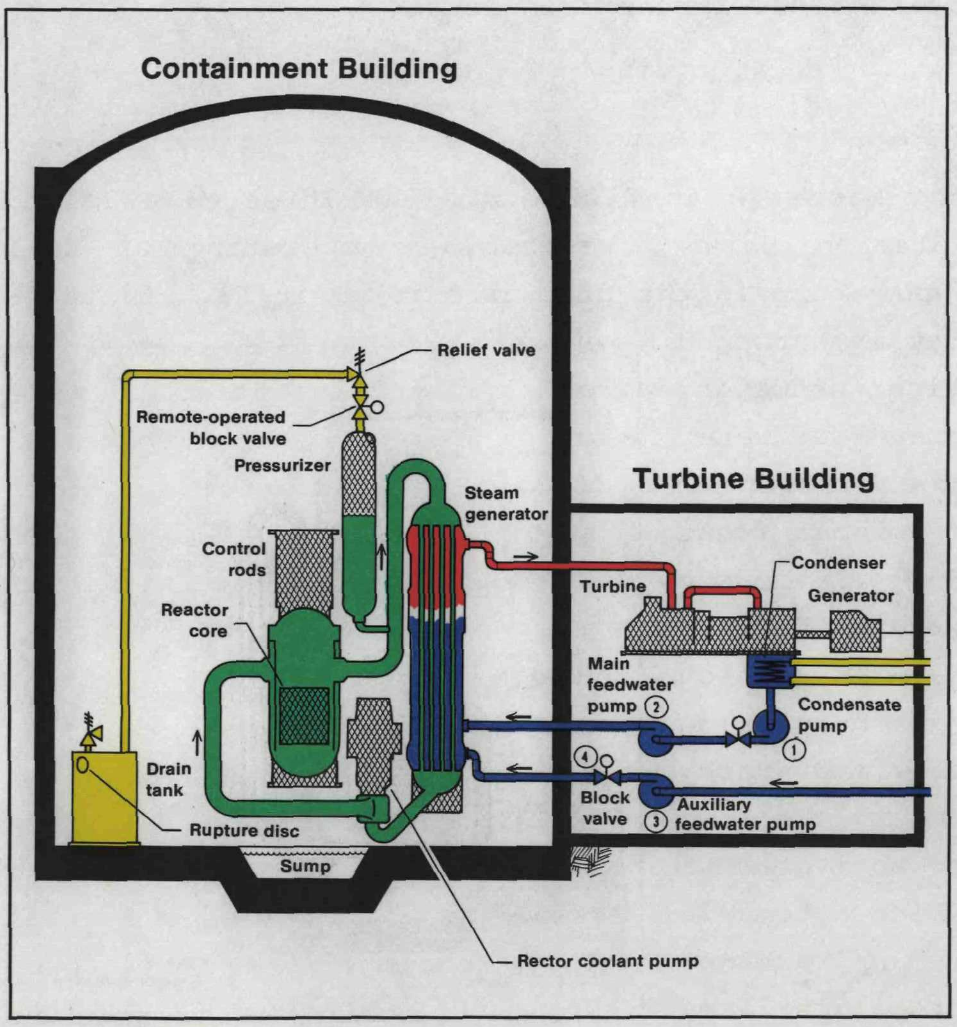


Figure 6.

Four seconds after the initial stoppage of the condensate pump, the water level and pressure in the pressurizer had increased so much that the power-operated relief valve automatically opened, as it was supposed to. Pressure continued to increase until, nine seconds after the first pump stoppage, the reactor shut down automatically in response to the high-pressure. After this, pressure in the primary circuit began to decrease, as normally anticipated, so the operators, in accordance with the procedures, turned on the high pressure injection system.

As would be expected, with the reactor shut down and the relief valve open, both the water level in the pressurizer and the pressure in the primary system continued to decrease. The decrease in pressure should have caused the relief valve on the pressurizer to close automatically when the system reached normal operating pressure. But for reasons not yet known, it stuck open. Had the relief valve closed the pressure would have ceased to fall, and the water level in the pressurizer would have stabilized or started to rise.

As escape of water and steam continued, the water level in the system fell. The pressure in the pressurizer also fell, falling so low that generation of steam started in the reactor core, a condition for which this type of reactor is not designed. This accelerated the normal rise of the water level in the pressurizer, and the operators, thereupon thinking that the system was filling up with water, turned off the high pressure injection system as was their normal procedure. (This procedure is necessary to avoid completely filling the system with water or "going solid", by eliminating the pressurizer steam bubble, without which it would be difficult to control primary system pressure.)

In this case, however, the increase in pressurizer level continued, due to the continued growth of the steam region in the core, even though high pressure injection water had been shut off. The combination of a rising water level in the pressurizer and a decreasing primary system pressure was new to the operators, and they did not understand what it signified until almost 2-1/2 hours later when they realized that the power operated relief valve had stuck open. They then closed its blocking valve thus stopping flow through it.

Meanwhile several important events took place. First, the escaping water and steam filled the drain tank to which they are piped and, as flow continued, caused the rupture disc in that tank to burst. This resulted in hot water and steam discharging into the reactor building.

Second, as the primary loop pressure fell and the water in it started boiling it was necessary to turn off the primary coolant circulating pumps since they would be severely damaged by pumping a mixture of steam and water. Up to this time the pumps had been forcing water into the core, thus providing adequate cooling in spite of the loss of coolant through the stuck open valve. But when the pumps were shut off (the last one at about 1 hour and 40 minutes into the accident) core cooling was no longer adequate to prevent fuel damage by overheating.

Third, the operators commenced a varied series of operations aimed at controlling the water level in the pressurizer. None of these operations eliminated the steam from the core; on the contrary, more and more of the core became filled with steam. The fuel rods in the steam filled regions boiled dry and then heated to temperatures so high that their zircaloy cladding reacted chemically with the steam in which they were bathed. A product of this chemical reaction is hydrogen, some of which mixed with the steam and was eventually discharged via the stuck open relief valve and the drain tank into the reactor building. The damaged cladding failed and liberated radioactive fission products from the nuclear fuel itself (uranium dioxide). These fission products then mixed with the water, steam and hydrogen and thus were also partly discharged into the reactor building.

After the relief valve block valve was closed, at almost 2-1/2 hours into the accident, attempts to halt steam production in the core continued, but were hampered at various times by several problems:

- The coolant temperature was so high that the normal coolant circulating pumps could not be used due to cavitation, as already mentioned.

- Not until some hours later was once-through cooling, injecting water with the high pressure injection pumps and discharging it through the relief valve, capable of removing all the heat generated in the core, because of the relatively small size of the relief valve.

- For part of this period the main condenser was out of service due to a problem with an oil-fired auxiliary boiler, while the State of Pennsylvania resisted discharging steam from the secondary loop to the atmosphere as would have been the normal safe procedure. Hence the steam generators could not function to remove heat from the primary loop. (The state authorities apparently feared that the steam was harmfully radioactive, which was not the case.)

- The hydrogen interfered with heat transfer in the steam generators, so that they were inefficient as heat sinks even when their steam could have discharged or condensed.

In spite of these difficulties the system was eventually cooled to the point that the reactor coolant circulating pumps could be started up again and the main condenser was restored to service. Thus, at about 16 hours into the accident, a stable condition was established, steam generation in the core had been arrested, and the core temperature was declining.

However, the presence of hydrogen presented problems both during and after this period. First, the hydrogen which had escaped into the reactor containment building and mixed with the reactor building air atmosphere reached a concentration such that the mixture could ignite. Numerous sources of ignition (switches, etc.) were present and conflagration took place. The pressure reached during this conflagration was about 28 psi, which is less than half of the design pressure of the building, so its integrity was not threatened. Hydrogen concentrations twice as high as those which existed would have been necessary for the pressure to reach the design pressure of the building, which has in it another factor of safety of 2 or 3. Second, after stabilization at 16 hours it was concluded that some of the hydrogen remaining in the reactor was in the form of a bubble in the top of the reactor pressure vessel. Since there was no way of venting this bubble, it had to be removed slowly by passing water through the reactor and allowing the hydrogen to dissolve in the water. This operation required several days. During this period there was widely publicized concern that this hydrogen bubble might explode and burst the pressure vessel. This concern was ill-founded since in order for it to explode there would have had to be oxygen mixed with the hydrogen. There was not and could not have been such oxygen present in the reactor pressure vessel.

Analysis after the accident has indicated that serious damage to the reactor fuel did not start until after about 1 hour and 40 minutes into the accident. Had the stuck open relief valve been discovered prior to that time no significant fuel damage would have occurred. Actual rupture of a number of fuel rods apparently did not occur until about the time the relief valve actually was closed. Thus, had discovery that the valve was stuck open occured even a quarter or half hour earlier, there would still have been severe fuel damage but only relatively little release of radioactivity into the containment.

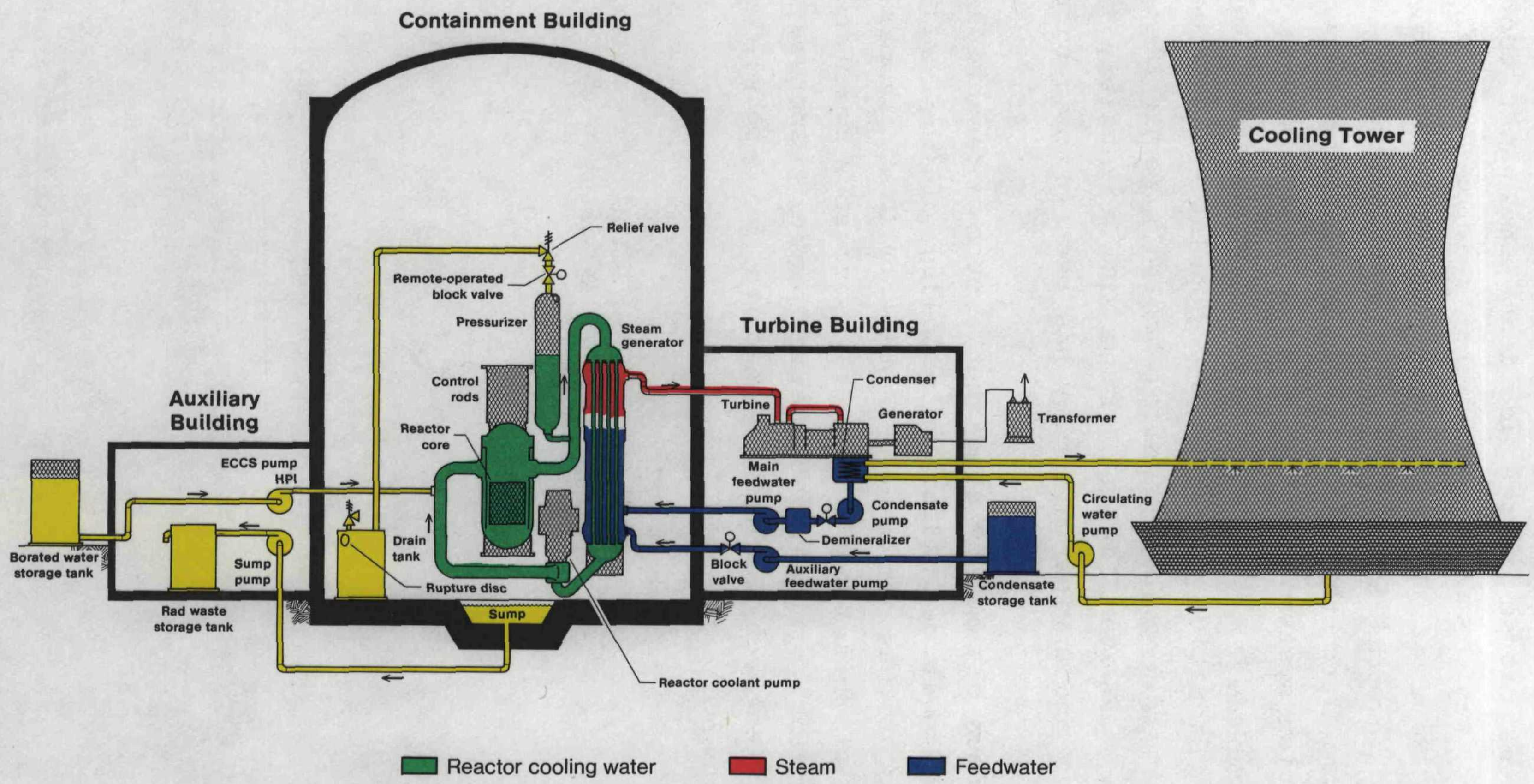
Post Accident Status

Within a few weeks after the accident the reactor heat had decayed sufficiently that the reactor coolant pumps could be shut off, and the heat could be carried away by natural circulation of the water. As of March, 1980, the reactor was being cooled by natural circulation at a heat generation level of 0.2 MW, or about 0.007% of rated power. Clean up operations are underway.

During the course of the accident some water containing radioactivity was removed from the reactor building at various times. Some of the radioactive gases dissolved in this water evolved into the atmosphere of the auxiliary building and found their way into the plant environs via the auxiliary building ventilation system. Radiation surveys were made around the plant during and after the accident at distances out to more than 40 miles. Radiation doses to the population were so small as to produce no detectable physical health effects (Kemeny Commission Report). However, mental trauma due to anxiety created by unnecessary and unwise publicity has been claimed by some persons.

It has been claimed by some that a much more serious accident, in which very large radiation exposure of the nearby population could occur, was imminent. Studies by NSAC, summarized in Appendix ASP of this report, show that numerous barriers, in the form of operator actions, automatic plant responses and built-in passive features would have had to fail before such an event could occur.

TMI-2 SCHEMATIC



TECHNICAL SUMMARY
OF THE THREE MILE ISLAND UNIT 2 ACCIDENT

The EPRI Nuclear Safety Analysis Center has conducted a detailed review of the Three Mile Island Unit 2 accident and of the lessons to be learned from it. This review has concentrated primarily on the events in the plant during the sixteen hours following initiation of the accident since after that time plant conditions were relatively stable and controllable. A sequence of events has been developed and has been verified and annotated by comparing oral and written statements with the instrumentation records, data logs, operator logs, and inferences which can be made from these records by straightforward calculations.

During the course of reviewing the accident at TMI-2, it is useful to keep in mind two basic nuclear reactor safety requirements: 1) the need to keep sufficient coolant in the core to keep the core from overheating; and 2) the need to remove heat from the reactor coolant system to keep the system from attaining excessive temperatures or pressures, that is, the need for a heat sink.

Initial Events (First Minute)

The accident was initiated at 4:00 a.m., March 28, by a loss of normal feedwater to the steam generators resulting in a turbine trip. The interruption of feedwater flow to, and of steam flow from, the steam generators caused a reduction in heat removal from the reactor coolant system. The reactor coolant system responded to this initiating event in a normal manner as follows: reactor coolant system pressure increased because heat was not being removed from the system at sufficient rates by the steam generators; the electromatic relief valve (also termed the power operated relief valve) operated to relieve pressure; the reactor shut down automatically because of a high-pressure trip signal; heat generation from the reactor dropped to the decay heat level; within a few seconds the system pressure dropped to normal values. To this point, normal reactor protection features had functioned as intended by design. Approximately forty seconds into the event, the steam generator water level dropped to the point where automatic controls called for emergency feedwater, to maintain a minimum steam generator water level. However, closed valves between the control valves and the steam generators prevented emergency feedwater from being delivered to the steam generators (these valves were opened by the operators approximately 8 minutes after the accident was initiated).* The opening of the electromatic relief valve in a loss of feedwater transient, whether or not emergency feedwater is available, is a normal response and in accordance with design.

Loss of Reactor Coolant

The electromatic relief valve, which relieved excess pressure as intended, should have closed when pressure was reduced sufficiently. Instead, it remained open, thereby allowing continued coolant discharge from the reactor coolant system, and causing a further decrease in reactor coolant system pressure. The discharge from the electromatic relief valve is piped to the reactor coolant drain tank. Because of the continuing flow into the drain tank, a safety valve on this tank lifted at approximately 3 minutes and a rupture disk on the tank burst at approximately 15 minutes into the accident. Reactor coolant continued to flow through the open electromagnetic relief valve, into the reactor coolant drain tank, and through the ruptured disk of this

*Appendix TH of this report indicates that early unavailability of feed water did not significantly affect the course of the accident, beyond possibly adding to the concerns of the control room operators.

drain tank into the reactor building sump. This loss of the reactor coolant continued without interruption until approximately 2.4 hours into the accident when the block valve, which is in series with the electromatic relief valve, was closed.

The indications in the control room of electromatic relief valve position were ambiguous to the operators. An indicating light on the control panel indicates only that the actuation solenoid is energized. There is no direct valve stem position indication. Temperature readings downstream of the electromagnetic relief valve were ambiguous because the temperature there had been elevated prior to the accident by leakage through one of the safety valves or the electromagnetic relief valve and by the known opening of the electromagnetic relief valve a few seconds after the start of the transient. The behavior of the reactor coolant drain tank pressure could have been used as an indication of continued discharge of reactor coolant water into that tank, but the reactor coolant drain tank pressure indicator is located on a back panel in the control room. An operator must walk to this panel to see it.

The accident still might have been terminated at up to 100 minutes with little or no damage to the reactor core by closing the block valve upstream of the relief valve, and/or pressurizing the system to above saturation pressure with the high pressure injection pumps, but the fact that the relief valve was open was not recognized until later.

Engineered Safety Features Actuation of the High-Pressure Injection System

At approximately 2 minutes into the accident, the safety injection system came on as designed, in response to a signal of low pressure (1640 psig) in the reactor coolant system. The high-pressure injection pumps immediately began delivering water to the reactor coolant system. At approximately 3 minutes into the accident, an operator bypassed the high pressure injection actuation signal. At approximately 4-1/2 minutes into the accident, in response to indications of high coolant level in the pressurizer, the operator turned off one of the high pressure injection pumps. Flow from the remaining high pressure injection pump was then reduced. Full high pressure injection flow was automatically initiated again about 3.3 hours into the event. During the intervening time, the addition of coolant to the reactor coolant system was less than the loss of coolant from the system through the open electromagnetic relief valve and through the letdown system. Steam voids accumulating in the reactor coolant system outside of the pressurizer prevented pressurizer coolant level instrumentation from serving as an indicator of total coolant inventory. The pressurizer surge line configuration includes a loop seal which can preclude, at times, a direct relationship between the pressurizer coolant level and the coolant level in the reactor vessel.

Reactor Coolant Pumps

During the first 73 minutes all four reactor coolant pumps continued to operate and circulate coolant through the reactor. However, because coolant continued to be discharged through the open electromatic relief valve, the reactor coolant system depressurized and became a circulating mixture of steam and water. The fraction of steam in the reactor coolant system continued to increase during this first 73 minutes. At this time both reactor coolant pumps in loop B were turned off in response to indications of low system pressure, high vibration, and low coolant flow. (Such indications normally require protective action since they imply potential for serious damage to the pumps.) The effect of turning off these pumps was to produce separation of the steam and water phases of the coolant in loop B, apparently precluding further significant flow in that loop. During the period up to 100 minutes, the reactor coolant pumps in loop A continued to circulate coolant through the reactor. At about 100 minutes the reactor coolant pumps in loop A were turned off in response to indications of low system pressure, high vibration, and low coolant flow. Procedures call for such action to protect the pumps.

The vibration and reduced flow of the pumps imply that a mixture of steam and water was passing through the pumps, piping, and reactor vessel. Such a mixture can still provide adequate cooling. However, when the pumps were turned off in loop A the steam and water mixture in that loop and in the reactor vessel separated. With continued leakage through the open electromatic relief valve, there was insufficient coolant in the reactor vessel to cover the reactor core or circulate through the loops. At approximately 1.9 hours into the event the reactor coolant outlet temperature started to rise rapidly and by 2.5 hours the indications were off the high end of the instrument scale at 620°F. This recorded temperature remained off scale until approximately 10 hours into the accident, indicating a superheated steam environment (and presumably some noncondensible gas, mainly hydrogen) in the reactor coolant outlet piping.

Subsequent Events

At approximately 2.4 hours into the accident the operators closed the block valve which is in series with the open electromatic relief valve. Over the next thirteen hours the operators were trying to reestablish a stable cooling mode using either of the following options:

1. Either natural or forced circulation of reactor coolant with heat removal through the steam generators.

2. Use of the decay heat removal system, which requires that the reactor coolant system pressure be below 320 psig.

During a period of approximately five hours after closure of the block valve, numerous attempts were made to establish heat removal through the steam generators. Attempts to establish forced circulation or promote natural circulation of reactor coolant were unsuccessful due to the noncondensible gas trapped in the reactor coolant system. Reactor cooling system pressure varied widely during this period, in response to the position of the electromatic relief valve block valve and actuations of the high pressure injection system. A sustained attempt to reestablish heat removal through the steam generators by pressurizing the reactor coolant system to approximately 2100 psig with continuous operation of high pressure injection was unsuccessful.

Over the subsequent four hours, the operators reduced pressure in the reactor coolant system in an attempt to actuate the core flooding system and establish heat removal through the low pressure decay heat removal system. During the depressurization, a large fraction of the hydrogen was vented from the reactor coolant system and the core flood tanks injected some water directly into the reactor vessel; however, the reactor coolant system pressure remained too high to initiate cooling using the decay heat removal system.

When the operators were unable to depressurize the reactor coolant system any further, the block valve in series with the open electromatic relief valve was closed. During a subsequent period of almost two hours, there was no effective mechanism for removing all the generated heat from the reactor coolant system. The block valve was closed during this time, except for two brief periods. Coolant injection was apparently at a low rate and nearly equal to letdown thus providing limited cooling and both steam generators were isolated. At approximately 13-1/2 hours into the event a sustained high pressure injection of reactor coolant was initiated, repressurizing the reactor coolant system. When this injection was terminated, subsequent system behavior indicated that the steam generators were still blocked by hydrogen. However, the venting of a major portion of the hydrogen during the depressurization was sufficient to allow operation of a reactor coolant pump, thus reestablishing forced circulation of reactor coolant and subsequent heat removal through steam generator A.

Radioactivity

Coolant escaping from the ruptured disk of the reactor coolant drain tank and draining to the reactor building sump was low in radioactivity (consistent with only limited escape of fission gases from fuel elements to the coolant) shortly prior to two hours into the accident. Some of this coolant was transported to the auxiliary building and overflowed onto the floor. A radiation survey was made about half an hour later by the health physics foreman. He found that activity levels of the water on the floor in the auxiliary building basement were not above normal. A few minutes later, the radioactivity level in the area of the makeup tank in the auxiliary building was climbing toward 1 R/hour from a nominal 5 mr/hr. At approximately the same time, radioactivity monitoring instrumentation on a sample line from the reactor coolant letdown system indicated that the radioactivity level in that system was climbing rapidly. A "site emergency" was declared at 6:55 a.m. (2 hours and 55 minutes after turbine trip) and communicated to civil authorities.

About 3 hours into the event, the radiation monitor readings in the reactor building, the auxiliary building, and the fuel handling building started increasing rapidly and a "General Emergency" was declared at 7:24 a.m. (3 hours 24 minutes after turbine trip). However, all radiation monitors and radiation surveys off the site continued to indicate less than 1 mr/hr until after 5 hours into the accident. During the next hour and a quarter, radioactivity levels of 3 to 9 mr/hr were measured on the site, outside of the buildings.

There are several routes by which some amount of radioactivity may have escaped from the reactor building. The relative contributions of the several routes are not yet quantified. The possible routes include the piping between the reactor building sump and the auxiliary building sump tank, the piping associated with the reactor coolant letdown system, the vent header system, and the piping associated with venting and draining the reactor coolant drain tank.

Work is being done to establish the details of radioactivity releases from the plant. Enough work has been done to show conclusively that offsite doses were negligibly low.

Hydrogen Behavior

In order to promote recombination of radiolyticaly produced hydrogen and oxygen, excess hydrogen is routinely added to reactor coolant.* During the venting of coolant from the open electromatic relief valve, a small amount of hydrogen from this normal hydrogen inventory was released into the reactor building.

When the reactor core became uncovered, largely during the period from about 1-1/2 hours to about 3-1/2 hours after the start of the accident, portions of the fuel cladding reached temperatures which were high enough to allow the zircaloy cladding to react with steam and produce excess hydrogen. The measurements of total hydrogen inventory after the accident indicate that about one half of all the zircaloy in the core reacted with the reactor coolant.

*"Radiolysis" is the decomposition of water into hydrogen and oxygen under the influence of high energy radiation. In the presence of an excess of hydrogen, the recombination reaction is faster than the decomposition reaction, so no net production of hydrogen or oxygen occurs. All pressurized water reactors use hydrogen addition so as to create an excess sufficient to prevent net production of hydrogen or oxygen.

It is believed that during the first 16 hours, a portion of the hydrogen produced by the zircaloy-water reaction was trapped in the upper region of the reactor vessel above the inlet and outlet nozzles and remained there for the next 4 to 5 days. The hydrogen distributed itself between the gas-steam bubble in the reactor vessel, gas dissolved in the reactor coolant, and gas which escaped to the reactor building. After about 9-1/2 hours, the concentration of hydrogen in a region of the reactor building which contains an air atmosphere, became high enough to support combustion and ignited. A reactor building pressure pulse of about 28 pounds per square inch was recorded. This pressure is well within the design capability of the reactor building. The fact that a hydrogen burn occurred was later confirmed by chemical analyses which showed that the oxygen in the air atmosphere of the reactor building had been depleted by several percent.

The hydrogen gas bubble in the top of the reactor vessel was gradually removed from the reactor coolant system during the first several days by continuing letdown of coolant to the makeup tank, and by spraying coolant into the pressurizer and then venting the pressurizer. These two pathways for hydrogen removal, venting and letdown, take advantage of the variation of solubility of hydrogen in water with temperature and pressure.

Reactor Core

The in-core instrumentation which monitors core status indicated high temperatures and some intermittent voiding of the coolant in at least the upper region of the core early in the accident. The in-core thermocouples, located in the core outlet plenum above the core, and the self-powered neutron detectors, located within the core, indicated high temperatures beginning about 2.3 hours into the accident. One set of temperature measurements made between 4 and 5.5 hours into the accident indicates some temperatures in the region of 2500°F, but others were below 700°F around the outer edge of the core. All temperatures above the core center remained above 700°F (scale limit of the recording instrumentation) for about 8 hours into the accident, and some remained above 700°F for up to 30 hours.

The neutron detectors outside the reactor vessel at the core mid-plane indicate uncovering of part of the core from about 1.8 hours into the accident. The sequence of events, instrumentation responses, core exit thermocouple readings, the hydrogen produced, and the fission product escape to the reactor building all suggest extensive damage (oxidation) of fuel-rod cladding. Because there was subsequent reentry of cooling water to the hot core, it is believed that some fragmentation of the oxidized cladding and some of the uranium oxide fuel has occurred. The present evidence makes it appear that there was no general fuel melting. However, localized melting of some of the nonfuel materials in the core can reasonably be inferred. Despite the extensive damage to the core region, the transition to cooling of reactor coolant by natural circulation (on April 27, 1979), with relatively low core outlet temperatures measured since then, indicates a coolable configuration was established.

General Comments

This brief narrative is an introduction to the factual basis for the phenomena which were important in this event. It is in many respects an oversimplification of the accident and cannot be regarded as fully accurate. There were many factors present which affected the operation. For example, some of the data now available were not available, or else not in readily accessible form, during the accident. Procedures, training, regulatory requirements, design, external advice or directives, and command and control organization may have had significant roles at various times during the accident. These factors have not been extensively treated in this report because the primary objective of this report has been to analyze and verify the events and main phenomena which were the essence of the accident sequence. Analyses of portions or phases of the sequence of events of particular interest, with supporting data and calculations, are incorporated in this report.

NUCLEAR SAFETY ANALYSIS CENTER

SEQUENCE OF EVENTS

TMI 2 Accident - March 28, 1979

COMMENTARY

The intent of this sequence of events (SOE) prepared by the Nuclear Safety Analysis Center (NSAC) staff is to present a factual account of the event at the Three Mile Island, Unit 2, nuclear power plant on March 28, 1979. It is not meant to explain why the event occurred or to speculate on why certain actions were taken. This SOE has been verified, when possible, by raw data such as computer output, reactimeter data, strip charts, etc. Events have also been identified and verified by thermal hydraulic or core analysis. Entries concerning radiation readings were the only ones which were not substantiated by raw data because of the unavailability of readable, reproduced or original, strip charts at NSAC. Limited use was made of interviews by others with persons involved in the event. NSAC had no first-hand interviews with TMI-2 operators on duty at the time. Commentary as to why certain actions were taken have been generally omitted, except by inference. Certain actions may have been omitted if they could not be confirmed, even though they have been reported by other sources, as long as these actions did not appear to play an important role in the event.

Appendices have been prepared to further explain the actions and remarks, and to give a narrative description of the events from a thermal hydraulic and core assessment viewpoint. System related appendices are not meant to be independent system descriptions, but are written to clarify or amplify remarks in the SOE.

In some cases, it has been difficult to interpret the available data and the interpretation is still in progress. An example of this is the interpretation of make up pumps status alarms and engineered safeguard actuation status alarms.

NUCLEAR SAFETY ANALYSIS CENTER

SEQUENCE OF EVENTS

TMI 2 Accident - March 28, 1979

Plant Status Prior to Start of Event

TMI Unit 2 was operating at 97% power with the integrated control system (ICS) in full automatic. The reactor coolant system was operating with four reactor coolant pumps at a pressure of 2155 psig. Reactor coolant makeup pump 1B was in service providing normal makeup and reactor coolant pump seal injection flow. Reactor coolant system letdown flow was approximately 70 gpm. The reactor coolant system boron concentration was approximately 1030 parts per million. Rod groups one through five were fully withdrawn, rod groups six and seven were 95% withdrawn and rod group eight was 27% withdrawn. Reactor coolant system leakage was approximately 6 gpm. The condensate system was operating with two condensate pumps and two condensate booster pumps in service. Both turbine driven feed pumps were in service. The pressurizer spray valve and the pressurizer heaters were in manual control while spraying the pressurizer to equalize the boron concentrations between the pressurizer and the rest of the reactor coolant system. This equalization was necessary because of steam leakage past either the electromatic relief valve or the pressurizer safety valves. This leakage was evidenced by periodic safety valve discharge header temperature alarms.

Operators were experiencing difficulties in transferring resins from an isolated condensate polisher to the receiving tank. Attempts to free the plugged transfer line had been in progress for about eleven hours.

TMI Unit 1 was in hot shutdown for low power physics testing following refueling. There was a vacuum on the condenser and auxiliary steam was being supplied from Unit 2.

TIME	EVENT	REMARKS & REFERENCES
-00:00:01 (0400:36)	Condensate pump 1A & 1B (CO-P-1A) tripped.	Ref. 3.a., Appendix C/FDW, p.3.

S T A R T   O F   E V E N T

TIME	EVENT	REMARKS & REFERENCES
00:00:00 (0400:37)	Feedwater pumps 1A & 1B (FW-P-1A & FW-P-1B) tripped.	Ref. 3.a., 1.r, 1.s. The trips were caused by low feedwater pump suction pressure. Feedwater flow was lost to both steam generators.
00:00:00 (0400:37)	The main turbine tripped.	Ref. 3.a. The turbine trip results automatically from the trip of both feedwater pumps.
00:00:00 (0400:37)	Three emergency feedwater pumps 1, 2A & 2B (EF-P-1, EF-P-2A, and EF-P-2B) started.	Ref. 3.a. Automatic start of these pumps is caused by the trip of both feedwater pumps. Emergency feedwater pump 1 (EF-P-1) is a steam turbine driven pump, and emergency feedwaer pumps 2A & 2B (EF-P-2A and EF-P-2B) are electric motor driven pumps.
00:00:03 (0400:40)	The pressure setpoint (2255 psig) of electromatic relief valve (ERV) (RC-2) located on the pressurizer was exceeded.	Ref. 1.k, 5.b. The rate-of-change of the reactor coolant system pressure indicated that the relief valve lifted.

TIME	EVENT	REMARKS & REFERENCES
00:00:04 (0400:41)	Pressure began increasing in the reactor coolant drain tank.	Ref. 1.j. This was another indication that the ERV lifted. Reactor coolant drain tank pressure at this time was approximately 4 psig.
00:00:06 (0400:43)	Secondary side steam pressure reached 1074 psig in steam generator A and 1052 psig in steam generator B.	Ref. 1.u, 1.v. Pressures from the reactimeter charts indicated that some main steam safety valves lifted.
00:00:08 (0400:45)	The reactor tripped on high reactor coolant system pressure. The nominal trip setpoint is 2355 psig.	Ref. 3.a, 1.k, 11 - Sect. 2.2.1, Appendix TH, p. 27. Reactor coolant system pressure reached 2344 psig at 0400:46, as indicated on the reactimeter. The higher reading of approximately 2435 psig observed on the wide range reactor coolant system stripchart could be caused by a 50 to 60 psig higher reading at the start of the event coupled with pen overshoot during the rapid pressure increase.
00:00:08 (0400:45)	Pressurizer heater groups 1 through 5 indicated off.	Ref. 3.a, 2.e - p. 2, Appendix RCPCS - pp. 2-5. This event could signify the operator's reported action of placing the pressurizer heater control in the automatic mode to mitigate expected reactor coolant system pressure transients following any reactor trip.

TIME	EVENT	REMARKS & REFERENCES
00:00:12 (0400:49)	Reactor coolant system pressure decreased below the setpoint value for ERV (RC-R2) closure.	Ref. 1.k, 5.b, Appendixes ERV - p. 1, RCPCS - pp. 3, 6-7. The ERV should have reseated (closure setpoint was 2205 psig), but it remained in the full open position.
00:00:12 (0400:49)	Indicated pressurizer coolant level peaked at 256 in. and began a rapid decrease.	Ref. 1.g. A momentary coolant insurge followed by a rapid coolant outsurge is a normal event following a reactor trip.
00:00:12 (0400:49) Approximate	Letdown flow was stopped.	Ref. 3.d, 2.d, - p. 1, 2.e - p. 2, Appendix HPI - pp. 1, 3, 6. This step and the next are initial actions for operators to take after reactor trip accidents to compensate for the expected reduction of pressurizer level.
00:00:13 (0400:50)	One or more attempts were made to start makeup pump 1A (MU-P-1A). The pump did not start.	Ref. 3.a., 3.d, 2.c - p. 2, 2.d. - p. 2, Appendix HPI - pp. 1, 6, 7.
00:00:13 (0400:50)	A condenser hotwell low water level alarm was received on the alarm typewriter. (Nominal alarm setpoint was 22.5 in.)	Ref. 3.a, Appendices C/FDW - p. 5, PDS - pp. 12-13. The level was 21.72 in. Because of the 15 second scan interval of this parameter, the actual alarm may have been received before this time.

TIME	EVENT	REMARKS & REFERENCES
00:00:14 (0400:51)	The emergency feedwater pumps (EF-P1, EF-P-2A, and EF-P-2B) low pressure alarms cleared on the alarm typewriter.	Ref. 3.a, Appendix PDS - pp. 12-13. EF-P1 1237 psig EF-P2-A 1471 psig EF-P2-B 1445 psig Because of the 15 second scan interval of these parameters, these pressures were probably reached before the time printed out.
00:00:14 (0400:51)	Pressurizer heater goups 1 through 5 indicated on.	Ref. 3.a, 2.e - p. 2, Appendix RCPCS - pp. 2, 3-5. A low reactor coolant system pressure with the heater control in automatic would cause the heaters to energize.
00:00:15 (0400:52)	Steam generator A water level indicated 74 in. on the startup range. Steam generator B water level indicated 76 in. on the startup range.	Ref. 1.q, 1.x, Appendix OTSG - pp. 2-3. Steam generator water levels were decreasing.
00:00:15 (0400:52)	The pressurizer spray valve (RC-V1) closed).	Ref. 1.i, 2.e - p. 2, Appendix RCPCS - pp. 5-6. This closure indicated that the spray valve was responding normally to reactor coolant system pressure transients and supports the operator's reported action of placing the pressurizer spray in automatic.

TIME	EVENT	REMARKS & REFERENCES
00:00:28 (0401:07)	The condenser hotwell low water level alarm cleared. (Nominal alarm setpoint was 22.5 in.)	Ref. 3.a, Appendix C/FDW - p. 5. The level was 26.44 in. The computer scan interval was 15 sec.
00:00:30 (0401:07)	The ERV (RC-R2) and pressurizer safety valve (RC-R1B) outlet temperatures alarmed high.	Ref. 3.c, Appendix ERV - pp. 3-4. RC-R2 292.2°F RC-R1B 203.5°F RC-R1B outlet temperature had been operating close to its high alarm setpoint prior to the reactor trip. Computer scan interval was 30 sec.
00:00:30 (0401:07)	The reactor coolant system low pressure trip setpoint was reached.	Ref. 1.q, 5.b, Appendices OTSG - pp. 2-4, PDS, p.2. Emergency feedwater valves EF-V11A & EF-V11B should open when levels reach 30 inches. Feedwater was not admited to the steam generators. Emergency feedwater block vlves EF-V12A and EF-V12B, which should have been open, were closed. This fact was not recognized by the operators at this time.
00:00:33 (0401:10)	Steam generator B water level decreased to 28.8 in. in the startup range.	Ref. 1.x, 5.b, Appendices OTSG - pp. 2-3, PDS - p.2. See entry at 0401:07.
00:00:41 (0401:18)	Makeup pump 1A (MU-P-1A) was started.	Ref. 3.a, 1.g. With makeup pumps 1A & 1B operating, the pressurizer level rate of decrease slowed.

TIME	EVENT	REMARKS & REFERENCES
00:00:48 (0401:25)	Pressurizer coolant reached an indicated minimum level of 158 in.	Ref. 1.g. Indicated pressurizer coolant level began to increase.
00:00:58 (0401:35)	A pressurizer low coolant level alarm was received. (Nominal alarm setpoint was 200 in.)	Ref. 3.a, Appendix PDS - pp. 2, 15, 12-13. Because of the scan intervals and computation involved with pressurizer coolant level, this alarm typewriter entry lagged the actual pressurizer low level.
00:01:00 (0401:37)	Pressurizer safety valve (RC-R1A) outlet temperature alarmed high.	Ref. 3.a, Appendix RCPCS - pp. 3, 5-6. RC-R1A 294.5°F
00:01:13 (0401:50)	A condenser hotwell high water level alarm was received. (Nominal alarm setpoint was 36 in.)	Ref. 3.a, 2.C - p. 3, 2.n - p. 2, Appendix C/FW - p. 5. Indicated level was 37.77 in. The computer scan interval was 15 sec.
00:01:26 (0402:03)	The indicated reactor coolant drain tank liquid temperature was 85.5°F.	Ref. 3.a. This alarm typewriter entry indicated that the reactor coolant drain tank liquid returned to a normal temperature range. Because of the heat addition to the tank at this time, it is probable that it was a low temperature alarm that had cleared.

TIME	EVENT	REMARKS & REFERENCES
00:01:45 (0402:22) Approximate	Both steam generators boiled dry on the secondary side.	Ref. 1.b, 1.c, 1.d, 1.e, 1.u, 1.v. This event was indicated by a steadily decreasing steam generator secondary side pressure while reactor coolant hot and cold leg temperatures were increasing. Effective heat transfer from the primary coolant to the secondary system is stopped when the steam generators are dry.
00:02:01 (0402:38)	Engineered safety features for high pressure injection actuated.	Ref. 3.a, 2.c - p. 2, 2.e - p. 4, 11 Sec. 3.3.2.1, Appendices ESF - pp. 6, 7, 9, HPI - p. 9. This was caused by reactor coolant system pressure dropping below 1640 psig.
	Makeup pump 1B (MU-P-1B) tripped.	Ref. 3.a, 2.g - p. 2, Appendix ESF - pp. 14-15. This is an automatic trip before high pressure injection pumps 1A & 1C (MU-P-1A and MU-P-1C) start.
through		
	High pressure injection pump 1C (MU-P-1C) started automatically.	Ref. 3.a, 2.g - p. 2, Appendices ESF - pp. 9, 14, HPI - p. 9. Pump 1A (MU-P-1A) was already operating.

TIME	EVENT	REMARKS & REFERENCES
00:02:04 (0402:41)	Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) started.	Ref. 3.a, Appendix ESF - pp. 2, 16. These pumps were running in a recirculation mode and were not injecting water into the reactor coolant system.
00:03:13 (0403:50)	The high pressure injection portion of engineered safety features was manually bypassed.	Ref. 3.a, 2.c - p. 2, 2.g - p. 2, Appendices ESF - p. 12, HPI - p. 9. This action permits throttling discharge valves of the high pressure injection pumps and/or shutting down the high pressure injection pumps.
00:03:13 (0403:50)	The reactor coolant drain tank relief valve (WDL-R1) lifted at approximately 122 psig. Nominal setpoint is 150 psig.	Ref. 1.j. This is an inferred event based on the reactor coolant drain tank pressure history. A lifted pressurizer relief or safety valve could cause the reactor coolant drain tank relief valve to lift.
00:03:26 (0404:03)	A reactor coolant drain tank high temperature alarm was received.	Ref. 3.a. This is another indication of a lifted pressurizer relief or safety valve. Indicated temperature was 127°F. Computer scan interval was 30 sec.
00:03:28 (0404:05)	A pressurizer high coolant level alarm was received. (Nominal alarm setpoint was 260 in.)	Ref. 3.a, 1.g, Appendices HPI - p. 4, PDS - pp. 2, 4, 12-13. The pressurizer coolant level as indicated by the reactimeter was 292.4 in. Computer scan interval was 15 sec.

TIME	EVENT	REMARKS & REFERENCES
00:04:38 (0405:15)	The operator stopped makeup pump 1C (MU-P-1C) and throttled the high pressure injection isolation valves (MU-V16A & MU-V16B).	Ref. 3.a, Appendix HPI - pp. 4, 9, 2.d - p. 5, 2.m - p. 1. The pressurizer coolant level rate of rise decreased. Hot leg temperatures started to increase, followed by an increase in cold leg temperatures.
00:04:52 (0405:29)	Intermediate closed cooling pump 1A (IC-P-1A) started.	Ref. 3.a. This is a normal procedural step prior to initiating high letdown flow.
00:04:58 (0405:35)	An alarm was received that letdown flow was off scale.	Ref. 3.a, 1.h, 2.c - p. 3, 2.m - p. 2. This alarm could indicate that letdown flow was off scale high, greater than 160 gpm.
00:05:00 (0405:37)	Pressurizer coolant level reached 377 in., decreased momentarily, then continued to rise.	Ref. 1.g.
00:05:15 (0405:52)	Condensate pump 1A (CO-P-1A) was started.	Ref. 3.a, Appendix C/FDW - p. 3.
00:05:15 (0405:52) through 00:06:29 (0407:06)	A condensate booster pump 2B (CO-P-2B) trip signal was received three times, followed by a trip clearing (normal) signal in each instance.	Ref. 3.a, Appendix C/FDW - p. 4. Operators were trying to reestablish secondary plant operating conditions. The inability to start the booster pump was apparently caused by a low suction pressure trip when the pump started turning.

TIME	EVENT	REMARKS & REFERENCES
00:05:30 (0406:07) Approximate	The indicated reactor coolant system hotleg temperature and pressure reached saturation conditions of 582°F and 1340 psig.	Ref. 3.d, 4.f, 1.b, 1.c.
00:05:46 (0406:23)	There were indications of a liquid discharge from the ERV.	Ref. 1.j. This was evidenced by a rapid increase in the reactor coolant drain tank pressure which indicated increased mass flow corresponding to liquid flow from the ERV.
00:05:51 (0406:28)	Pressurizer coolant level indication went off scale high, greater than 400 in.	Ref. 1.g.
00:06:54 (0407:31)	A letdown cooler 1A high temperature alarm was received. (Nominal alarm setpoint was 135°F.)	Ref. 3.a, 3.d, 1.h, Appendix HPI - p. 12. This alarm could be associated with high letdown flow. Indicated temperature was 139°F. The computer scan interval was 30 sec. Letdown flow is automatically isolated when this alarm is received by closure of the letdown isolation valve (MU-V-376).
00:06:58 (0406:35)	Letdown flow came back on scale and indicated 71.4 gpm.	Ref. 3.a, Appendix HPI - p. 12. Letdown flow was decaying following closure of the letdown isolation valve. Computer scan interval was 30 sec.
00:07:29 (0408:06)	Reactor building sump pump 2A (WDL-P-2A) started.	Ref. 3.a, 2.c - p. 4, Appendices SP, ROUTES - p. 2.

TIME	EVENT	REMARKS & REFERENCES
00:08:18 (0408:55)	Emergency feedwater block valves 12A & 12B (EF-V12A and EF-V12B) were opened.	Ref. 1.b, 1.c, 1.d, 1.e, 1.q, 1.u, 1.v & 1.x, 2.c - p. 2, 2.d - p. 3, 2.e - p. 3, 2.n - p. 1. Normally these valves should have been open. Opening them admitted water to the steam generators. Increases in steam generator levels and pressures followed by responses of hot and cold leg temperatures indicate feed was established to the steam generators.
00:08:30 (0409:07) Approximate	The reactor coolant system pressure began to decrease further.	Ref. 4.d, Appendix OTSG - p. 3. The initiation of cold emergency feedwater to the steam generator resulted in rapid decreases of the reactor coolant system temperatures and pressures.
00:08:58 (0409:35)	Condensate pump 1A (CO-P-1A) tripped.	Ref. 3.a, Appendix C/FDW - p. 3.
00:09:13 (0409:50)	A condensate booster pump suction header low pressure alarm was received.	Ref. 3.a, Appendix C/FDW - p. 4. The pressure was 14.7 psig.
00:09:23 (0410:00) Approximate	Letdown flow isolation valve (MU-V-376) was opened.	Ref. 3.d, 1.h.

TIME	EVENT	REMARKS & REFERENCES
00:10:15 (0410:52)	Pressurizer coolant level indication came back on scale, less than 400 in.	Ref. 1.g. The initiation of emergency feedwater to the steam generators could result in coolant outsurges from the pressurizer.
00:10:19 (0410:56)	Reactor building sump pump 2B (WDL-P-2B) started.	Ref. 3.a. Appendix SP.
00:10:24 (0410:56)	Letdown cooler outlet temperature returned to its normal range and the alarm cleared. (Nominal alarm setpoint was 135°F.)	Ref. 3.a. The temperature was 123.7°F indicating that letdown flow had been reestablished. The computer scan interval was 30 sec.
00:10:24 (0411:01) through 00:11:43 (0412:20)	Makeup pump 1A (MU-P-1A) tripped and was restarted three times.	Ref. 3.a. The pump was operating at the end of this sequence.
00:10:48 (0411:25)	A reactor building sump high level alarm was received.	Ref. 3.a, Appendix SP. Indicated level was 4.65 feet.
00:13:13 (0413:50)	Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) were shut down.	Ref. 3.a, Appendix ESF - p. 2. These pumps served no function at that time.

TIME	EVENT	REMARKS & REFERENCES
00:13:27 (0414:04)	The condensate booster pump suction header low pressure alarm cleared. (Nominal alarm setpoint was 15 psig.)	Ref. 3.a. The pressure was 17 psig. The computer scan interval was 15 sec.
00:14:48 (0415:25)	The reactor coolant drain tank rupture disc (WDL-U26) failed at 191.6 psig.	Ref. 1.j, Appendix ROUTES - p. 1. Design failure pressure is 200±25 psig. Discharge through the ruptured disc was to the reactor building atmosphere.
00:14:50 (0415:27)	At this time, reactor coolant pump related alarms began coming in on the alarm type-writer.	Ref. 3.a. These alarms could indicate abnormal reactor pump operation caused by degraded conditions in the reactor coolant system and reactor build-ing. Reactor coolant flow had been steadily decreasing since the beginning of the event.
00:15:43 (0416:20)	The condensate booster pumps low discharge pressure alarm was received.	Ref. 3.a, Appendix C/FDW - p. 5. Discharge pressure was 307 psig.
00:16:12 (0416:49)	A condensate booster pump suction header low pressure alarm was received. (Nominal alarm setpoint was 310 psig.)	Ref. 3.a, Appendix C/FDW - p. 4. Indicated pressure was 14.8 psig. The computer scan interval was 15 sec.

TIME	EVENT	REMARKS & REFERENCES
00:19:23 (0420:00) Approximate	Reactor Building Purge Air Exhaust Duct A Monitor (HP-R-225) particulate channel count rate increased from 1×10^2 to 5×10^2 cpm. Slight increases were also indicated on the Duct B Monitor (HP-R-226) and on HP-R-222 (before the filter) and HP-R-228 (after the filter).	Ref. 12.a, Appendix RM - p. 3. These indications appear to be the result of the reactor coolant drain tank rupture disc blowout.
00:20:00 (0420:37) Approximate	The indicated source range neutron flux signal departed from the expected normal flux decay for a reactor trip.	Ref. 4.f. This departure was caused by the buildup of steam voids in the system. Source range count rate leveled and then began a gradual increase.
00:22:17 (0422:54)	The operator depressed the reactor trip push-button.	Ref. 3.a, Appendix CI - p. 12. This precautionary action was in response to the observed abnormalities in neutron flux levels.
00:22:44 (0423:21)	Steam generator A water level increased to approximately 30 in. in the startup range. The low level alarm cleared.	Ref. 3.a, 1.q, Appendix OTSG - pp. 2-3. The design condition for heat removal from the steam generator was reestablished.
00:24:58 (0425:35)	The ERV outlet temperature was 285.4°F. Safety valves R1A and R1B read 263.9° and 275.1° respectively.	Ref. 3.c, Appendix RCPCS - pp. 3, 6-7. This was operator requested information.

TIME	EVENT	REMARKS & REFERENCES
00:25:44 (026:21)	An emergency feedwater pump 1 (EF-P-1) low discharge pressure alarm was received.	Ref. 3.a. Discharge pressure was 9 psig. This could indicate that emergency feed pump 1 was shut down. No other status indication was available for this pump.
00:26:26 (0427:03) through 00:27:51 (0427:28)	RC Loop A outlet temperature 551.9°F RC Loop B outlet temperature 550.9°F RC Loop A inlet temperature 548.1°F RC Loop A inlet temperature 547.0°F RC Loop B inlet temperature 547.0°F RC Loop B inlet temperature 546.8°F RC Loop A wide range pressure 1040 psig. RC Loop B wide range pressure 1043 psig.	Ref. 3.c, Appendix PDS - pp. 1-3, 13-14. This was operator requested plant status information on the utility typewriter. Reactimeter temperatures are in close agreement with these values.
00:26:46 (0427:33)	Steam generator B water level increased to approximately 28 in. in the startup range. The low level alarm cleared.	Ref. 3.a, 1.x, Appendix OTSG - pp. 2-3. Conditions for effective heat removal from the steam generator were reestablished.

TIME	EVENT	REMARKS & REFERENCES
00:29:23 (0430:00) through 00:49:23 (0450:00) Approximate	Reactor Building Air Sample Monitor (HP-R-227) gas channel count rate increased from $1x10^3$ to $5x10^4$ cpm and then decreased to $1x10^3$ cpm.	Ref. 12.a, Appendix RM - p. 3.
00:29:56 (0430:33)	Both emergency diesel generators were manually tripped during the next 30 sec.	Ref. 3.a.
00:32:23 (0433:00) Approximate	Radiation readings of the following monitors increased and then leveled off: gas channel of Station Vent (HP-R-221A, -221B) H_2 Purge Duct particle and iodine channels (HP-R-229).	Ref. 12.a, Appendix RM - p. 3.
00:32:36 (0433:13)	Incore thermocouple (Location 10-R) indicated off scale.	Ref. 3.a, Appendix CI - p. 16. Top of the scale is 700°F.
00:36:08 (0436:45)	Emergency feedwater pump 2B (EF-P-2B) was shut down.	Ref. 3.a, 1.q, 1.x. Steam generator A water level indicated 35.3 in. in the startup range. Steam generator B water level indicated 40.3 in. in the startup range. These levels are normal for shutdown conditions. Emergency feedwater pump 2A was running to maintain shutdown levels in the steam generators.

TIME	EVENT	REMARKS & REFERENCES
00:38:10 (0438:47)	Reactor building sump pump 2A (WDL-P-2A) was stopped.	Ref. 3.a, 2.c - p. 4, 2.f - p. 4, Appendix SP. This pump ran for approximately 31 min.
00:38:11 (0438:48)	Reactor building sump pump 2B (WDL-P-2B) was stopped.	Ref. 3.a, 2.c - p. 4, 2.f - p. 4. All pumping of water from the reactor building to the auxiliary building was stopped. This pump ran for approximately 28 min.
00:40:00 (04:40:37) Approximate	An increasing count rate continued to be indicated on the source range neutron detector.	Ref. Appendix CI - p. 12. This increase was caused by the decreasing density of the coolant passing through the reactor downcomer annulus which shields the core.
00:46:23 (0447:00) Approximate	Letdown Cooler A Monitor (IC-R-1092) count rate began increasing from approximately 2000 cpm and reached over 2×10^4 cpm about 40 minutes later.	Ref. 12.a, 2.c - p. 4, 2.e - p. 7, Appendices RM - p. 1, ROUTES - p. 3.
00:59:12 (0459:49)	The condensate booster pump suction header low pressure alarm cleared. (Nominal alarm setpoint was 15 psig.)	Ref. 3.a, Appendix C/FDW - p. 4. Pressure was 89.2 psig. The computer scan interval was 15 sec.
00:59:21 (0459:58)	A condensate high temperature alarm was received.	Ref. 3.a, Appendix C/FDW - p. 4-5. Indicated temperature was 118.5°F.

TIME	EVENT	REMARKS & REFERENCES
01:00:49 (0501:26)	Condenser circulating water pumps 1B, 1C, 1D, & 1E (CW-P-1B, CW-P-1C, CW-P-1D, & CW-P-1E) were shut down.	Ref. 3.a, 2.d - p. 9, 2.f - p. 2, 2.n - p. 4, Appendix STEAM DUMP - pp. 2-3, 5. Steam flow control was shifted from the turbine bypass valves to the atmospheric dump valves.
01:13:23 (0514:00)	The alarm history was lost until 0648:08.	Ref. Appendix PDS - pp. 12-13.
01:13:29 (0514:06)	Reactor coolant pump 2B (RC-P-2B) was stopped.	Ref. 3.f, 1.m, 2.c - p. 5, 2.d - p. 5, 2.e - p. 5, 2.n - p. 6. Secondary side steam pressure in loop B began to drop sharply, indicating stagnation in reactor coolant system loop B flow.
01:20:31 (0521:08) through 01:20:58 (0521:35)	Pressurizer Relief and Safety Valve T Temperatures ERV RC RV2 - 283°F Safety valve RC R1A - 211°F Safety valve RC R1B - 218°F	Ref. 3.f, Appendix ERV - pp. 3-5, RCPCS - pp. 3, 6-7. This was an operator requested computer printout of relief and safety valve outlet temepratures. It was not recognized that the ERV temperature indicated that the relief valve was open.
01:30:00 (0530:37) Approximate	Reactor out-of-core intermediate range neutron instrumentation channel NI-3 came on scale and began increasing.	Ref. 4.f, 2.d - p. 6, Appendix CI - p. 12. This response was consistent with the steadily increasing source range count rate.

TIME	EVENT	REMARKS & REFERENCES
01:30:40 (0531:17)	There was a marked increase in secondary side steam flow from steam generator A.	Ref. 1.q, 1.a. The combination of changes in steam generator secondary side pressure and water level indicates steam flow.
01:31:22 (0531:59)	Secondary side steam flow from steam generator A decreased rapidly.	Ref. 1.u. The combination of changes in steam generator secondary side pressure and water level indicates steam flow.
01:32:04 (0532:41)	Feedwater flow to steam generator B was increased.	Ref. 1.x. This served no apparent purpose because flow in the reactor coolant system loop B had been stopped.
01:32:19 (0532:56)	Steam generator A indicates dryout on the secondary side.	Ref. 1.q, 1.u. The conditions for effective heat removal using the steam generators had been lost because there was no reactor coolant system flow in loop B and steam generator A was dry.
01:34:10 (0534:47)	Feedwater flow to steam generator A was increased.	Ref. 1.u, 2.c - p. 6, 2.d - p. 5. This was an apparent effort to regain conditions for heat transfer from the reactor coolant system using the steam generators.
01:34:16 (0534:53)	Loop A cold leg temperatures started to decrease.	Ref. 1.d. This was an indication of reestablishment of heat transfer from the reactor coolant system.

TIME	EVENT	REMARKS & REFERENCES
01:34:16 (0534:53)	Feedwater flow to steam generator B was reduced.	Ref. 1.x, 2.c - p. 5.
01:40:37 (0541:14)	Reactor coolant pump 2A (RC-P-2A) was stopped.	Ref. 3.f, 1.f., 2.c - p. 5, 2.e - p. 5. Appendix PDS - pp. 1-2. The reactimeter showed a marked decrease in reactor coolant flow beginning at 0540:57.
01:40:45 (0541:22)	Reactor coolant pump 1A (RC-P-1A) was stopped.	Ref. 3.f, 1.f, 2.c - p. 5, 2.3 - p. 5, Appendix PDS - pp. 1, 2. At this point there was no forced reactor coolant system flow. The reactimeter showed a marked flow decrease beginning at 0541:08.
01:41:00 (0541:37) Approximate	Out-of-core neutron instrumentation indicated a decreasing flux level.	Ref. 4.f, Appendix CI - p. 13. This temporary decrease was caused by a coolant phase separation with the liquid filling the downcomer annulus. The phase separation is attributed to stopping the reactor coolant pumps and the resultant flow coastdown.
01:42:00 ((0542:37)	Steam generator B was isolated on the secondary side.	Ref. 1.v, 1.x, 2.c - p. 5, 2.m - p. 7, Appendices OTSG - p. 4, TH - p. 52. A leak was suspected because of difficulties with water level control in steam generator B and a 300 psi pressure differential between steam generators A and B.

TIME	EVENT	REMARKS & REFERENCES
01:42:25 (0543:02)	Reactor coolant system loop A cold leg temperature stopped decreasing and small temperature oscillations began.	Ref. 1.d. This could indicate stopping circulation of coolant through the loops.
01:42:30 (0543:07) Approximate	Out-of-core nuclear instrumentation indicated increased flux levels.	Ref. 4.f, Appendix CI - p. 13. These indicated higher flux levels could be caused by the boil-off of the coolant in the core and the resultant decrease in coolant level in the downcomer annulus.
01:51:27 (0552:04) through 02:29:18 (0629:55)	Loop A and B hot leg temperatures were increasing and continued upward until they went off scale high, greater than 620°F.	Ref. 1.b, 1.c, 2.c - p. 6, Appendix TH - p. 53. These temperatures and the reactor coolant system pressure indicate the presence of superheated steam in the reactor coolant system.
02:03:57 (0604:34)	An operating range water level of 51% was established and subsequently maintained in steam generator A.	Ref. 1.p, 1.q, 2.c - p. 6. This is a requirement for establishing natural circulation in the reactor coolant system.

TIME	EVENT	REMARKS & REFERENCES
02:14:23 (0615:00) Approximate	The Reactor Building Air Sample Monitor (HP-R-227) particulate channel radiation reading increased and eventually went off scale high. The gas channel radiation reading began to increase at 0625 and then went off scale high. Iodine channel count rate began to increase about 0645 and went off scale high.	Ref. 12.a, Appendix RM - p. 3.
02:15:00 (0615:37)	Self powered neutron detector readings began to rise rapidly.	Ref. 4.h, Appendix CI - p. 21. Because the core was partially uncovered, the self powered neutron detectors were responding as local temperature detectors which indicated high core temperatures. This response was because the self powered neutron detectors are also thermionic emitters that generate a measurable signal at high temperature.
02:17:53 (0618:30)	ERV (RC-R2) outlet temperature was 228.7°F.	Ref. 3.e, 2.r - p. 1, Appendix ERV - pp. 3-5. The ERV outlet temperature was operator requested.

TIME	EVENT	REMARKS & REFERENCES
02:22:00 (0622:37) Approximate	The ERV block valve (RC-V2) was closed. Reactor coolant system pressure began to increase.	Ref. 4.a, 4.d., 2.c - p. 5, 2.m - p. 6, 2.4 - p. 2. Reactor coolant leakage through the ERV was stopped. Decreasing reactor building pressure and increasing reactor coolant system pressure indicated the primary source of system leakage was through the ERV.
02:33:27 (0634:04)	Increasing the level in steam generator B was begun.	Ref. 1.w, 1.x, Appendix OTSG - pp. 2-3. Steam generator B level was 5% on the operating range.
02:34:23 (0635:00) Approximate	An additional makeup pump was started.	Ref. Appendix TH - p. 60.
02:38:23 (0639:00) Approximate	The Makeup Tank Area Monitor (HP-R-206) (in the auxiliary building), Fuel Handling Bridge South Monitor (HP-R-210) in the reactor building, and Reactor Building Dome Monitor (HP-R-214) radiation readings began to increase.	Ref. 12.a, Appendix RM - p. 2.
02:38:23 (0639:00) Approximate	Letdown Cooler A Radioactivity Monitor (IC-R-1092) readings pegged off scale high.	Ref. 12.a, Appendix RM - p. 1.

TIME	EVENT	REMARKS & REFERENCES
02:39:23 (0640:00) Approximate	Two boron analyses indicating boron concentrations in the reactor coolant system of approximately 400 ppm were received by the shift supervisor. Emergency boration of the reactor coolant system was started.	Ref. 2.a - p. 3, 2.b - p. 2, 2.m - p. 5, 4.f. These analyses, in conjunction with the increased neutron level indications, prompted emergency boration to maintain the reactor subcritical.
02:44:23 (0645:00) Approximate	The In-core Instrument Panel Area Radioactivity monitor (HP-R-213) (in the reactor building) radiation readings increased off scale high.	Ref. 12.a, Appendix RM - p. 2.
02:44:23 (0645:00) Approximate	Reactor Building Purge Air Exhaust Ducts A & B Monitors (HP-4-225 & HP-4-226) particulate and gas channels radiation readings increased rapidly.	Ref. 12.a, Appendix RM - p. 3.
02:44:23 (0645:00) Approximate	The makeup pump started at 0635 was stopped.	Ref. Appendix TH - p. 61.
02:45:23 (0646:00) Approximate	Fuel Handling Storage Area Monitor (HP-R-218) radiation readings began increasing.	Ref. 12.a, Appendix RM - p. 2.

TIME	EVENT	REMARKS & REFERENCES
02:46:23 (0647:00)	An attempt was made to start reactor coolant pump 1A (RC-P-1A).	Ref. 2.a - pp. 3 & 13, 2.c - p. 6, 2.d - p. 6. The pump did not start.
02:47:31 (0648:08)	Current alarm typewriter indications showed that self-powered neutron detectors were responding to high temperatures down to the four foot level of the core (approximate). Ninety percent of the core exit thermocouples were reading in excess of 700°F.	Ref. 3.g, Appendix CI - p. 16. These were the first alarms that were received following the period when alarm history was lost.
02:49:23 (0650:00) Approximate	The radiation readings of all channels of the following monitors increased steadily and by 0721 were off-scale: Station Vent (HP-R-219), Fuel Handling Building Exhaust Duct (HP-R-221 A & B), Hydrogen Purge (HP-R-229). The radiation readings of all channels of Control Room Intake Monitor (HP-R-220) remained below 10 cpm from 0000 hours March 28 until 0950 hours March 28.	Ref. 12.a, Appendix RM - p. 3.

TIME	EVENT	REMARKS & REFERENCES
02:49:23 (0650:00) through 02:59:23 (0700:00) Approximate	Condenser Vacuum Pump Exhaust Radiation Monitor (VA-R-748) radiation readings increased rapidly from 1×10^2 to 8×10^5 cpm. At approximately 0850 this monitor's radiation reading decreased to 1×10^4 cpm.	Ref. 12.a, Appendix RM - p. 3. This monitor, located in the turbine building on the 281 ft - 6 in level, samples condenser exhaust from the vacuum pumps, and the readings could be indicative of a primary to secondary leak.
02:51:57 (0652:34)	The operator attempted to start reactor coolant pump 2A (RC-P-2A).	Ref. 3.g, 2.d - p. 6, 2.a - pp. 3 & 13, 2.c - p. 6. The pump would not start. Indications show that preliminary steps were taken to start reactor coolant pump 2A (RC-P-2A).
02:52:30 (0653:07)	The condenser hotwell high water level alarm cleared. (Nominal alarm setpoint was 36 in.)	Ref. 3.g. Indicated level was 34.94 in. The computer scan interval was 15 sec.
02:53:16 (0653:53)	The operator attempted to start reactor coolant pump 1B (RC-P-1B).	Ref. 3.g, 2.d - p. 6, 2.a - pp. 3 & 13, 2.c - p. 6. The pump would not start. Indications show that preliminary steps were taken to start reactor coolant pump 1B (RC-P-1B).
02:54:09 (0654:46)	The operator started reactor coolant pump 2B (RC-P-2B). Flow was indicated for only a few seconds and then returned to 0.	Ref. 3.g, 1.d, 1.e, 2.d - p. 6, 2.a - pp. 4 & 13, 2.c - p. 6, 2.e - p. 6, Appendix TH - pp. 61 - 63. Forced reactor coolant system flow was reestablished. Reactor coolant pump 2B (RC-P-2B) was running with high vibration alarm.

TIME	EVENT	REMARKS & REFERENCES
02:54:15 (0654:52) Approximate	The reactor out-of-core nuclear instrumentation showed sharp neutron flux decreases, followed by increases which approach the levels prior to the starting of reactor coolant pump 2B (RC-P-2B).	Ref. 4.f, Appendix CI - p. 12. The downcomer annulus was temporarily filled with coolant from the cold leg piping, shielding the detector from the core.
02:54:19 (0654:56)	Pressurizer heater groups 1 through 5 tripped.	Ref. 3.g, 2.g - p. 7, Appendix RCPCS - pp. 2, 3 - 5.
02:54:23 (0655:00) Approximate	Waste Gas Discharge Monitor radiation readings (WDG-R-1480) began to increase and went off scale high.	Ref. 12.a, Appendices RM - p. 3, AUX BLDG. This is at the 305 ft elevation in the auxiliary building.
02:54:50 (0655:27)	High pressure injection engineered safety features actuation logic automatically reset on increasing reactor coolant system pressure.	Ref. 3.g, 1.k, Appendix ESF - p. 12.
02:54:50 (0655:27)	Circulating water pump 1B (CW-P-1B) was started.	Ref. 3.g.
02:55:00 (0655:37) Approximate	A site emergency was declared.	Ref. 2.c - p. 7, 2.d - pp. 6 & 7, 2.a - p. 4, 2.b - p. 6, 2.m - p. 2, 2.p - p. 3, 2.m - pp. 2 & 3. Notification of offsite authorities was begun.

TIME	EVENT	REMARKS & REFERENCES
02:55:13 (0655:50)	The engineered safety features bypasses were cleared.	Ref. 3.g, Appendix ESF - p. 12.
02:55:26 (0656:03)	A condenser hotwell low water level alarm was received. (Nominal alarm setpoint was 22.5 in.)	Ref. 3.g. Indicated level was 21.82 in. The computer scan interval was 15 sec.
02:55:38 (0656:15)	Circulating water pump 1E (CW-P-1E) was started.	Ref. 3.g.
02:56:12 (0656:49)	Main steam isolation valves MS-V4B & MS-V7B indicated open.	Ref. 3.g.
02:56:19 (0656:56)	Main steam isolation valves MS-V4B and MS-V7B indicated closed.	Ref. 3.g, 2.c - p. 7. Stroke time for these valves is approximately 117 seconds.
02:59:23 (0700:00) Approximate	The Fuel Handling Building Air supply fan stopped and remained off the remainder of March 28.	Ref. 12.a.

TIME	EVENT	REMARKS & REFERENCES
(0700:00 March 28 to 1100:00 April 2) Approximate	Radiation monitors HP-R-222, 225, 226, 228 were off scale high or nearly off scale during this period.	Ref. 12.a, Appendix RM - p. 3.
03:00:00 (0700:37)	Reactor coolant system pressure was approximately 2045 psig.	Ref. 1.k.
03:00:56 (0701:33)	Condensate hotwell water level was off scale low.	Ref. 3.g, Appendix C/FDW - p. 5. The scale indicates from 10 to 50 in.
03:01:11 (0701:48)	The condensate storage tank B low water level alarm cleared. (Nominal alarm setpoint was 20 ft.)	Ref. 3.g, Appendix C/FDW - p. 6. Indicated level was 20.31 ft. The computer scan interval was 15 sec.
03:02:56 (0703:33)	A condenser hotwell low water level alarm was received, indicating that level had come back on scale.	Ref. 3.g, Appendix C/FDW - p. 5. Indicated level was 9.68 in.
03:03:39 (0604:16)	Turbine bypass valves from steam generator B were isolated by closing block valve MS-V15B.	Ref. 3.g, Appendix STEAM DUMP - pp. 2-4. Steam generator B was isolated.

TIME	EVENT	REMARKS & REFERENCES
03:04:00 (0704:37)	Steam generator B water level was established at approximately 60% on the operating range and was maintained during the next 8.5 hr.	Ref. 1.w, 1.x, Appendix OTSG - pp. 2-3.
03:06:40 (0707:17)	The condensate storage tank 1B low water level alarm was received. (Nominal alarm setpoint was 20 ft.)	Ref. 3.g, Appendix C/FDW - p. 6. Indicated level was 19.96 ft.
03:10:27 (0711:04)	Emergency feedwater pump 2A (EF-P-2A) was stopped.	Ref. 3.g, 1.p, 1.w. The water levels of steam generators A & B were 60.8% and 62.5%, respectively, on the operating range. All three emergency feed-water pumps were shut down.
03:11:10 (0711:47)	The condenser hotwell low water level alarm cleared. The nominal low level alarm setpoint was 22.5 in.	Ref. 3.g. Indicated level was 23.07 in.
03:12:28 (0713:05) Approximate	The operator opened the ERV block valve (RC-V2). The ERV high outlet temperature alarm was received.	Ref. 3.g, 4.a, 4.d, 4.h, 1.k, 1.g, 2.c - p. 7, 2.m - p. 2, Appendix ERV - pp. 3-5, Appendix TH - p. 63. ERV outlet temperature was 247.7°F. Reactor coolant system pressure and pressurizer coolant level began decreasing, followed by an increase in reactor building pressure.

TIME	EVENT	REMARKS & REFERENCES
03:12:53 (0713:30)	Reactor coolant pump 2B (RC-P-2B) was stopped.	Ref. 3.g. Appendix TH - p. 63. Attempts at forced circulation were again stopped. There had been no indication of flow and motor current had been lost.
03:13:58 (0714:35)	Pressurizer safety valves (R1A & R1B) outlet high temperature alarms were received.	Ref. 3.g. R1A 202.6°F R1B 202.8°F
03:14:23 (0715:00) through 03:20:23 (0721:00)	Intermediate Cooling Pump Area Monitor (HP-R-207) radiation reading increased and leveled off at 100 mr/hr.	Ref. 12.a, Appendix RM - p. 2.
03:17:00 (0717:37) Approximate	The ERV block valve (RC-V2) was closed.	Ref. 4.a, 4.h.
03:19:45 (0720:22)	An engineered safety features actuation of high pressure injection was manually initiated.	Ref. 3.g, Appendix ESF - p. 11. Manual initiation is indicated by the computer typewriter entries that groups 1, 2 and 3 of trains A and B have been placed in TEST.

TIME	EVENT	REMARKS & REFERENCES
03:20:13 (0720:50)	Reactor coolant makeup pump 1C (MU-P-1C) started.	Ref. 3.g, Appendix ESF - pp. 11, 14. Pumps 1A & 1C were operating. The reason for the 28 second delay in this pump start following the engineered safety features actuation cannot be explained at this time.
03:20:23 (0721:00) Approximate	All monitors on HP-UR-3264 stripchart were showing increasing radiation levels. These monitors are: Primary Coolant Letdown (MU-R-720) both channels, Intermediate Letdown Cooler B (IC-R-1091), Intermediate Letdown Cooler A (IC-R-1092), Intermediate Letdown Cooler Outlet (IC-R-1093) Plant Effluent Unit II, (WDL-R-1311), Decay Heat Closed Loop A (DC-R-3399), Decay Heat Closed Loop B (DC-R-3400), Nuclear Services Closed Cooling (NS-R-3401), Spent Fuel Cooling (SF-R-3402).	Ref. 12.a, Appendix RM - p. 1.
03:21:00 (0721:37) Approximate	Out-of-core nuclear instrumentation indicated sharp decreases in level.	Ref. 4.f, Appendix CI - p. 12. The downcomer annulus was filled with coolant from high pressure injection, shielding the detector from the core.

TIME	EVENT	REMARKS & REFERENCES
03:21:23 (0722:00) Approximate	Reactor Building Purge Air Exhaust (HP-R-225 & HP-R-226) Auxiliary Building Purge Air Exhaust (HP-R-222) gas radiation monitor readings increased off scale high. The gas channel auxiliary building heating and ventilating radiation monitor was almost off scale. The iodine channel of HP-R-225 & 226 indicated $1x10^5$ cpm.	Ref. 12.a, Appendix RM - p. 3.
03:21:23 (0724:00) Approximate	A general emergency was declared. Notification of off-site authorities was begun.	Ref. 2.b - p. 7, 2.r - p. 3. This action was initiated because of an 8 R/hr radiation reading on the Reactor Building Dome Monitor (HP-R-214).
03:25:56 (0726:33)	The pressurizer high coolant level alarm cleared. (Nominal alarm setpoint was 260 in.)	Ref. 3.g, 1.g, Appendix PDS - pp. 2, 12-13. Indicated level was 238 in. The computer scan interval was 15 sec.
03:27:33 (0728:00) Approximate	The auxiliary building access corridor radiation monitor readings began to increase.	Ref. 12.a, Appendices RM - p. 2, AUX BLDG.
03:29:23 (0730:00) Approximate	The fuel handling building air exhaust fan was turned off. Between 0730 and 1100, this exhaust fan was turned off and on several times with 30 to 60 minute run times.	Ref. 12.a, Appendix RM - p. 3.

TIME	EVENT	REMARKS & REFERENCES
03:30:58 (0731:35)	Pressurizer safety valve (RC-R1A) high outlet temperature alarm cleared.	Ref. 3.g. This indicated a decreasing temperature in the outlet header. Indicated temperature was 192.4°F.
03:32:26 (0733:03)	The pressurizer high coolant level alarm was received.	Ref. 3.g, 1.g. Indicated level was 271 in.
03:34:28 (0735:05)	Pressurizer safety valve (RC-R1B) high outlet temperature alarm cleared.	Ref. 3.q. This indicated decreasing temperature in the outlet header. Indicated temperature was 192.6°F.
03:35:06 (0735:43)	Emergency feedwater pump 2A (EF-P-2A) was started.	Ref. 3.g, 1.w. Steam generator A water level began decreasing from approximately 70% of the operating range and had reached about 43% by 0742.
03:37:00 (0737:37)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.g, 1.g. Pressurizer coolant level indication was increasing rapidly. Indicated level was 374 in.
03:40:00 (0740:37) Approximate	The ERV block valve (RC-V2) was opened.	Ref. 4.a, 4.h.
03:40:28 (0741:05)	Pressurizer safety valves (RC-R1A & RC-R1B) outlet temperatures alarmed high.	Ref. 3.g. Indicated temperatures were 201.6°F and 205.2°F, respectively.

TIME	EVENT	REMARKS & REFERENCES
03:46:23 (0747:00) Approximate	Out-of-core nuclear instrumentation indicated a sharp increase in count rate. Subsequently, the self powered neutron detectors responded to the high temperature conditions over broad sections of the core down to the lowest level.	Ref. 4.f, Appendix CI - p. 12. This change in count rate does not appear to be linked with any operational evolutions associated with the reactor coolant system.
03:48:23 (0749:00) Approximate	Makeup Tank Area Monitor (HP-R-206) radiation reading were increasing off scale. HP-R-210 and HP-R-214 radiation readings leveled out at $1.5x10^2$ R/hr.	Ref. 12.a, Appendix RM - p. 2.
03:55:39 (0756:16)	Engineered safety features actuation system B actuated on reactor building high pressure. The reactor building was isolated by train B valve closures.	Ref. 3.b, 2.a - p. 5, 2.q - p. 4, Appendix ESF - pp. 3, 6, 8, 10, 16 - 18. The nominal setpoint is 4 psig; the trip occurred at 3.2 psig. The actuation of the engineered safety features at this time could be attributed to the opening of the ERV block valve. Letdown flow is stopped whenever the building isolates. It is not possible to verify reinitiation of letdown flow subsequent to building isolations throughout the day. However, it does not appear that letdown flow was reestablished following this isolation.
03:55:39 (0756:16)	Intermediate cooling pump 1B (IC-P-1B) tripped.	Ref. 3.b. This trip was a result of reactor building isolation.

TIME	EVENT	REMARKS & REFERENCES
03:55:40 (0756:17) Approximate	Waste Gas Tank Discharge A Monitor (WDG-R-1485) radiation readings began increasing from 500 cpm and reached 3000 cpm at 0900.	Ref. 12.a, Appendices RM - p. 3, AUX BLDG. The monitor was located in the spent resin storage valve room on the 305 ft elevation in the auxiliary building.
03:55:46 (0756:23)	Engineered safety features actuation system A actuated on reactor building high pressure.	Ref. 3.b, 2.a - p. 5, 2.q - p. 4, Appendix ESF - pp. 6, 8. The nominal setpoint is 4 psig; the trip occurred at 3.1 psig.
03:55:46 (0756:23)	Intermediate cooling pump 1A (IC-P-A) tripped.	Ref. 3.b. This trip was a result of reactor building isolation.
03:56:04 (0756:41)	Makeup pump 1C (MU-P-1C) started.	Ref. 3.b. There was an unexplained delay again from the time engineered safety features actuated and makeup pump 1C started.
04:00:00 (0800:37) through 05:30:00 (0930:37) Approximate	Core thermocouple readings manually obtained indicated core exit temperatures in the range of 217°F to 2580°F.	Ref. 8.b, 2.b - p. 24, Appendix CI - p. 16. It is inferred from the observed temperature pattern that quenching in the core caused breakup of the oxidized fuel cladding and fuel debris redistribution in the core upper regions, blocking or restricting normal coolant flow paths.

TIME	EVENT	REMARKS & REFERENCES
04:00:00 (0800:37) Approximate	Pressurizer coolant level was 380 inches. Reactor coolant pressure was 1500 psig.	Ref. 1.g, 4.d.
04:00:13 (0800:50)	The engineered safety features actuation signal for building isolation cleared.	Ref. 3.b, Appendix ESF - p. 7.
04:00:13 (0800:50)	Intermediate cooling pumps 1A and 1B (IC-P-1A & 1B) started. Pump 1A tripped immediately and was restarted.	Ref. 3.b. Flow from the intermediate cooling system is required prior to reactor coolant pump startup.
04:08:37 (0809:14)	Reactor coolant pump 1A (RC-P-1A) was started.	Ref. 3.b.
04:09:14 (0809:51)	Reactor coolant pump 1A (RC-P-1A) was tripped.	Ref. 3.b. There was no flow indication and running current was low.
04:10:10 (0810:47)	Intermediate cooling pump 1B (1C-P-1B) tripped.	Ref. 3.b.
04:17:17 (0817:54)	Makeup pump 1A (MU-P-1A) tripped.	Ref. 3.b, 2.d - p. 10.

TIME	EVENT	REMARKS & REFERENCES
04:17:22 (0817:59)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b. No makeup pumps were operating. Coolant injection to the reactor coolant system was completely stopped.
04:18:16 (0818:53)	The operator attempted to restart makeup pump 1A (MU-P-1A).	Ref. 3.b, 2.d - p. 10. The pump would not start. The operator reported locking the pump out at this time.
04:19:02 (0819:39)	Intermediate cooling pump 1B (IC-P-1B) was started.	Ref. 3.b.
04:19:05 (0819:42)	Engineered safety features actuation system A actuated on reactor building high pressure. The reactor building was isolated by train A valve closures.	Ref. 3.b, 4.a, Appendix ESF - pp. 3, 6, 8, 10, 16 - 18. The nominal setpoint is 4 psig. The actuation occurred at 3.2 psig. There was no indication of makeup pumps 1A or 1B starting, which could confirm that makeup pump 1A was locked out. There is no explanation for pump 1C not starting.
04:19:06 (0819:43)	Decay heat removal pump 1A (DH-P-1A) started.	Ref. 3.b. This was a result of the engineered safety features actuation.
04:19:06 (0819:43)	Intermediate cooling pump 1A (IC-P-1A) tripped.	Ref. 3.b. This was a result of the engineered safety features actuation.

TIME	EVENT	REMARKS & REFERENCES
04:19:24 (0820:01)	The engineered safety features actuation system A actuation signal cleared.	Ref. 3.b, Appendix ESF - p. 8.
04:19:29 (0820:01)	Intermediate cooling pump 1A (IC-P-1A) was started.	Ref. 3.b.
04:22:02 (0822:39)	Makeup pump 1B (MU-P-1B) was started.	Ref. 3.b, Appendix RCPCS - pp. 2, 3-5.
04:26:22 (0826:59)	Letdown cooler 1A & 1B high temperature alarms were received. (Nominal alarm setpoint was 135°F.)	Ref. 3.b. 1A - 137.7°F 1B - 137.4°F These alarms could be indicative of reinitiation of letdown flow after the engineered safety features actuation signal cleared. The computer scan interval was 30 sec.
04:27:02 (0827:39)	Reactor coolant makeup pump 1C (MU-P-1C) was started. It tripped and was restarted.	Ref. 3.b, 2.d - p. 10.
04:30:30 (0831:07)	Pressurizer heater group 10 tripped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
04:30:45 (0831:22)	Condenser vacuum pumps 1A & 1C (VA-P-1A & VA-P-1C) were stopped and main condenser vacuum was broken.	Ref. 3.b, 2.r - p. 4, Problems had been encountered with the auxiliary boiler.
04:35:22 (0835:59)	Both letdown cooler 1A and 1B high temperature alarms cleared in a 30 sec. period.	Ref. 3.b. Indicated temperatures were 125.9°F and 130.8°F respectively.
04:42:14 (0842:51)	Emergency feedwater pump 2A (EF-P-2A) was shut down.	Ref. 3.b, 1, 1.p, 1.w, Appendix OTSG - pp. 2-3. Water levels in steam generators A and B were 44% and 66% respectively, on the operating range. All three emergency feedwater pumps were shut down.
04:44:23 (0845:00) Approximate	Letdown Cooler A Monitor (IC-R-1092) radiation readings began decreasing.	Ref. 12.a, Appendix RM - p. 1.
04:46:21 (0846:58)	Pressurizer heater groups 4 and 5 tripped off.	Ref. 3.b.
04:59:23 (0900:00) through 05:49:23 (0950:00) Approximate	HP-R-207 and RB Emergency Cooling Booster Pump Area Monitor (HP-R-204) radiation readings increased to $4x10^3$ mR/hr and then decreased.	Ref. 12.a, Appendix RM - p. 2.

TIME	EVENT	REMARKS & REFERENCES
05:18:00 (09918:37) Approximate	The operator closed the ERV block valve.	Ref. 4.a, 4.d, 4.h. This was an apparent attempt to reestablish normal system pressures.
05:18:47 (0919:24)	Decay heat removal pump 1A (DH-P-1A) was tripped.	Ref. 3.b.
05:20:00 (0920:37) Approximate	Reactor coolant system pressure started to increase from 1250 psig.	Ref. 1.k, 4.d. The increasing trend continued for about 30 minutes until pressure control was established within a band of 1865 to 2150 psig. This increase was a result of the block valve closure.
05:23:34 (0924:11)	Engineered safety features actuation system A actuated on high reactor building pressure. The reactor building was isolated by train A valve closure.	Ref. 3.b, Appendix ESF - pp. 3, 6, 8, 10, 16 - 18. There was no computer typewriter indication of makeup pump 1A (MU-P-1A) starting.
05:23:34 (0924:11)	Intermediate cooling pump 1A (1C-P-1A) tripped.	Ref. 3.b. This trip resulted from the train A reactor building isolation.
05:23:47 (0924:24)	The engineered safety features actuation system A actuation signal cleared.	Ref. 3.b, Appendix ESF - p. 8.

TIME	EVENT	REMARKS & REFERENCES
05:23:57 (0924:34)	Intermediate cooling pump 1A (IC-P-1A) was started.	Ref. 3.b.
05:29:23 (0930:00) Approximate	Fuel Handling Building Monitor(HP-R-215) and Control and Service Building Corridor Monitor (HP-R-234) radiation readings increased to 40 and 70 mr/hr.	Ref. 12.a, Appendix RM - p. 2.
05:29:23 (0930:00) through 14:59:23 (1900:00) Approximate	Radiation readings of the Auxiliary Building Access, Reactor Building Purge Unit Area, and Fuel Handling Building Exhaust Unit Area monitors all increased to off scale high. HP-R-324 indicated several rapidly increasing and decreasing radiation readings as the trace trended to off scale high.	Ref. 12.a, Appendix RM - p. 2.
05:34:57 (0935:34)	ERV (RC-R2) and pressurizer safety valve (RC-R1A) high outlet temperature alarms cleared.	Ref. 3.b. Indicated temperatures were 192.0°F and 192.9°F, respectively.
05:39:27 (0940:04)	Pressurizer safety valve (RC-R1B) high outlet temperature alarm cleared.	Ref. 3.b. Indicated temperature was 192.9°F.

TIME	EVENT	REMARKS & REFERENCES
05:43:06 (0943:43) Approximate	By cycling the ERV block valve open and closed, approximately 30 times, reactor coolant pressure was maintained between 1865 and 2150 psig during the following two hours.	Ref. 1.k, 4.a, 4.h.
05:43:27 (0944:04) through 05:46:27 (0947:04)	ERV (RC-R2) and pressurizer safety valves (RC-R1A & RC-R1B) outlet temperatures alarmed high.	Ref. 3.b. Indicated temperatures were 214.9°F, 205.9°, and 205.4°F, respectively.
05:49:23 (0950:00) Approximate	Control Room Intake Monitor (HP-R-220) radiation readings increased; the particulate channel reached $1x10^4$ cpm, and the iodine channel reached $3x10^3$ cpm. By 1100 hours, readings on all channels decreased to below $1x10^2$ cpm.	Ref. 12.a, Appendix RM - p. 3.
05:59:23 (1000:00) Approximate	The auxiliary building heating and ventilating exhaust fan tripped.	Ref. 12.a, 2.r - pp. 11 & 12, Appendix RM - p. 3. This was indicated by a decrease of flow on a strip-chart. This trip was reportedly caused by high radiation.

TIME	EVENT	REMARKS & REFERENCES
06:04:00 (1004:37) Approximate	Filling of steam generator A was started.	Ref. 1.p, 1.q, Appendix TH - p. 68. This could indicate an attempt to establish natural circulation in the reactor coolant system.
06:13:39 (1014:16)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.
06:14:06 (1014:43)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
06:14:23 (1015:00) Approximate	The auxiliary building heating and ventilating exhaust fan started.	Ref. 12.a, 2.r - pp. 11 & 12. The high radiation trip was reportedly bypassed.
1100:00 (March 28) through 0100:00 (March 29) Approximate	The fuel handling building air exhaust fan ran steadily.	Ref. 12.a.
07:08:31 (1109:08)	Emergency feedwater pump 2A (EF-P-2A) was started.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
07:14:06 (1114:43)	Indicated steam generator A level reached 100% on the operating range.	Ref. 1.p.
07:17:01 (1117:38)	Emergency feedwater pump 2A (EF-P-2A) was tripped.	Ref. 3.b.
07:38:54 (1139:31)	The ERV block valve (RC-V2) was opened.	Ref. 1.k, 4.a, 4.d, 4.h, Appendix TH, p. 74. A rapid sustained depressurization of the reactor coolant system was begun.
07:41:35 (1142:12)	The engineered safety features actuation signals were bypassed.	Ref. 3.b, Appendix ESF - p. 12. This event prevents the engineered safety features of the high pressure injection system from actuating.
07:43:44 (1144:21)	Pressurizer heater groups 1 and 2 tripped off and indicated on 2 seconds later.	Ref. 3.b.
07:44:23 (1145:00) Approximate	The auxiliary building heating and ventilating exhaust fan stopped.	Ref. 12.a.
07:50:16 (1150:53)	Pressurizer heater groups 1 and 2 tripped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
08:01:13 (1201:50)	A letdown cooler 1B high temperature alarm was received. (Nominal alarm set point was 135°F.)	Ref. 3.b. The indicated temperature was 140°F. Letdown flow is automatically isolated when this alarm is received.
08:15:22 (1215:59)	A letdown cooler 1A high temperature alarm was received. (Nominal alarm set point was 135°F.)	Ref. 3.b. The indicated temperature was 137.4°F.
08:30:00 (1230:37) Approximate	The power operated emergency main stream dump valve (MS-V3A) was shut at the request of corporate management.	Ref. 2.a - p. 21, 2.d - p. 9, 2.r - p. 4, Appendix STEAM DUMP - pp. 5 - 6. This was in response to concern expressed by the state government.
08:31:06 (1231:43)	Decay heat closed cooling water pumps 1A & 1B (DC-P-1A & DC-P-1B) were started.	Ref. 3.b. This system provides cooling to the decay heat removal system coolers and pumps.
08:31:10 (1231:47) Approximate	Reactor coolant pressure had decreased to 600 psig.	Ref. 3.h, 4.d, Appendices TH - p. 75, CF. Core flood should have initiated at this time.
08:34:23 (1235:00) Approximate	The auxiliary building heating and ventilating exhaust fan started momentarily, then remained off for about one and a quarter hours.	Ref. 12.a.
09:04:18 (1304:55)	Makeup pump 1C (MU-P-1C) stopped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
09:10:00 (1310:37) Approximate	ERV block valve (RC-V2) was closed. Reactor coolant system pressure had decreased to approximately 435 psig and then began to increase.	Ref. 4.a, 4.d., 4.h, Appendices TH - p. 75, CF. Core flood apparently stopped after approximately 100 ft^3 (770 gal) of coolant had been injected into the reactor coolant system from each of the two core flood tanks.
09:16:58 (1317:35)	ERV (RC-R2) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature was 192.7°F.
09:19:52 (1320:29)	The letdown cooler 1A high temperature alarm cleared.	Ref. 3.b. The indicated temperature was 131.3°F.
09:20:28 (1321:05)	ERV (RC-R2) outlet temperature alarmed high.	Ref. 3.b. The indicated temperature was 220.4°F. The ERV block valve may have been opened.
09:31:28 (1332:05)	ERV (RC-R2) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature was 192.2°F. The ERV block valve may have been closed.
09:48:58 (1349:35)	ERV (RC-R2) outlet temperature alarmed high.	Ref. 3.h. The indicated temperature was 225.7°F. It had been reported that the ERV block valve was opened at the time of the hydrogen burn.

TIME	EVENT	REMARKS & REFERENCES
09:49:23 (1350:00) Approximate	The auxiliary building heating and ventilating exhaust fan was started and ran for about 30 minutes.	Ref. 12.a.
09:49:43 (1350:20) through 09:49:50 (1350:27)	Engineered safety features high pressure injection, reactor building isolation, reactor building spray pumps and valves, and decay heat removal pumps were actuated. Makeup pump (MU-P-1C) started. There was no indication of makeup pump 1A (MU-P-1A) running or starting.	Ref. 3.b, 4.a, 2.a - p. 22, 2.1 - p. 4, 2.4 - p. 8, 2.5 - p. 1, Appendix ESF - pp. 6, 8 - 10. Actuation was caused by reactor building high pressure which was caused by a hydrogen burn.
09:49:43 (1350:20) Approximate	Motor control centers 32A & 42A were lost.	Ref. 2.a - p. 24, 2.r - p. 7, 2.q - p. 8. These motor control centers supplied all the auxiliaries for the four reactor coolant pumps and seal water pumps for many of the pumps in the auxiliary building. This event coincided with the hydrogen burn.
09:49:58 (1350:35)	Reactor coolant pumps 1A and 1B (RC-P-1A & RC-P-1B) inlet air temperature high alarms were received, and pressurizer safety valves (R1A & R1B) discharge line temperature high alarms annunciated.	Ref. 3.b. RC-P-1A 157.5°F RC-P-1B 124.7°F R1A 203.7°F R1B 205.0°F These temperature alarms could have been caused by the hydrogen burn.

TIME	EVENT	REMARKS & REFERENCES
09:50:11 (1350:48)	The reactor building isolation and cooling actuation signal was defeated.	Ref. 3.b, Appendix ESF - pp. 12-13. Shutdown of the makeup pumps operating in the high pressure injection mode is made possible by defeating the engineered safety system actuation signal.
09:50:11 (1350:48)	Intermediate cooling pumps 1A and 1B (IC-P-1A & IC-P-1B) were started.	Ref. 3.b.
09:50:24 (1351:01)	Makeup pump 1C (MU-P-1C) was stopped.	Ref. 3.b.
09:51:58 (1352:35)	The ERV (RC-R2) and pressurizer safety valves (RC-R1A) discharge line high temperature alarms cleared.	Ref. 3.b. RV2 180.5°F R1A 178.6°F The ERV block valve may have been closed.
09:52:28 (1353:05)	ERV (RC-R2) outlet temperature alarmed high.	Ref. 3.b. The indicated temperature was 208.8°F. The ERV block valve may have been opened.
09:54:28 (1355:05)	Pressurizer safety valve (RC-R1B) high outlet alarm cleared.	Ref. 3.b. The indicated temperature was 177.3°F. The ERV block valve may have been closed.
09:55:10 (1355:47)	Pressurizer heater group 8 tripped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
09:55:28 (1356:05)	Pressurizer safety valve (RC-R1B) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature alarm was 190.3°F.
09:55:30 (1356:07)	Reactor building spray pumps were stopped.	Ref. 3.b, Appendix ESF - pp. 12-13.
09:56:58 (1357:35)	Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) were stopped.	Ref. 3.b, Appendix CF.
09:58:38 (1359:15)	The engineered safety features actuation signal cleared.	Ref. 3.b, Appendix ESF - p. 8.
10:00:00 (1400:37) Approximate	The ERV block valve (RC-V2) was opened.	Ref. 4.a, 4.h, Appendix TH - p. 76.
10:00:58 (1401:35)	ERV (RC-R2) outlet temperature alarmed high.	Ref. 3.b. The indicated temperature was 206.7°F.
10:05:25 (1406:02)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
10:07:19 (1407:56)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
10:26:15 (1426:52)	The reactor coolant system loop A hot leg temperature decreased to less than 620°F.	Ref. 1.b, 2.s - p. 1.
10:31:30 (1432:07)	Makeup pump 1C (MU-P-1C) was started.	Ref. 3.b. The operation of this pump during the following hour could be indicative of efforts to maintain or increase the coolant inventory in the reactor coolant system.
10:32:36 (1433:13)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
10:34:00 (1434:37) Approximate	Reactor coolant pressure decreased again to approximately 435 psig.	Ref. 4.d, Appendices TH - p. 76, CF. Core flood should have initiated again.
10:35:00 (1435:37) Approximate	Reactor coolant pressure had decreased to approximately 410 psig and began to increase.	Ref. 4.b, Appendices TH - p. 76, CF. Core flood injection apparently stopped. An additional estimated 22 ft^3 (161 gal) of coolant had been injected into the reactor coolant system from each flood tank.
10:35:55 (1436:32)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
10:38:57 (1439:34)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.
10:39:51 (1440:28)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
11:08:00 (1508:37) Approximate	The ERV block valve (RC-V2) was closed.	Ref. 3.h., 4.a, 4.h.
11:18:34 (1519:11)	Makeup pump 1C (MU-P-1C) was started.	Ref. 3.b.
11:24:29 (1525:06)	ERV (RC-R2) high outlet temperature alarm cleared.	Ref. 3.b. The indicated temperature was 191.9°F.
11:28:12 (1528:49)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.
11:28:52 (1529:29)	Pressurizer heater groups 1 and 2 tripped off.	Ref. 3.b.
11:32:37 (1533:14)	Makeup pump 1C (MU-P-1C) was started.	Ref. 3.b.

TIME	EVENT	REMARKS & REFERENCES
11:33:44 (1534:21)	Emergency feedwater pump 2B (EF-P-2B) was started.	Ref. 3.b.
11:35;48 (1536:25)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.
11:38:57 (1539:34) Approximate	Feedwater flow to steam generator B was started.	Ref. 1.w.
14:45:17 (1545:54)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
11:52:04 (1552:41)	Emergency feedwater pump 2B (EF-P-2B) was tripped. Feedwater flow to steam generator B was stopped.	Ref. 3.b., 1.w.
12:34:00 (1636:37) Approximate	ERV block valve (RC-V2) was opened.	Ref. 3.h, 4.a, 4.h.
12:34:29 (1635:06)	ERV (RC-R2) outlet temperature alarmed high.	Ref. 3.b. Indicated temperature was 233°F.

TIME	EVENT	REMARKS & REFERENCES
12:46:00 (1646:37) Approximate	ERV block valve (RC-V2) was closed.	Ref. 3.h, 4.a, 4.h.
12:52:00 (1652:37) Approximate	ERV block valve (RC-V2) was opened.	Ref. 4.a, 4.h.
13:00:00 (1700:37) Approximate	ERV block valve (RC-V2) was closed.	Ref. 4.a, 4.h.
13:02:23 (1703:00)	Condenser vacuum pump 1C was started.	Ref. 3.b. This event and the subseuent one indicate the reestablishment of vacuum in the main condenser.
13:13:10 (1713:47)	Condenser vacuum pump 1A was started.	Ref. 3.b.
13:23:04 (1723:41)	Makeup pump 1C (MU-P-1C) started.	Ref. 3.b.
13:24:59 (1725:36)	ERV (RC-R2) high outlet temperature alarm cleared.	Ref. 3.b. Indicated temperature was 192.9°F.

TIME	EVENT	REMARKS & REFERENCES
13:26:09 (1726:46)	Pressurizer heater groups 1 and 2 tripped.	Ref. 3.b.
14:25:26 (1826:03)	Pressurizer heater groups 1 and 2 indicated on.	Ref. 3.b.
14:43:15 (1843:52)	Makeup pump 1C (MU-P-1C) tripped.	Ref. 3.b.
14:55:35 (1856:12)	Reactor coolant system pressure was 2327 psig.	Ref. 3.h.
14:59:23 (1900:00) Approximate	The Fuel Handling Building Exhaust Unit Area Monitor, Reactor Building Purge Unit Area Monitor, and Auxiliary Building Access Area Monitor radiation readings came back on scale and were decreasing.	Ref. 12.a, Appendix RM - p. 2.
15:32:42 (1933:19) Approximate	Reactor coolant pump 1A (RC-P-1A) was started and run briefly, then stopped.	Ref. 3.h, 1.f, 2.a - p. 25. There was still no power available from motor control centers 32A and 42A for reactor coolant pump auxiliaries. Necessary circuit bypasses were installed to permit operation of the pump.

TIME	EVENT	REMARKS & REFERENCES
15:49:36 (1950:13) Approximate	Reactor coolant pump 1A (RC-P-1A) was restarted.	Ref. 3.h, 1.f, 2.a - p. 25, 2.s - p. 4. Flow was indicated at 1950:46 on the reactimeter.
16:29:23 (2030:00) Approximate	Radiation readings showed marked increases and went off or nearly off scale on the Fuel Handling Building Exhaust Unit Area Monitor (HP-R-3236), Auxiliary Building Access Corridor Monitor (HP-R-232), Waste Disposal Storage Area Monitor (HP-R-218).	Ref. 12.a, Appendix RM - p. 2.

Plant Status at 2000:00

At approximately 2000:00 on March 28 (16 hours after the initiating event, indications show that forced circulation had been reestablished using reactor coolant pump 1a (RC-P-1A). The reactor coolant system pressure was being maintained at approximately 1000 to 1100 psig with temperatures indicating a cooling trend. Heat was being removed from the reactor coolant system using steam generator A. Steam generator B was isolated and condenser vacuum had been established.

APPENDIX SOE COMP

COMPARISON OF SEQUENCES OF EVENTS

Sequences of events (SOEs) relating to the accident that occurred on March 28, 1979, at the Three Mile Island nuclear power plant, Unit 2, have been prepared by several organizations. This appendix will briefly discuss the various SOEs, identify any significant differences among them, and reconcile the differences, if possible. The SOEs of interest were prepared by the following groups:

- o Nuclear Safety Analysis Center - NSAC-1
- o General Public Utilities - GPU SOE
- o Nuclear Regulatory Committee - NUREG-0600
- o The President's Commission - Kemeny Commission Report
- o NRC Special Inquiry Group - Rogovin Report

NSAC-1 - The objective of the NSAC-1 SOE is to explain what happened on the day of the accident and to document the verification of the events. It is meant to be a factual account and provides limited discussions of why the operators took the actions that they did. Primary sources of information and verification included the reactimeter, plant computer alarm and utility typewriter output, control room strip charts, and operator interviews that were performed by GPU. Interviews primarily were used to substantiate verified events, but a few events were identified based only on operator interviews. Appendices are used to support and explain the SOE and to provide narratives of the thermal-hydraulic and core damage analyses.

GPU - The GPU SOE differs in style from NSAC-1 in that it calls out information available to the operator at the time each event or action occurs, includes discussions as to why certain actions are taken, and provides periodic plant status updates. Reference data sources are the same as NSAC-1 with the exception that the latest version incorporates references to inverviews conducted by the NRC.

NUREG-0600 - This SOE employs rather extensive event descriptions which include comments and the reasons that operators were taking certain actions. NRC reports, notes and tapes are used in this SOE in addition to the previously mentioned data sources. NUREG-0600 is a more critical report than NSAC-1 because it includes evaluations of operator performance and identifies areas of noncompliance with NRC regulations.

Kemeny Commission Report - In this report, the account of the event is a chronological narrative of the accident, not a sequence of events, per se. The Kemeny Commission used all previously noted reference information, as well as interviews which they personally conducted.

Rogovin Report - The SOE in this report was prepared using the NSAC-1, NUREG-0600, and the GPU SOEs, along with the reference material used by all. Using this approach, an attempt was made to flag any events where major differences occur and to reconcile any differences among the SOEs.

As one might expect, there are many dissimilarities among the SOEs because of the different styles and formats used, the intents of the author organizations, and the events or items chosen for inclusion. There is, however, substantial agreement among the SOEs with regard to the significant factual events during the first sixteen hours of the accident. There are no differences which would alter the understanding of what happened or would lead one to new areas of investigation.

Minor time differences occur among the reports for events that are not verified by plant computer typeouts or reactimeter data. For example, opening and closing time of the electromatic relief block valve are determined by the temperature and pressure responses of the containment building and the temperature alarms of the valve discharge header. Differences in interpretation of the data and lag times in parameter response to the valve actuation are responsible for these variations in times. These small variations, however, are not important to the understanding of the event or to the results of the analysis and they need not be resolved.

Another difference, the time at which the atmospheric dump valves were closed, is a reported time of 1230 in the NSAC-1 and GPU reports versus 1315 in NUREG-0600. References in the first two reports were based on interviews and those in NUREG-0600 were based on an interview, written notes, and a written statement by the station manager. It is not clear that either time can be truly verified, because all personnel involved in the event acknowledge that their sense of time was very distorted, and accurate logs were not maintained. This is another time difference whose resolution is not necessary for understanding or analyzing the accident.

Based on an interview, early versions of the GPU SOE had makeup pump 1C stopped at approximately 0645 with no previous startup shown. NUREG-0600 made note of this and stated that the event was unsupported because the alarm typewriter history was lost during that time. The latest GPU SOE deleted the reference to that pump stop. NSAC thermal hydraulic analysis indicates, however, that a makeup pump was probably operated for about ten minutes between approximately 0635 and 0645; therefore, a makeup pump start and stop has been included in NSAC-1.

The GPU SOE states that pressurizer spray flow was initiated at 0655:43 and was maintained until 0714:04. The Rogovin SOE states that the pressurizer spray valve was opened at that time, but that without reactor coolant pump operation spray flow is impossible. Other SOEs omit this event, presumably because it doesn't appear to be relevant. Similar entries appeared in the GPU and Rogovin SOEs at 0745:04 and 1158:52.

A manually initiated actuation of the high pressure injection portion of the engineered safety features at approximately 0541 was noted in the latest GPU SOE. This occurred at a time when the computer alarm history was lost, so it cannot be verified. The entry was based on two separate interviews of one control room operator. A review of those interviews indicates that there may have been a manual actuation about the time the reactor coolant pumps were shut down, but this actuation does not appear to be certain enough to be considered factual.

Another time discrepancy between the GPU SOE and the others is the time of emergency boration. At one time all of the SOEs agreed that this time was approximately 0640, but the latest GPU SOE was changed to 0617 based on interviews. This time appears to be a judgement, and one other interview clearly gives the time as 0640. Considering the uncertainties in the operators' sense of time on the first day, emergency boration could have been started at either time or more likely, some time within that general time interval.

Both the GPU and NUREG-0600 SOEs reference indicated core flood tank 1A level changes, and the former attributes an indicated change at about 1358 to a discharge from the tank to the reactor coolant system. NSAC-1 deleted any references to core flood tank alarms because between 1212 and 1505, nine high level alarms were received. After the first seven high level alarms, a return to normal signal was received very shortly. After the eighth alarm, the reading went off scale high, returned to the high level range, and went off scale permanently. Because the alarms did not coincide with the probable core flood times as determined by thermal hydraulic analysis and core flood tank 1B did not alarm, it was concluded that the validity of those alarms was questionable and that they may have resulted from degraded instrumentation. Similarly, reference to the reactor building air cooling coils B emergency discharge temperature alarms were not incorporated in the NSAC-1 SOE.

There are many other differences among the reports, but as stated before, none that would lead to a different perception of the accident or cause confusion in understanding its progression. These reports can be used to supplement one another and to provide slightly different insights into the events on the day of the accident. Further refining of the existing SOEs is not warranted.

APPENDIX ASP

ALTERNATE SUCCESS PATHS

Any small-break loss-of-coolant accident (LOCA) in a pressurized water reactor (PWR), including the type of event experienced at Three Mile Island Unit 2 (TMI-2) with an open relief valve, can be successfully terminated at virtually any phase in the transient. If such an event were left totally unchallenged, it would lead to significant core damage and might lead to subsequent release of radiation. Prior to the TMI-2 accident, the most attention was usually focused on the large-break LOCA. Frequently it represented the worst possible condition. If that event could be controlled, all other possibilities would be encompassed. TMI-2 emphasized that this is not always true. This report discusses the manual and automatic actions that are typically available to prevent a small-break LOCA from proceeding further and thereby posing any threat.

It is recognized that the small-break LOCA covers a wide range of break sizes and resulting reactor system responses. Also, that for many small-break LOCAs, early and automatic safety system responses are required to mitigate or control the initial thermal-hydraulic transients that can potentially cause core damage early in the accident sequence (within about the first 10 minutes). The general theme of this appendix is to conceptually describe the means available for mitigating a small-break LOCA throughout an entire hypothetical accident sequence. These means involve both the time frame available for decisions to be made and the range of options potentially available for such mitigation.

Comparison with WASH-1400

The scenarios provided in this Appendix for countering a generic small-break LOCA in a PWR use the TMI-2 system as a base for determining the types, numbers and capacities of Engineered Safety Feature (ESF) systems[1,2]. These scenarios point to a lower probability of core melting than the Rasmussen Report (WASH-1400), for several reasons. Among these are the following:

1. WASH-1400 tacitly assumes that, once fuel temperature has reached 2200°F, the accident progresses inevitably to gross core melting. It is clear that several barriers to such progress exist in the form of operator actions which can be taken, equipment actions which will occur, and physical behavior of the system.

2. WASH-1400 tacitly assumes that gross core melting progresses to rupture of the reactor pressure vessel and that an inevitable consequence of this is one or another type of containment failure and release of fission products to the environment. It has now been agreed by several independent groups that this overall sequence terminating in containment failure has a relatively low probability. Similarly, WASH-1400 gives no credit to the possibility that an operator error may be discovered and corrected before irretrievable damage is done.

The quantitative evaluation of the degree to which WASH-1400 overestimates the probability of progressing from a badly damaged core to core melting and containment failure is still under study by NSAC but it is clear that a substantial margin of overestimate exists and that the perception of an imminent and massive release of radioactivity at TMI-2 was incorrect.

Mitigating Features for Small-Break LOCAs

Several features of a small-break LOCA in a PWR strongly enhance the ability to respond effectively and to minimize the resulting damage. These features again assume initially automatic ESF responses for the category of small-break LOCA that require such mitigating actions very early in the accident sequence. For example:

1. Sufficient time is available to select and apply effective countermeasures that can control or reverse the approach to or progress of core damage at virtually any stage in the sequence. Such an accident can typically require tens of minutes to hours to reach a damage threatening stage and similar additional times to reach severe damage levels. For example, at TMI-2 the core damage did not begin until about 2 hours into the accident.

2. The magnitude of the countermeasures necessary to control a small-break LOCA are well within the capacity of each of several of the typical installed PWR coolant injection and heat sink systems. As a result there is inherent redundancy and a high probability of the availability of at least one effective system. For example, if 200 to 300 gallons per minute of water had been supplied continuously to the TMI-2 core from shutdown of the reactor coolant pumps (at about 100 minutes) until a reliable adequate heat sink had been established (such as steam generator condensing with the block valve closed) no core damage would have occurred (see Table 1).

3. The deviation from normal system parameters and heat sink capabilities that develop during a small-break LOCA provides many observable conditions (observables) which indicate both current accident state and trends. The

scope and time scale of these observables permit rational selection of effective countermeasures and bases for use in conservatively projecting potential public danger and for making emergency planning decisions.

Table 1 indicates the minimum flow rate to the core that is necessary to maintain a cooled core. This minimum flow rate is essentially directly proportional to the decay heat and thereby decreases along with the decay heat as time following reactor scram increases.

If this minimum flow rate begins early in the accident (anytime up to about 10 minutes after observables confirm that actual uncovering and dryout of the reactor core has started) there will be little or no core damage. Additional delay in starting this minimum flow would lead to a cooled but increasingly damaged core.

Table 1

Required Coolant Flow Rates for Removing Decay Heat from 880-MWe Power Plant

Decay Heat (% Full Power)	Time (Hours)	Minimum Coolant Flow to Core Inlet Plenum* (Gallons of water per minute)	
		Complete Subcooling	No Subcooling
2%	0.24	340	540
1%	2.6	170	270
0.5%	22	85	135
0.25%	113	43	68

NOTE: The core coolant flow rates represent the actual flow that is necessary to completely remove decay heat by producing steam.

In summary, the small-break LOCA represents a class of accidents requiring relatively minor system responses in the form of increased flow to the core inlet plenum (e.g., less than about 300 GPM for times greater than about 20 minutes) and available heat sinks to assure a continuously cooled core.

Scenario Presentation

The format of the scenario is in the form of charts presenting a pathway of increasing actual and potential core damage due to nonactions or insufficient responses. This pathway is developed schematically in Table 2 for the four categories of 1) preventing

*Based on an assumed system pressure of 680 psia or 500°F saturation temperature. The pressure is of only minor importance. If atmospheric pressure is chosen (212°F saturation temperature), flow rates would be increased by less than 5% for the completely subcooled case and decreased by 26% for the saturated (no subcooling) case.

core damage, 2) terminating core damage, 3) terminating melting of the core, and 4) protecting the reactor containment building. Table 3 presents the same scheme in more detail with observables and sequential automatic/operator responses that can be used for mitigating the accident. The last 10 charts present Table 3 information in greater detail, emphasizing the observables and the hierarchy of responses within the same four basic damage categories noted above. The WASH-1400 study equates a condition somewhere between categories 2 and 3 (when the core reaches 2200°F) with both massive core melt <u>and</u> eventual failure of the containment.

The overall objective throughout these sequences, up to reactor vessel failure, is to maintain primary system coolant inventory control with sufficient coolant flow into the core region (Table 1) to maintain its coolability.

As a point of reference in Tables 2 and 3, the current best estimates of TMI-2 core damage indicate that it reached the realm of an advanced condition 2 of Tables 2 and 3 (terminating core damage) or early condition 3(terminating core melting). The core was significantly disrupted (partially rubbilized) and once sufficient coolant flow was reestablished, the core was coolable in place.

If nonactivation or inadequate activation of automatic or operator-induced remedies persists to the point where the reactor vessel fails, the objectives change to the cooling of a core debris bed in the reactor cavity. This cooling utilizes both 1) coolant injection systems that deliver water directly to the reactor vessel (with the ensuing drainage onto the debris bed) and 2) containment building heat removal systems that can also supply water onto the debris bed (if needed) and can handle the decay heat removal without endangering the containment integrity.

Increasing small-break LOCA severity and core damage occurs in moving from left to right in Tables 2 and 3, with progress governed by the non-performance of automatic and operator responses. These tables are useful in the following discussion of the accident scenario. The progression through Tables 2 and 3 and each of the 10 charts means that all previously described automatic and operator responses did not occur or were insufficient (e.g., defeated by operator action prior to reaching long-term stability) - hence the progression of accident severity to the level currently being discussed. Note that time and effective countermeasures, including reinstituting defeated ESF systems, are still available for mitigating the accident even at the advanced stages of the accident.

Comparison of large-break LOCA conditions with small-break LOCA conditions makes evident the much faster and stronger countermeasures required to control the initial phases of a large-break LOCA. However, the large break LOCA rapidly diminishes to small-break conditions if successfully countered for the first 10 to 20 minutes. After that period the large-break LOCA has virtually identical requirements for core cooling as a large small-break LOCA which has continued leakage, as described in this report.

Table 2

SCHEMATIC SMALL BREAK LOCA ACCIDENT MITIGATION FOR PRESSURIZED REACTOR SYSTEMS

INCREASING TIME
INCREASING DAMAGE
DECREASING PROBABILITY OF OCCURRENCE

TIME REGIME:	EARLY	INTERMEDIATE		LONG TERM
CATEGORY:	1	2	3	4
OBJECTIVE:	PREVENT CORE DAMAGE	TERMINATE CORE DAMAGE	TERMINATE CORE MELTING AND MAINTAIN CONTAINMENT INTEGRITY	ESTABLISH COOLING OF CORE DEBRIS
	PROGRESSIVE RESPONSES OF OPERATORS AND/OR AUTOMATIC SYSTEMS TO CURRENT OBSERVABLES	PROGRESSIVE RESPONSES BASED ON CURRENT OBSERVABLES	PROGRESSIVE RESPONSES BASED ON CURRENT OBSERVABLES	PROGRESSIVE RESPONSES BASED ON CURRENT OBSERVABLES
	NO AUTOMATIC RESPONSES	NO AUTOMATIC RESPONSES	NO AUTOMATIC RESPONSES	
	NO OPERATOR RESPONSES INADEQUATE RESPONSES (E.G., ESF DEFEATED)	NO OPERATOR RESPONSES INADEQUATE RESPONSES	NO OPERATOR RESPONSES INADEQUATE RESPONSES	
RESULTS:	CONDITION MITIGATED, CONTROLLED COOLING OF ALL SYSTEMS IN PROGRESS	CONDITION MITIGATED CONTROLLED COOLING IN PROGRESS	CONDITION MITIGATED CONTROLLED COOLING IN PROGRESS	CONDITION MITIGATED CONTROLLED COOLING IN PROGRESS CONTAINMENT PROTECTED

Table 3

SUMMARY OF SMALL-BREAK LOCA ACCIDENT MITIGATION PRESSURIZED WATER REACTOR SYSTEM

Increasing Damage Due to Small-Break LOCA →

Decreasing Probability of Occurrence

TIME REGIME:	EARLY	INTERMEDIATE		LATE	LONG-TERM
INITIAL CONDITION:	INITIAL CORE UNCOVERING	CORE UNCOVERING	CORE MELTING	REACTOR VESSEL FAILURE	CORE DEBRIS BEING COOLED
CATEGORY:	1	2	3	---------- 4 ----------	
OBJECTIVE:	Prevent Core Damage	Terminate Core Damage	Terminate Melting	Establish Cooling of Core Debris	Maintain Containment In-Integrity
PRIMARY INDICATORS OF INCREASING DAMAGE POTENTIAL (ASSUMES NO EARLIER ACTIONS)	1. $P_p\downarrow$; Pressurizer level ↑ or ↓ 2. HPI Actuates 3. $P_c\uparrow$; $T_c\uparrow$ 4. Deviation in Source-range Monitor Signals 5. $T_{SAT} \rightarrow T_p$ 6. Boiling Noises/Coolant Pump Vibrations 7. $\Delta T_{SUP} > 0$	1. $\Delta T_{sup} > 0$ 2. P_p Low or ↓ 3. Radiation Signals ↑ 4. H_2 in Containment 5. P_c and/or T_c High 6. Containment Spray and/or Coolers Actuate	1. to 6. Same as Previous But of Increasing Severity 7. Core Flood Actuates 8. LPI Actuates	1. $P_p \rightarrow \sim 0$ 2. P_c & T_c Suddenly ↑ 3. Radiation Signals Large ↑ 4. Core Flood Actuates 5. LPI/RHR Actuates 6. Pressurizer Dumps 7. Containment Spray Actuates 8. Containment Coolers Actuate	Responding to Previous Conditions and Actions
OPERATOR/SYSTEM ACTIONS WITH SUCCESSIVE FALL-BACK POSITIONS	1. Increase Normal Make-up; Stop Letdown (very small breaks) 2. Actuate HPI 3. Align HPI/LPI to Sump NOTE: P_p & T_p = Primary Pressure and Temperature P_c & T_c = Containment Pressure and Temperature T_{SAT} = Primary Saturation and Temperature ΔT_{SUP} = Primary Superheat Temperature Difference	1. HPI 2. Core Flood 3. LPI (for larger breaks) 4. Align HPI/LPI to Sump	1. HPI 2. Core Flood 3. LPI (for larger breaks) 4. Align HPI/LPI to Sump 5. Preventive Containment Spray to Predetermined Level in Containment	1. Core Flood 2. LPI 3. Pressurizer 4. Actuate Pressurizer Spray 5. Actuate Containment Spray 6. Actuate Containment Coolers 7. River Water Flooding	Establish Reflux Cooling 1. Initial Containment - Spray 2. LPI & RHR 3. Pressurizer Spray & RHR 4. LPI with 1 of 5 Independent Fan Coolers 5. Pressurizer Spray with 1 of 5 Fan Coolers 6. Quiescent Pool Boiling with 1 of 5 Fan Coolers 7. Natural convection alone after 50 days. 8. Many other options available or producible
RESULT	Core Covered and Cooled	Core Recovered & Cooled	Core Recovered and Cooled	Core Debris Cooled	Core Debris Cooled/ Containment Protected

Chart 1

OBSERVABLES FOR CATEGORY 1

PREVENTING CORE DAMAGE

CONDITION: Initial Core Uncovering

Primary Indications of Increasing Damage Potential (in order of occurrence assuming no prior corrective action)

1. Primary system pressure decreasing -- there is a leak in the primary system; pressurizer level deviates from expected values.

2. High pressure injection (HPI) actuates on low pressure signal (<1600 psia) -- the leak is continuing.

3. Containment building pressure and temperature increase -- the containment building is filling with leaked steam.

4. Deviation from expected (post-SCRAM) values in power-range and/or source-range monitor signals.

5. Saturation tempearture conditions reached -- this is a necessary precursor to drying out any portion of the core.

6. Boiling noise detected in primary system; main recirculation pumps have unusual flow and power loadings or vibrations.

7. Any primary system temperature measurement exceeding saturation temperatures -- this is a positive indication that core uncovering has been initiated.

8. As will be the case for the following "Primary Indications . . .", there are many other indicators which will either directly actuate engineered safety features or initiate operator actuation of these features or other options.

This and following charts are by no means complete, but they represent the primary observable signals, in approximately sequential order, that core damage may occur or is in progress, and options available for mitigating such progress.

Chart 2

ACTIONS FOR CATEGORY 1

PREVENTING CORE DAMAGE

CONDITION: Initial Core Uncovering

Operator/System Actions with Successive Fallback Positions:

- Stop letdown; increase makeup (300 gpm max.)
  - High Pressure Injection (HPI) actuation - 2 independent systems (900 gpm max.) — Common System
    - Align HPI systems with containment sump - 2 independent systems (900 gpm max.)

RESULT: Core covered and cooled - by about 20 minutes, only ~300 gallons per minute (gpm) effective makeup flow from all sources (e.g., MU, HPI, forced or natural circulation) to reactor core is required for this result; by 3 hours, ~200 gpm required; by 24 hours, <100 gpm required.

Chart 3

OBSERVABLES FOR CATEGORY 2

TERMINATION OF CORE DAMAGE

CONDITION: Core Uncovering in Progress

Primary Indications of Increasing Damage Potential (in order of occurrence assuming no prior corrective action)

1. Measured temperatures in primary system exceed saturation temperature -- core damage is imminent or progressing.

2. Primary system pressure remains low (<1600 psig) or continues to decrease -- primary system is still leaking and HPI (2 independent systems) did not automatically actuate (or was manually defeated).

3. Containment area radiation monitors increased noticeably from previous accident background levels -- core damage has progressed to probable failure (rupture) of some fuel rods.

4. High levels of hydrogen detected in containment and hydrogen recombining system actuates -- core damage has progressed to onset of notable zircaloy oxidation.

5. Containment pressure and/or temperature are at abnormally high levels.

6. Containment spray and/or coolers (5 independent systems) actuate and containment isolates (if containment pressure of 10 psig is reached).

Chart 4

ACTIONS FOR CATEGORY 2

TERMINATION OF CORE DAMAGE

CONDITION: Core Uncovered

Operator/System Actions with Successive Fallback Positions:

- High Pressure Injection (HPI) actuation
  - 2 independent systems (900 gpm max.)
  - Core Flood Actuation
    - 2 independent systems (15,600 gallons stored in accumulators) can remove more than 1 hour of equivalent core decay heat beyond 2 hours.
    - System then can be realigned with decay heat removal system for continuously supplying water to the reactor vessel.
    - Operation of this system at this stage is conditional on break size versus decay of primary system presure.
    - Low Pressure Injection (LPI) Actuation
      - 2 independent systems (6,000 gpm max.)
      - Operation of this system at this stage is conditional on break size versus decay of primary system pressure.
      - Align HPI/LPI Systems with containment sump
        - 2 independent systems (6,000 gpm max.)

RESULT: Core re-covered and cooled (<300 gpm total flow to core required for long-term cooling)

Chart 5

OBSERVABLES FOR CATEGORY 3

TERMINATION OF CORE MELTING

CONDITION: Core Melting

Primary Indications of Increasing Damage Potential (in order of occurrence assuming no prior corrective action)

1.- 6. Same indications as for prior condition (core uncovering--chart 3) but longer period of time with no corrective action.

- More individual indications of primary system temperatures exceeding boiling temperature, with these temperatures continuing higher.
- Primary system pressure has dropped drastically - e.g., less than 1/2 normal operating value.
- More individual containment area radiation monitors increase with signals progressively rising and, in some cases, saturating.
- Containment hydrogen levels could reach burn levels with burn(s) possibly occurring.

7. Core flooding actuates -- primary system pressure has decreased to $\lesssim$600 psig indicating that much of the core is uncovered.

8. Low pressure injection actuates -- primary coolant system has decreased to the point where higher pressures cannot be sustained and LPI actuates at $\lesssim$200 psig indicating all or nearly all of the core is uncovered.

Chart 6

ACTIONS FOR CATEGORY 3

TERMINATION OF CORE MELTING

CONDITION: Core Melting

Operator/System Actions with Successive Fallback Positions:

- High Pressure Injection (HPI) actuation
  - 2 independent systems (900 gpm max.)
  - Core Flood Actuation
    - 2 independent systems (15,600 gallons) stored in accumulation can remove more than 1 hour of equivalent core decay heat beyond 2 hours.
    - System then can be realigned with decay heat removal system for continuously supplying water to the reactor vessel.
    - Low Pressure Injection (LPI actuation)
      - 2 independent systems (6,000 gpm max.)
      - Align HPI/LPI Systems with containment sump
        - 2 independent systems (6,000 gpm max.)
        - Precautionary Containment Spray operated to predetermined level in containment.

RESULT: Core re-covered and cooled (<300 gpm total flow required for long-term cooling).

Chart 7

OBSERVABLES FOR CATEGORY 4

COOL CORE DEBRIS

CONDITION: Reactor Vessel Failure

Primary Indications of Increasing Damage Potential (in order of occurrence assuming no prior corrective action)

1. Primary system pressure suddenly drops to near zero - vessel has failed.

2. Containment building pressure and temperature suddenly rise - some core material has fallen into water-filled reactor cavity.

3. Containment area radiation monitors increase rapidly and significantly.

4. Core flood actuates.

5. Low pressure injection actuates.

6. Pressurizer level drops (dumps inventory into reactor vessel).

7. Containment building spray actuates and the building isolates (if not already isolated).

8. Containment building coolers actuate (if not previously actuated).

Chart 8

ACTIONS FOR CATEGORY 4

COOL CORE DEBRIS

CONDITION: Reactor Vessel Failure

Operator/System Actions with Successive Fallback Positions:

Core Flood Actuation

- 2 independent systems (15,600 gallons) stored in accumulation can remove more than 1 hour of equivalent core decay heat beyond 2 hours.

- System then can be realigned with decay heat removal system for continuously supplying water to the reactor vessel.

LPI Actuation

- 2 independent systems (6,000 gpm.)

Pressurizer Dump (≲11,000 gallons)

Pressurizer Spray Actuation

- Can be realigned to decay heat removal system for continuously supplying water to the reactor vessel.

Containment Spray Actuation - 3,000 gpm

Containment Coolers Actuation

- 5 independent units (70 mWth max.)

River Water Flooding

- As an ultimate heat sink, river water can be pumped into the containment building via installed or improvised piping arrangements.

RESULT: Core debris cooled.

Chart 9

OBSERVABLES FOR CATEGORY 4

COOL CORE DEBRIS

Primary Assurances that Core Debris is being and can be Cooled:

Reactor system is responding to previous conditions and actions.

1. Reactor primary systems such as LPI/decay heat removal are available for use as needed (to provide water directly into reactor cavity).
2. Containment building protection systems, containment spray and coolers, are available as needed.
3. Water pooled on containment floor provides intimate and continuous local heat removal capability for core debris.
4. Additional heat removal systems can be arranged external to containment and jury-rigged to existing hydraulic systems penetrating containment for backup if necessary.

As observables that cooling is established, containment pressure and temperature will stabilize and decrease with time.

Chart 10

ACTIONS FOR CATEGORY 4

MAINTAIN CONTAINMENT INTEGRITY

CONDITION: Core debris being cooled

Operator/System Actions with Successive Fallback Positions:

ESTABLISH REFLUX COOLING -

- Containment Spray for Initial Period -
  - Immediate pressure suppression
  - LPI and decay heat removal (2 indep. systems)
    - Pressurizer Spray and RHR (2 independent systems)
      - Pressurizer Spray and one of five containment coolers
        - Quiescent Pool Boiling and one of five containment coolers
          - Natural Convection - No Coolers Required after 50 days
            - Many similar options are also immediately available or can be arranged as required.

RESULT: Core debris cooling maintained; containment intact

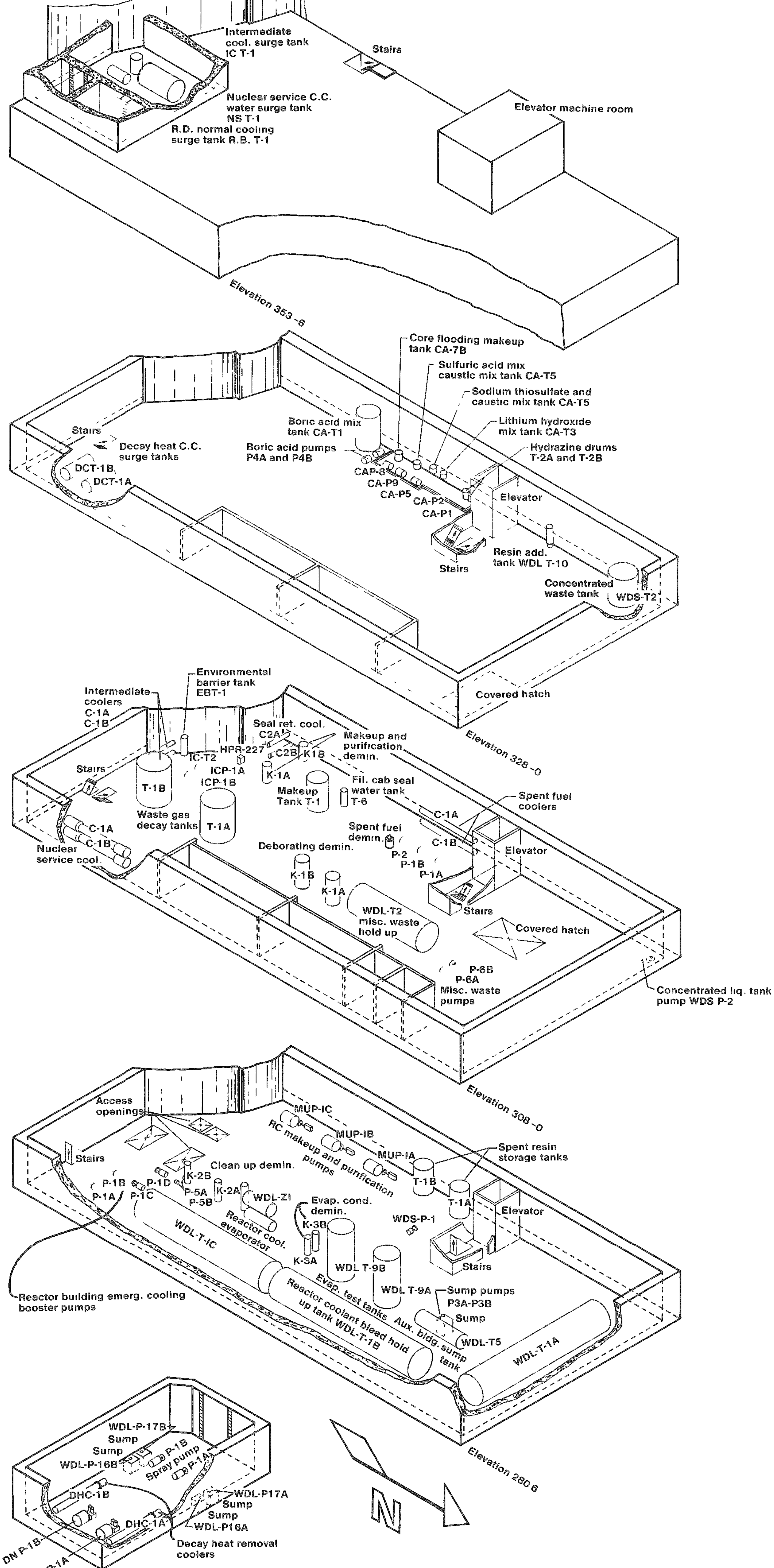
NOTE: The five containment coolers can handle 14 MWth each (70 MWth total)

At 2 hours, decay heat	~ 30 MWth
5	~ 22
20	~ 14
50	~ 9

APPENDIX AUX BLDG

AUXILIARY BUILDING

The accompanying Figure "Expanded Isometric of TMI-2 Auxiliary Building" displays the arrangement of the equipment in that building which is referred to in other parts of this report, or which may have figured in the release of radioactive effluents during or after the accident.



Expanded Isometric of TMI-2 Auxiliary Building

APPENDIX CF

CORE FLOOD SYSTEM

<u>Core Flood System Description</u> (Figure CF-1)

Core flood damage provides an Engineered Safety Features Function to limit fuel damage in the event of a LOCA by flooding the core with borated water. The system consists of two tanks located within the reactor building. Each tank outlet connects to one of the two flooding nozzles located in diametrical opposition to the reactor vessel above the core zone. Each of the tanks and its related equipment function as an independent circuit; however, both circuits are required for the system to meet its design requirements. Release of the stored water to the reactor core is independent of actuation signals, electric power supplies or operator action. The core flooding water is released by action of check valves open when the coolant system pressure is reduced below 600 psig. This pressure is maintained in the flooding tanks during normal operation by an overpressure of nitrogen gas. The loss of reactor coolant system integrity should ultimately lead to initiation of core flooding. Each tank contains approximately 7800 gallons of borated water at a minimum concentration of 2270 ppm boron and is pressurized with nitrogen gas to 600 ($\pm$25) psig.

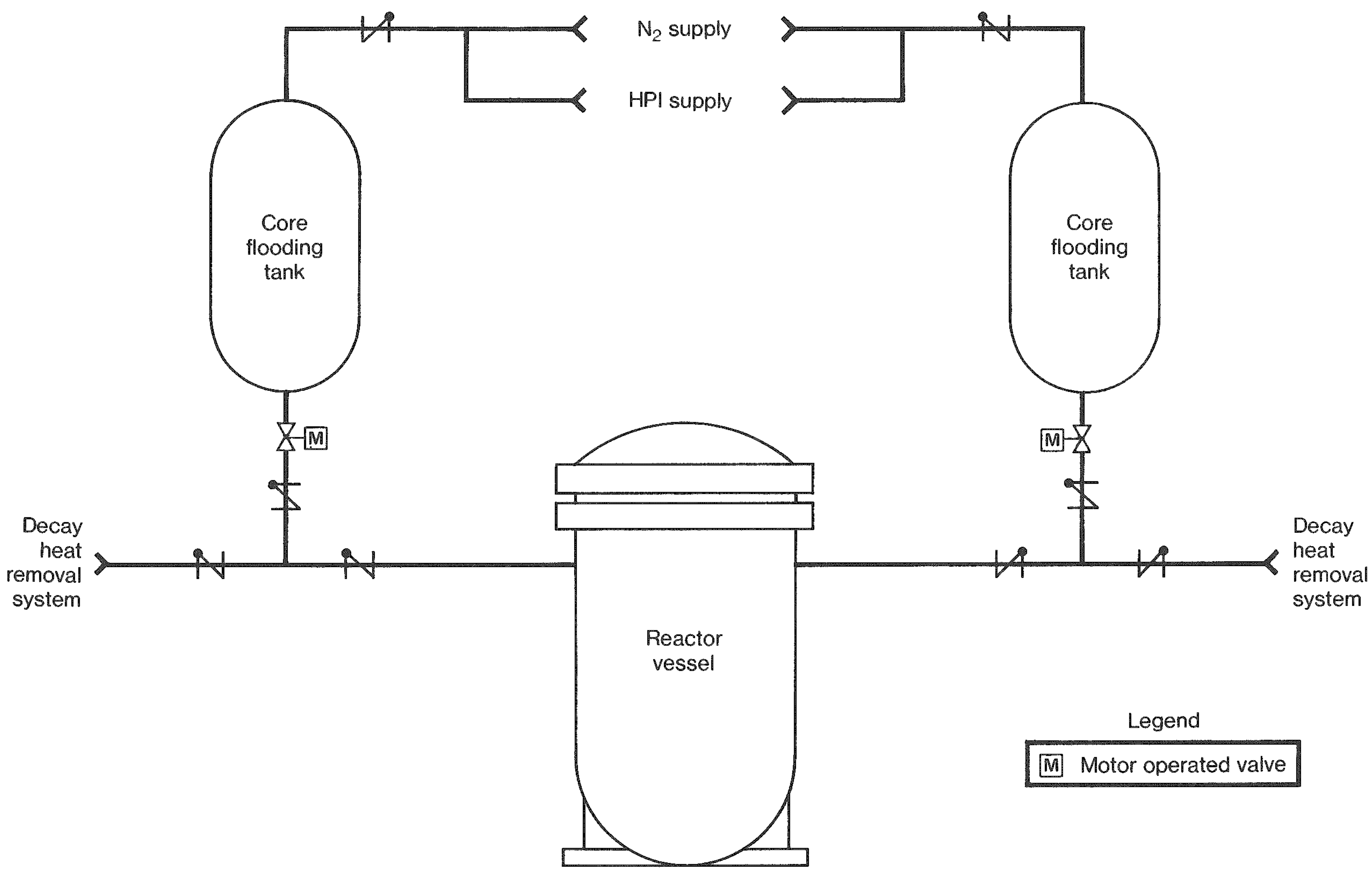

N2 supply
HPI supply
Core flooding tank
Core flooding tank
M
M
Decay heat removal system
Decay heat removal system
Reactor vessel
Legend
M Motor operated valve

Figure CF-1

APPENDIX C/FDW

CONDENSATE/FEEDWATER SYSTEM

1. Pre-event Status

Prior to the event, the plant was operating at approximately 97% power. The condensate and feedwater system (Figure C/FDW-1) was operating with two motor driven condensate pumps, two motor drive condensate booster pumps, and two steam turbine driven main feed-water pumps. Operation of the system was with the master automatic mode selector (AMS) switch in the MANUAL mode. In this MANUAL mode, automatic start of the third condensate and condensate booster pumps would not take place. Pumps in operation prior to the event were the following:

- Condensate pumps 1A (CO-P-1A) and 1B (CO-P-1B)
- Condensate booster pumps 2A (CO-P-2A) and 2B (CO-P-2B)
- Main feedwater pumps 1A (FW-P-1A) and 1B (FW-P-1B)

Condensate pump 1C (CO-P-1C) and condensate booster pump 2C (CO-P-2C) were off. It should be noted, also, that with the AMS switch in the MANUAL position, condensate and condensate booster pumps may be operated in any combination as opposed to paired pump operation (1A with 2A, 1B with 2B, 1C with 2C) with the AMS switch in AUTOMATIC.

Normal operation and control of the condensate/feedwater system utilizes both electrical and air service systems.

All air dependent systems appeared to be operating normally based on no pre-event indication of trouble. Systems included in this category were the following:

- o Main turbine exhaust hood spray system
- o Condensate polishing system (full-flow mixed-bed demineralizers)
- o Gland steam condenser system
- o Condenser hotwell level control system
- o Condensate booster pump recirculation system
- o Gland steam desuperheating system
- o Main feedwater recirculation system
- o Startup feedwater flow control system
- o Main feedwater flow control system.

In the case of the condensate polishing system, the plant operating staff had been working for about eleven hours prior to the event trying to clear clogged resins in polishing vessel No. 7. This clearing is normally accomplished by injecting a combination of compressed air and demineralized water into the polishing vessel to transfer the resins to a resin receiving tank on the regeneration skid. Compressed air is provided from the plant service air system at a normal 80-100 psig pressure. Demineralized water is normally provided at pressures equal to or exceeding 150 psig.

2. Cause of Loss of Feedwater

It is inferred that water was forced from the demineralized water system back into the service air system during the above period because of check valve leakage. Water from the service air lines entered the instrument air system through a crosstie valve which was open by design. This inference is substantiated by the fact that it took the plant operating staff approximately thirty minutes to drain the water out of all the air receiver tanks (instrument and service air). It is further judged that the water injected into the instrument air system traveled to the condensate polishing system control, a low point in the system, and caused the polishing system outlet valves to close. Stoppage of condensate flow to the feedwater system resulted. Closure of these valves was confirmed by plant operating personnel in the vicinity of the polishing system.

3. Pump Operation

From computer typewriter data, it is known that condensate pump 1A and feedwater pumps 1A and 1B tripped. Because the condensate/feedwater system was being operated with the AMS switch in the MANUAL mode, no trip indication from the condensate booster pumps was sent to the computer. This trip signal is only active with the AMS switch in the AUTOMATIC mode or with the switch in manual if a trip occurs within 5 seconds of pump start.

Upon trip of the main feedwater pumps, tripping of the main turbine-generator took place, as designed.

There is no evidence that condensate pump 1B tripped during the initial transient period. Trip contacts from the condensate pumps are connected to both the computer and the annunciator circuitry. The computer "memory trip review" log also showed a continuation of condensate pump discharge header pressure after the trip of condensate pump 1A. The most logical assumption is that condensate pump 1B continued operating. Condensate pump 1A was restarted at 00:05:15 in the event, and it continued to operate until approximately 00:08:58 when the computer typewriter indicated a TRIP signal. Apparently no attempt was made to restart the pump after this time.

Condensate booster pump operation during the loss of feedwater event is somewhat uncertain, although it appears that condensate booster pumps 2A and 2B may have tripped when the condensate polishing system outlet valves closed. Trip of the 2A pump could have caused condensate pump 1A to trip because of an apparent wiring error in the AMS switch.* Further substantiation is provided by the fact that condensate booster pump 2A was already stopped when its suction valve was manually closed by the plant operators to stop a leak in the pump seals. (The pump seal leakage may have been caused by a water hammer noted by the operators shortly after the event. This water hammer apparently caused a sizeable movement of the piping in the vicinity of the booster pump suction. This water hammer could have been caused by condensate overheating in the gland steam condenser.)

During this same period, the booster pump suction pressure was indicated to be just above the computer low alarm set-point of 15 psig. An alarm was printed out when the pressure dropped to 14.7 psig at 00:09:13 into the event. It is inferred that the booster pump suction pressure may have been high enough to allow condensate booster pump 2B to be started, but then an almost instantaneous trip of the pump occurred when the suction pressure could not be maintained under flow conditions. This event occurred three times between 00:05:15 and 00:06:29 into the event.

There were several attempts made from the control room to establish booster pump suction pressure by opening the condensate polisher bypass valve (CO-V12, Figure C/FDW-1). These attempts were unsuccessful because the high differential pressure across the valve set up a condition for which the valve operator was not properly adjusted. Opening of the condensate polisher system bypass valve was eventually accomplished manually by the plant operating personnel at approximately 00:59:12 into the event when the computer typewriter indicated a return to normal of booster pump suction header pressure. Also, immediately after the opening of the bypass valve a high temperature alarm was indicated in

*A recent TMI plant checkout of condensate pump 1A circuitry identified the existence of a jumper wire across the AMS contact in the trip circuit. The presence of this jumper would cause condensate pump 1A to trip on a low lube oil or low suction pressure trip of condensate booster pump 2A.

the condensate pump discharge header, which indicated flow through the condensate system. The water in condensate pumps 1B was being heated during the time the condensate polisher bypass valve was closed by operation of this condensate pump with no minimum flow. For a period of approximately ten minutes after the transient began, the condensate booster pump discharge line maintained a pressure in excess of the shutoff head of the condensate booster pumps. This pressure was apparently caused by leakage of water from the secondary side of the steam generator system back through check valves, and into the main feedwater and condensate systems. The condensate booster pump discharge pressure remained above the computer low pressure alarm setpoint of 310 psig until 00:15:43 into the event, at which time a low pressure alarm was typed out on the alarm typewriter.

4. Condensate System Water Inventory

Within a few minutes after the event began, there was concern over the rising water level in the condenser hotwell. Water level initially dropped below the low level alarm of 22.5 in. and then rapidly increased abovethe high level alarm of 36 in. This level increase is an expected occurrence following a loss of feedwater trip. The condensate system is designed to restore condenser hotwell water level to the normal range by opening the condensate reject line between the condensate booster pump suction header and the condensate storage tanks. (Condensate can then be pumped to the storage tanks using the condensate pumps.) The system did not work as designed because a severed airline to the feedwater pump suction pressure protection valve (CO-V57, Figure C/FDW-1) caused this valve to remain closed. Normal hotwell water level was eventually restored by manually opening valve CO-V57, manually opening the condensate polishing system bypass valve (CO-V-12), and then allowing the condensate reject system to function automatically as designed.

At approximately three hours into the event, a low water level alarm was received from condensate storage tank B. It can be assumed that this resulted from the condensate inventory reduction caused by the loss of steam which was released to the atmosphere during this three hour period.

Many condenser hotwell level signals were printed out by the computer typewriter during this period, with the condenser hotwell level first going high, then returning to normal, then going to a low level, and eventually returning to normal again.

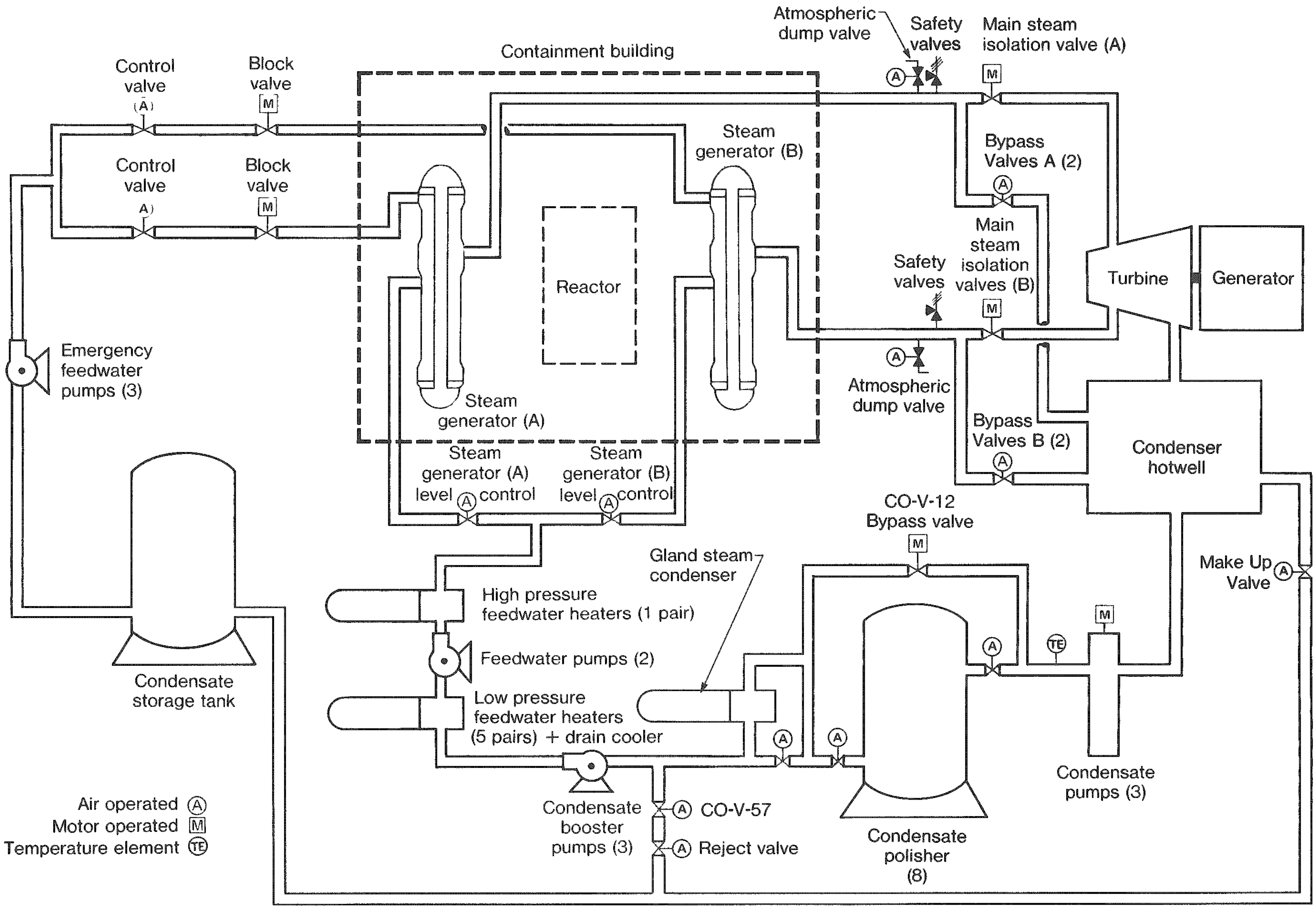


Figure C/FDW-1

APPENDIX CI

CORE INSTRUMENTATION AND ANALYSIS OF COOLANT LEVEL

PREFACE

Analysis of what happened to the Three Mile Island core is a complex, difficult undertaking which can be factually concluded only after the reactor vessel head has been removed and inspections carried forward. In spite of the obvious handicap, it is possible to examine plant data, make inferences about core conditions, and combine them with analytical tools to "scope" the damage sustained. This appendix represents the first step in such a process. It is an attempt to reconcile the known sequence of events, instrument data, and basic thermal-hydraulic principles to provide a self-consistent picture of what went on. The primary objective is to provide a refined set of boundary conditions to use as input for detailed thermal-hydraulic and core material analysis.

1. Introduction

The instrumentation arranged inside and immediately surrounding the reactor core provides important information regarding core thermal-hydraulic conditions during the accident. The instrumentation of interest consists of ex-core source and intermediate range neutron detectors, core exit thermocouples (52), and in-core self-powered neutron detectors (364). The location of the instrumentation is as shown in Figures CI-1 and CI-2. The different, somewhat complementary responses of each instrument to particular core conditions enable the development of a reasonably consistent (if somewhat qualitative) picture of core water level during the initial period. In this context, it should be recognized that these instruments have been utilized beyond their intended design applications; ex-core source-range detectors are employed as water level detection devices; in-core self-powered neutron detectors are used as temperature detectors and as a secondary indication of water level.

In the following subsection an inferred time history of core water level is described. Subsequent subsections describe the function of each instrument contributing to the analysis, including: functional description, relevant data, and inferences and judgments made therefrom. This preliminary analysis is based on thermal-hydraulic considerations.

2. Core Coolant Time History

The major period of concern over potential damage to the core starts at about 100 minutes after turbine trip. Up until this time, operation of the reactor primary coolant pumps should have assured adequate cooling of the fuel. The inferred time history which has been developed for the reactor core coolant after 100 minutes is illustrated in Figure CI-3. The critical parameter of interest is the steam-water mixture level, not the collapsed water level which would result from all voids or steam bubbles being removed. Below the mixture level, efficient heat removal occurs and fuel temperatures are kept within acceptable limits, i.e., near saturated coolant temperatures. Above this mixture level comparatively inefficient heat transfer takes place resulting in relatively rapid buildup of fuel and clad temperatures. The time history in Figure CI-3 is based on a transient thermal/hydraulic analysis of core boil down. The result has been baselined against the known sequence of events and the pertinent instrument responses described in this appendix.

The potential for core damage developed as primary coolant continued to be lost through the stuck-open electromatic relief valve. This produced a steady increase in steam voids throughout the primary system and core, as illustrated by Figure CI-4. However, forced coolant flow and the homogeneous distribution of the steam voids prevented core overheating. When the operator secured the second set of reactor coolant pumps RP-C-IA and RP-C-2A at 100 minutes, flow stopped and steam collected at the high points of the primary system. The liquid phase collected in three primary system low points: the core, loop B and loop A.

The vessel downcomer water level settled out at about a foot above the top of the active core. The core region entered a "pool boiling" mode, with the phase disengagement or two phase mixture level positioned 3-1/2 to 4 feet above the top of the active core (approximately 102 min.). Make-up to the core was insufficient to compensate for boil-off and the levels in both the core and downcomer began to drop. The downcomer water level was always lower than in the core because of the steam void content in the core region. At approximately 113 minutes the top of the core began to dry-out, causing superheated steam temperatures to be sensed in the A loop hot leg. Prior to this time, modest superheat conditions are attributed to thermal pick-up from upper vessel internals and piping as saturation temperatures fell off.

Some of the steam produced in the core during this time period was condensed in the pressurizer and passed through the stuck open relief valve. However, a substantial fraction of the steam produced in the core was condensed in the A loop steam generator (OTSG). This was caused by the addition of cold feedwater to the steam generator secondary side in preparation for attempted natural circulation cooldown.

Steam condensation in the A OTSG facilitated core heat removal; however, the loop A liquid level was too low for the condensed liquid to complete a reflux boiling path by spilling back into the core. This caused a core mixture level drop in excess of that expected if coolant loss through the stuck open relief valve were the sole consideration.

The boiling-condensation transfer of coolant inventory between the core and the A loop is supported by boron analyses made over this period. Primary samples are drawn from a low point in this A primary loop rather than the core, and would reflect dilution produced by condensation of boron-free vapor into the A loop side. Dilution factors on the order of 2.5 to 3.0 are estimated, suggesting substantial coolant transfer out of the core, into the A loop side. This observation tends to lend credence to the argument that significant core boil-off occurred during this period.

The B OTSG played a minor role in the core boil-down process. The steam generator had been isolated due to a suspected leak. Very little heat transfer to the steam generator was exhibited, suggesting that reflux boiling through the B primary loop was not established to any substantial degree.

The coolant level in the core continued to drop as in Figure CI-3, uncovering self-powered neutron detectors (SPNDs) at the 9.5 foot level. As these detectors heated up they began to respond thermionically (after 135 minutes). Continued core uncovery and heat-up of exposed fuel elements resulted in fuel cladding failures after 142 minutes. Additional self-powered neutron detectors began to respond thermionically as the coolant level in the core reached lower levels.

Detailed analysis of make-up/letdown system data, operator interviews, and the plant response indicate make-up flow rates in the neighborhood of 110 gpm over this period of time (113 to 140 minutes). In addition, natural condensation recirculation through the vessel vent valves is believed to have contributed approximately 30 gpm makeup.

Transient core thermal hydraulic simulations using these make-up rates provide SPND response and fuel failure times that are consistent with the available data. The uncovery history developed using these make-up rates also compares favorably with a core coolant level reconstruction, based on neutron transport analysis of the source range neutron detector behavior.

After 140 minutes the make-up rate is believed to have been reduced to about 70 gpm, with a corresponding reduction in local condensation/recirculation. The core coolant level continued to fall monotonically after the PORV block valve was closed (142 min.).

For the time period after 160 min., a net reduction in make-up to the core was caused by the production of hydrogen from the zirconium-water reaction, interfering with recirculation and condensation through the vent valves. However, there is evidence for increased make-up to the core (approximately 120 gpm) beginning at about 160 minutes, which was supplied by greater letdown flow drawn from the A loop side. The net result was a trend towards a stabilized coolant level at about 3.0 ft. (Ref. Fig. CI-3).

By 174 minutes it is believed that the core coolant level had reached a minimum value, based upon estimated system make-up and analysis of the source-range neutron detector response. The operator started the 2B reactor coolant pump. Operation of the 2B reactor coolant pump produced only momentary flow, as noted on core thermocouples located near peripheral fuel assemblies and indicated in Figure CI-3. A substantial volume of coolant was added to the vessel (approximately 1000 ft^3). This is believed to have quenched the core and effectively terminated the auto-catalytic $Zr-H_2O$ reaction which had been producing high fuel temperatures.

The actual core coolant level perturbation produced by the reactor coolant pump operation is not well characterized. The rapid recovery of the ex-core neutron detector signal after the pump is started suggests that the downcomer was not completely filled and that forced flow through the core was not established. It is believed that flow into the core was caused by gravity acting on the difference between downcomer and core water levels (gravity head). Quenching of exposed lower core regions would have produced vigorous boiling with entrainment of water droplets in steam flow up the fuel channels. Quenching would preferentially occur in peripheral fuel assemblies. At any rate, it appears that a substantial portion of the coolant that was added to the core was boiled off and condensed in the pressurizer and in the steam generators.

At approximately 192 minutes the operator opened the electromatic relief block valve and core boil-off continued. Pressure decay was rapid; the electromagnetic relief valve block valve was closed about five minutes later; and safety injection was initiated manually an additional two minutes later. Subcooled liquid moved into the core and began to quench exterior regions. The operator continued to add coolant to the system, though at a reduced flow rate over the next seventeen minutes (to 217 minutes elapsed).

The mixture level following the 174 minute mark is poorly defined, and is only schematically indicated in Figure CI-3. Make-up to and partial quenching of exterior core regions may have eventually resulted in unstable thermal-hydraulic conditions, leading to possible distortion of the core geometry at approximately 226 minutes or 3 hours 46 minutes elapsed time. This interpretation is derived from ex-core and in-core neutron detector data and thermocouple readings during (and subsequent to) this time period. Analysis of possible fuel dislocation and damage subsequent to the 226 minutes mark is continuing. Some evidence exists for one or more additional periods of core uncovery later in the accident. This matter is presently under study; however, it is believed that most damage to the core occurred during the initial period of core boil-off (100 minutes to about 226 minutes after turbine trip).

The core water level is not a directly measurable parameter and therefore a water level time history must involve some measure of speculation. The foregoing description represents an attempt to reconcile the interpretations of instrument recordings with simple heat transfer relationships. Refinements and additional analyses might lead to revisions to this event description.

3. Analysis of Ex-Core Neutron Detector Data

Ex-core neutron detector data provides a major resource for understanding core conditions during the early stages of the TMI accident. The importance is threefold: (1) continuous recordings of instrument signals exist; (2) detectors are out-of-core and therefore are assumed to be protected and operating normally; (3) analytical tools exist for relating detector readings to in-core thermal-hydraulic conditions. As will be discussed, these detectors are considered responsive to the presence or absence of liquid water in the downcomer and core regions near the axial location of the detectors. The absence of liquid in either region affects detector response to neutrons lost from the peripheral core regions.

4. Ex-core Neutron Detector Description and Operation

The ex-core nuclear instrumentation system consists of three ranges of detectors (source, intermediate, and power) to monitor neutron flux levels overlapping approximately twelve decades of power change. The source-range detectors are the most sensitive of these instruments, and will serve as the principal subject of this analysis.

The two source-range instruments are used for shutdown power level monitoring and also for the initial stages of reactor startup. They are situated at diametrically opposite sides of the core at the midplane (6-foot level). Refer to Figures CI-1 and CI-2. The 26-inch detector chambers are filled with boron tri-fluoride (BF_3) gas which interacts with neutrons leaving the core region. Neutrons are electronically discriminated from gamma radiation because of the greater energy produced by neutron interaction with gas molecules.

When the reactor is in the subcritical shutdown mode, there are not enough neutrons being produced to sustain the fission chain reaction. Without some kind of extrinsic neutron source, the fission process will eventually cease. Two types of source neutrons were present in the TMI core at the time of the accident. One type, photo-neutrons, were produced by the interaction of gamma radiation from fission product and heavy water molecules. The other type was the neutrons from the two start-up neutron sources (Americium - Beryllium - Curium) which were used to supply neutrons to the core for its first criticality. These source neutrons make a negligible direct contribution to the ex-core neutron detector signals, however, as initiators of fission-chain reactions (which produce more neutrons in turn), they were ultimately responsible for the neutrons that were detected.

Steam voids in the core produce several competing effects: (1) voids displace boron, contributing to greater numbers of fissions in local regions; (2) voids decrease water density, reducing the moderating effect necessary for neutron-fuel interactions; (3) voids enable more neutrons to leak out of the core and escape the fission-chain process entirely. The latter effect is the most important insofar as ex-core detector count rates are concerned. Although fewer neutrons remain in the core to help sustain the fission process, many more are able to escape to the detector for measurement.

It is important to recognize that voiding different regions of the core will have a varying effect on ex-core detector readings. For example, voiding the center of the core may affect the neutron population in that vicinity; but, any change will be "shielded" from the detector by peripheral fuel assemblies and fluid in the downcomer annulus. Conversely, voiding the downcomer region adjacent to the detector will have a major effect, since neutrons reaching that region will be able to travel largely unimpeded to the detector.

One-dimensional neutron transport calculation suggests that voiding the downcomer annulus will result in a count rate increase by a factor of 400, and is the dominant mechanism by which source- and intermediate-range neutron detector signals are affected (under these conditions).

When reactor coolant pumps were operating (during the initial phase of the accident) the steam voids in the down-comer were homogeneously distributed. The ex-core detectors essentially responded to fluid density in the downcomer region. When the pumps were stopped, however, the steam and liquid separated and the detector response was somewhat more complex. The interpretation of this response is, at this point, speculative.

It is useful to consider the downcomer liquid as a shutter - the level determines the peripheral region of the core that the detector is able to see. The problem is that above the core mixture level, fission chain reactions do not readily occur, because of the absence of water and its moderating effect. Moreover, the absence of water is accompanied by a reduction in photo-neutron source strength. This would tend to argue for a lower count rate from the uncovered core region.

There is, nevertheless, a modest difference in height between the core mixture and downcomer levels. This is due to the steam void content within the coolant mixture when the core is boiling down (refer to Figure CI-6). The extended layer of mixture comes increasingly in view of the detector as the water levels drop. Some neutrons emanating from the top surface of the mixture interact with exposed fuel above it, producing additional neutrons and modest neutron multiplication. Effectively, the water serves as a neutron source to "drive" the subcritical multiplication process in the upper, "bare" core region.

Although neutron multiplication in the bare core region is reduced, the neutron source strength is not reduced as much. When core water levels are high the gain in detection efficiency caused by voiding the downcomer more than compensates for the reduction in neutrons produced in the bare core region. Accordingly, it is believed that the detector count rate should increase as the core begins to uncover. This interpretation is supported by two-dimensional neutron transport (ANISN) calculations.

With continued core uncovery the loss of neutron source (i.e., removal of H_2O from the core region) begins to compensate for the shielding variations in the downcomer; the count rate reaches a maximum value. Thereafter, the count rate will fall as the water level decreases.

This behavior presents a major difficulty in water level interpretation, since the neutron transport analysis cannot distinguish core refill from continued boil-down after a certain point. The difficulty may be resolved by referencing other data (SPND, make-up/letdown) to provide a clue as to the proper coolant level trajectory. In this case (i.e., over 100-174 minute interval) it is believed that the core coolant level continued to fall in a fairly monotonic fashion. (Reference Fig. CI-3)

5. Interpretation of Source-range Recording

By applying an understanding of the source-range detector response to void formation, basic thermal-hydraulic principles, and the established sequence of events, it is possible to develop an interpretation of the source-range recording, shown in Figure CI-7. The relevant features of the recording are designated by letters and are referenced in the commentary that follows:

A. For the first 20 minutes, source-range instrument behavior was consistent with a normal post-trip decay rate of about one-third decade per minute.

B. After approximately 20 to 30 minutes, the source count rate should be decreasing through the 600-700 counts per second (cps) range. Instead, the curve leveled out at about 5000 cps due to buildup of voids (steam bubbles) in the downcomer and core regions. This is consistent with the fact that pressure had reached saturation (approximately 6 minutes after turbine trip), and net outflow through the open electromatic relief valve continued to empty the system. Void formation is also consistent with the observed drop in reactor coolant flow rate because of the reduced pumping head produced by two-phase flow conditions (not shown).

C. Continued loss of coolant from the primary system leds to increased voidage and increased detector count rates. The recording began to exhibit noise, which is reflective of unsteady flow (pump surging) and phase separation characteristic of "slug flow". This phenomenon increased with time.

D. At 73-74 mintues the B reactor pumps were secured by the operator.

E. At 100 minutes the A reactor coolant pumps were secured. This caused a flow transient and separation of voids to the upper regions of the system. Voids rising to the top and coolant fill from the hot legs produced a "solid" water condition seen at the detector. The detector count rate abruptly dropped.

F. The minimum count rate is suggestive of the fact that the downcomer water level was at or near the top of the active core level.

G. Continued release of fluid out of the relief valve began to boil off inventory from the core and downcomer area. Makeup flow, assumed to be in the neighborhood of 140 gallons per minute, was not sufficient to maintain downcomer water level. The core mixture came within better view of the detector as the water levels dropped; the count rate increased.

H. The signal level continued to increase but at a slower rate as shielding variations began to be counterbalanced by the loss of neutron source (i.e., UO_2 being removed from the core region). In this period also, the rate of uncovery is believed to have slowed somewhat as core boil-off tended to equilibrate with a relatively unchanging make-up flow rate.

I. Over this period the count rate was decreasing as the loss of neutron source and reduced neutron multiplication became predominant. The turn-around in detector count rate can be interpreted either as core refill or continued uncovery. However, the weight of evidence from make-up flow estimates and other core instrumentation suggests that the decreasing count rate was in response to continued core uncovery.

J. The operator started reactor coolant pump 2B, sending a slug of cold water into the downcomer and essentially filling it.

K. Loop flow data indicates that the pump worked effectively for a very brief period. This is corroborated by the abrupt turn around in the source range detector trace, as flow ceased and excess downcomer fluid moved into the core and was boiled off; equilibrium levels were re-established.

L. High pressure injection flow was initiated at 200 minutes, 8 minutes after the electromatic relief block valve was opened by the operator. Coolant passed into the downcomer, filling it. Detector count rates dropped sharply.

M. Continued addition of high pressure injection flow began to quench the core. It is conjectured that the coolant first re-wetted the outer region of the core, bypassing the hot center.

N. Water entering the core eventually led to an unstable thermal-hydraulic condition. It is speculated that major portions of the core coolant may have been suddenly quenched with a resulting large amount of coolant flashing to steam, accompanied by possible core and fuel rearrangement. The jump in detector counts may be due to the displacement of fuel and/or sustained voiding of peripheral fuel regions.

6. Analysis of Core Thermocouple Data

Core thermocouples are installed in the outlet region of 52 instrumented fuel assemblies and, under normal circumstances, provide accurate readings for core outlet temperature conditions. All but one thermocouple appear to have survived the accident. Variations between installation and post-accident resistance measurements of the thermocouple junctions are inconclusive.

The value of the core outlet data is the possibility of inferring core spatial temperature distribution during the accident sequence. Unfortunately, the scope of these data is limited because: (1) with one notable exception in the approximate 240 to 330 minute time period, wide-range temperature readout (>700°F) was not available due to pre-set scale limits; (2) the alarm printer failed and/or the process computer failed to retain records during a crucial phase of the accident, causing an irretrievable loss of temperature data.

7. Core Thermocouple and Instrument Assembly Arrangement

Chromel-alumel thermocouples are inserted into an instrumented channel in the 52 selected fuel bundles (Figure CI-2). The thermocouples are part of a sheathed instrument assembly, shared by a string of seven self-powered neutron detectors (SPNDs) and a reference flux wire (in the general arrangement shown in Figure CI-1). The instrument assembly passes from an instrument pad in the reactor building, through the pressure vessel and upwards into the fuel bundle. The physical (axial) location of the thermocouple junction is six inches above the active part of the core.

8. Interpretation of Core Thermocouple Data

The apparent survival of the thermocouples may be indicative of limited core damage. However, there are contrary views regarding the matter. In contrast to the self-powered neutron detectors, the thermocouples are less sensitive to water intrusion (sheath degradation), which may have occurred. The possibility of thermocouple junction melt and reformation has also been raised in connection with the observed time-dependent temperature patterns. Post-accident thermocouple lead wire resistance measurements and time-domain reflectometry measurements are inconclusive regarding a definitive measurement of damage to these thermocouples. Analysis of potential thermocouple failure modes suggests that their survival as effective temperature sensitive devices is likely; however, the axial location of measurement and absolute calibration may be in doubt. This matter will bear further investigation.

The loss of data records in the computer output buffer restricted the availability of temperature data between 73 and 168 minutes. Thereafter the core temperature map, Figure CI-8, shows a large fraction of thermocouples reading above the 700 F recording limit. The map represents a top view of the core; each element denotes an individual fuel assembly, which is designated by letter and number loci. The temperature distribution shown in the figure is consistent with core uncovery and the substantial fuel and clad temperatures believed present in the upper regions. Subsequent core temperature mappings, Figures CI-9 and CI-10, show a gradual cooling trend, starting from the core periphery and working towards the central fuel assemblies. The temperature mappings do not show the brief temperature perturbation produced by reactor coolant pump operation, commencing at 174 minutes (Item K of Section 5). Cooling of peripheral (outer three to four) assemblies was indicated by exit thermocouple readings dropping below 700 F for a short period (10 to 15 minutes) after the start of reactor coolant pump operation.

The long-term cooling process is illustrated in Figure CI-11 by a plot of percentage of in-core thermocouple readings in excess of 700°F versus time. Notable is the fact that some thermocouples continued to read off-scale (above saturation temperature). The high thermocouple readings may be due to packing of fuel around core instrument assemblies with an inability to establish effective cooling in those regions. This may have the effect of producing virtual junctions and temporary de-calibration of the instruments. Also, as noted, the point of measurements for degraded thermocouples may be in doubt; and these elevated temperatures may, in fact, be representing conditions within rather than above the original fueled region of the core.

The majority of core thermocouple data has been reduced from the utility printer which was then serving as an alarm printer. The printer only reported when an alarm setpoint was exceeded, when the monitored parameter dropped back below the alarm setpoint, or when an operator request for data was initiated. The core mappings in Figures CI-8 to CI-10 therefore indicate general trends only.

A more representative set of measurements was obtained directly by a Metropolitan Edison instrument engineer, using a digital voltmeter, approximately between 0800 and 0930 hours during the morning of the accident. These readings are shown on the core map provided in Figure CI-12. The picture is one of a chaotic core situation; subcooled regions are noted at the core exterior with extremely high temperatures at the center. Remarkable temperature variations between adjacent fuel assemblies were observed, e.g., almost 2000 F temperature difference indicated between assemblies M9 and M10. This is hardly credible unless one conjectures either: (1) substantial movement of fuel out of assembly M10 into the vicinity of M9, blocking cooling in those regions and contributing to the heat load; or (2) variation in thermocouple readings attributed to thermocouple deterioration, resulting in the formation of multiple junctions or "virtual" junctions at various axial locations. (For example, temperatures shown in assemblies E10 and H5 of Figure CI-12 are very near melting temperature for the Inconel sheaths of these thermocouples.)

An important observation which can be made from Figure CI-12 is the presence of subcooled temperatures at peripheral fuel assemblies. This would suggest that the water level may have risen to above the top of the core sometime prior to 0900 hours.

9. Analysis of In-core Self-powered Neutron Detector Data

The in-core self-powered neutron detectors (SPNDs) are installed for the purpose of measuring neutron flux in the core to provide a history of power distribution during power operation. As indicated in Figures CI-1 and CI-2, there are seven SPNDs in each of the 52 instrument channels which provide a detailed (364 point) spatial representation of core power distribution. During periods of core uncovery at TMI-2, these detectors responded to abnormally high temperatures, rather than neutron flux levels present in the subcritical core configuration. The responsiveness of the SPNDs to temperature provided core spatial information that can be cross-referenced with core exit thermocouple and ex-core neutron detector data.

Most of the SPND data were obtained from alarm and utility printer output. As previously noted, this data resource is limited by overloading of the process computer output buffer, and failure of the alarm printer during the accident; also limiting was the irregular output of SPND data generated as a function of a detector going off or coming back on scale. Studies are still underway which may substantially change the interpretation of the SPND data.

The printer output was supplemented by multipoint recordings of 36 SPNDs over the same time period. This offers some data continuity; however, the interpretation of the multipoint readings by individual channels has turned out to be a difficult task.

An important consideration derives from the ultimate failure of a substantial number of SPNDs, as determined by post-accident diagnostic examinations. The distinction of "valid" from "failed" detector readings is a problem that may be difficult to resolve.

10. <u>In-core Self-powered Neutron Detector Description and Operation</u>

The self-powered neutron detector consists of a coaxial cable with a section of the central conductor replaced by a material (rhodium) that emits energetic elesponsiveness of the SPNDs to temperature provided core spatial information that can be cross-referenced with core exit thermocouple and ex-core neutron detector data.

A design consideration in the development of SPNDs is the competing effect of temperature, which may influence current readings. The thermionic effect permits electrons to move away or diffuse from the host emitter to the collector. Different thermionic properties of emitter and collector materials permit a net, if rather weak current to flow. The higher the temperature the more energetic electrons will be able to overcome the restraining effect of the emitter's intrinsic electric field and diffuse to the collector.

Under normal reactor operating conditions, thermionic emission accounts for only a few picoamperes. However, the thermionic effect varies exponentially with temperature, as shown in Figure CI-13.

The elevated temperatures present in the TMI-2 core following uncovery should have resulted in significant SPND currents by thermionic effect.

The apparent responsiveness of SPNDs to temperature suggests the possibility of converting detector currents to "measured" temperatures by suitable calibration. However, there are major difficulties with this approach. The foremost difficulty stems from the fact that only weakly positive thermionic currents have been developed by SPNDs when subjected to high temperature furnace tests. Strongly negative thermionic currents were observed, in contrast to the predominantly positive SPND behavior exhibited on back-up (B/U) trend recorders at the time of the accident.

A careful examination of B/U recorded data shows evidence of negative current thermionic behavior after 135 min., as signals were being driven hard-limited in the negative direction. Assuming that these SPNDs had undergone some thermal conditioning prior to the accident, it is possible to conclude that the uppermost recorded SPNDs (level 6) were undergoing a shift between the weakly positive to the strongly negative thermionic behavior at the 135 minute mark. Furnace test data suggests that this transition occurs when SPND temperatures exceed approximately 1000°F.

The subsequent, rather erratic positive trending on SPNDs was not duplicated under furnace tests. Quite possibly these currents developed after thermal failure of the SPNDs brought about by major structural changes in the detector, sheath, or lead wire configurations. Furnace tests indicate that thermally-induced failure will occur in the 2400-2600°F temperature range.

11. Interpretation of the Self-powered Neutron Detector Data

When the reactor tripped at TMI, all SPND responses dropped to a few nanoamperes as expected. The 36 SPND signals available from B/U recorders remain quiescent for the first 135 minutes. During the next 15 minutes there is clear evidence of some SPND signals being driven hard negative (currents less than minus 20 na) by thermionic activity. This behavior would be exhibited by SPNDs first uncovered at the top of the core. The level 6 SPNDs on the B/U recorder are positioned at approximately the 9 ft level (measured from the bottom of the core). Using furnace test results, it might be concluded that the first negative SPND currents at this level are indicative of steam and SPND temperatures in excess of 900°F.

The positive shift in SPND currents that followed (after 150 min.) have not been explained by furance test results. However, these large positive currents may be indicative of SPND failure. B/U recorder data shows upscale swings in SPND currents for detectors at the 9.5 and 6.0 ft. levels before 160 minutes. During this first period of SPND activity in the top half of the core, the detectors in the lower levels (at 9 and 30 inches above the bottom of the core) remained inactive. This is confirmed by B/U recorder as well as alarm data. The interpretation is that the bottom one-third of the core was cooled and probably covered during most of the first 226 minutes. The upper two-thirds of the core was uncovered for an extended period, with local steam temperatures in excess of 1000°F.

At about 226 minutes into the accident (07:47), something traumatic happened in the core. SPNDs throughout the core went offscale, possibly indicating a rapid temperature increase. Some of the SPNDs located in the central region of the bottom of the core (levels 1 and 2) remained offscale only a few minutes then returned on-scale. The heat-up rate above level 3 seems to indicate an exterior to interior heat-up pattern.

It is difficult to postulate core conditions without a clear spatial picture of SPND behavior. The 07:47 event does occur while substantial makeup was being supplied to the system. The refill process and the possible development of unstable thermal-hydraulic conditions in the course of refill is under investigation. At any rate, the accumulated degradation of fuel cladding and high temperatures (at this point) may have set the stage for some disarray and fuel dislocation by quenching. This would explain the SPND behavior after 226 minutes, as well as the core exit thermocouple readings that were obtained.

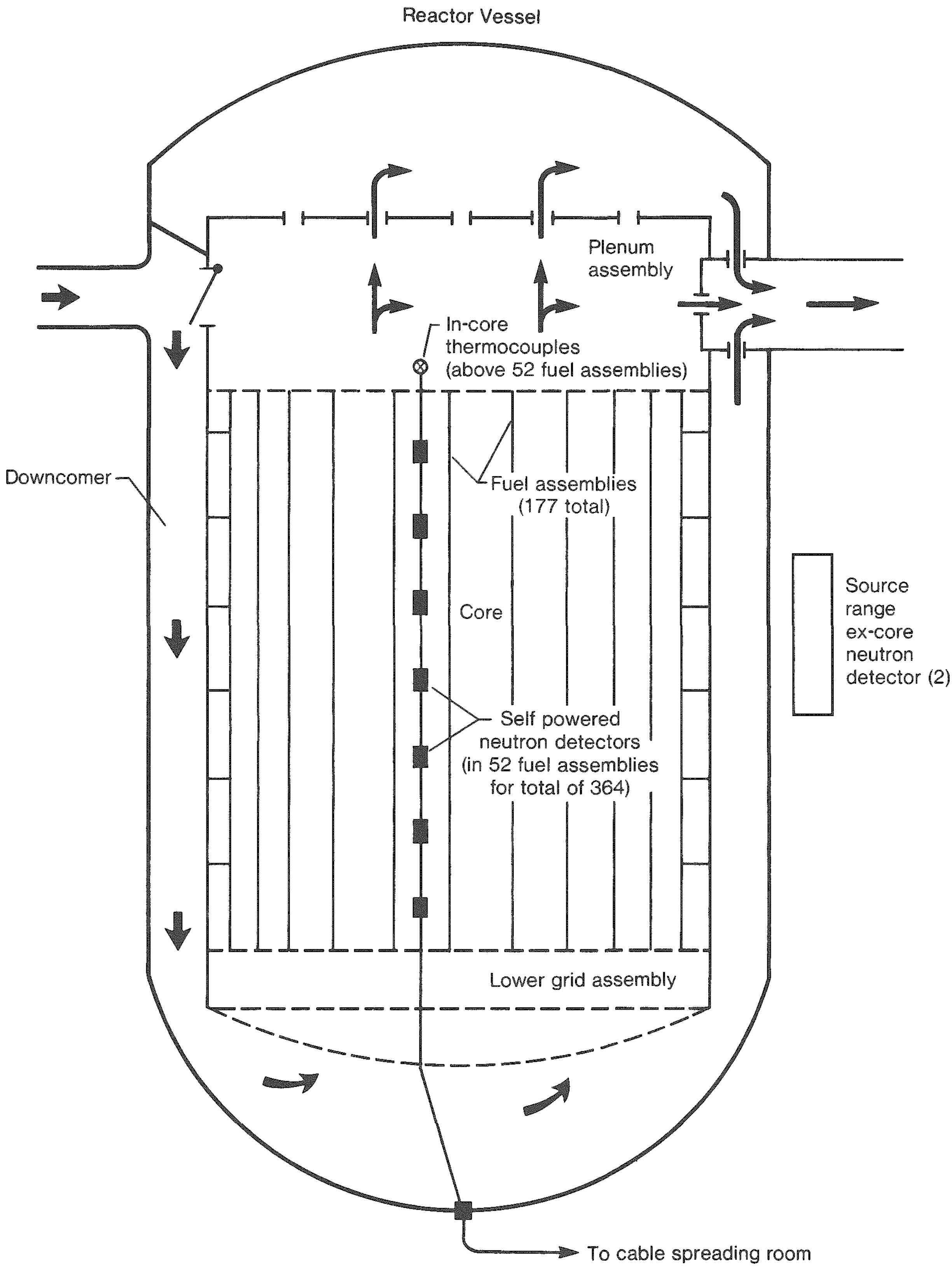


Figure CI-1. Basic reactor configuration and instrument locations.

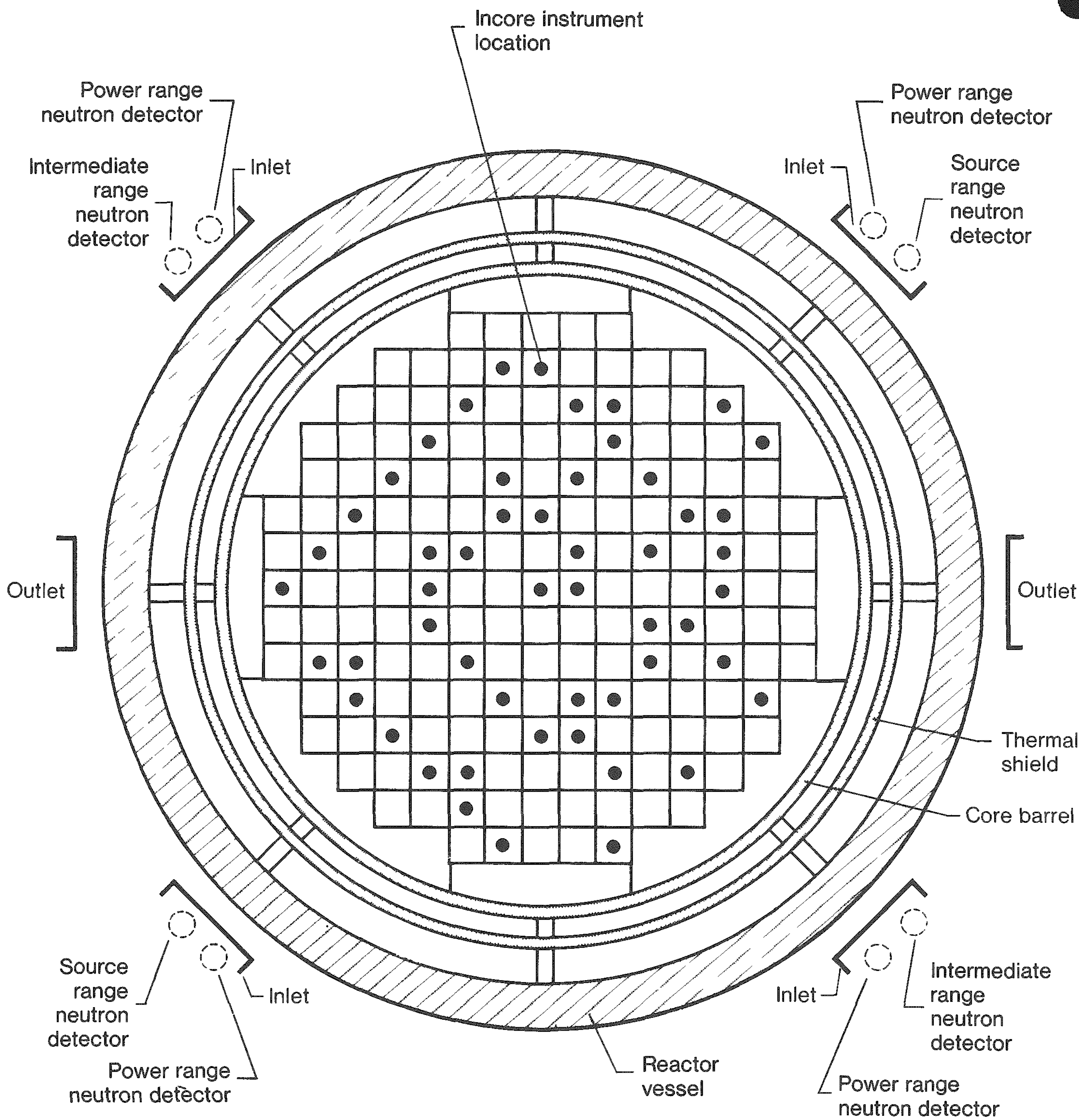


Figure CI-2. Top view of reactor core.

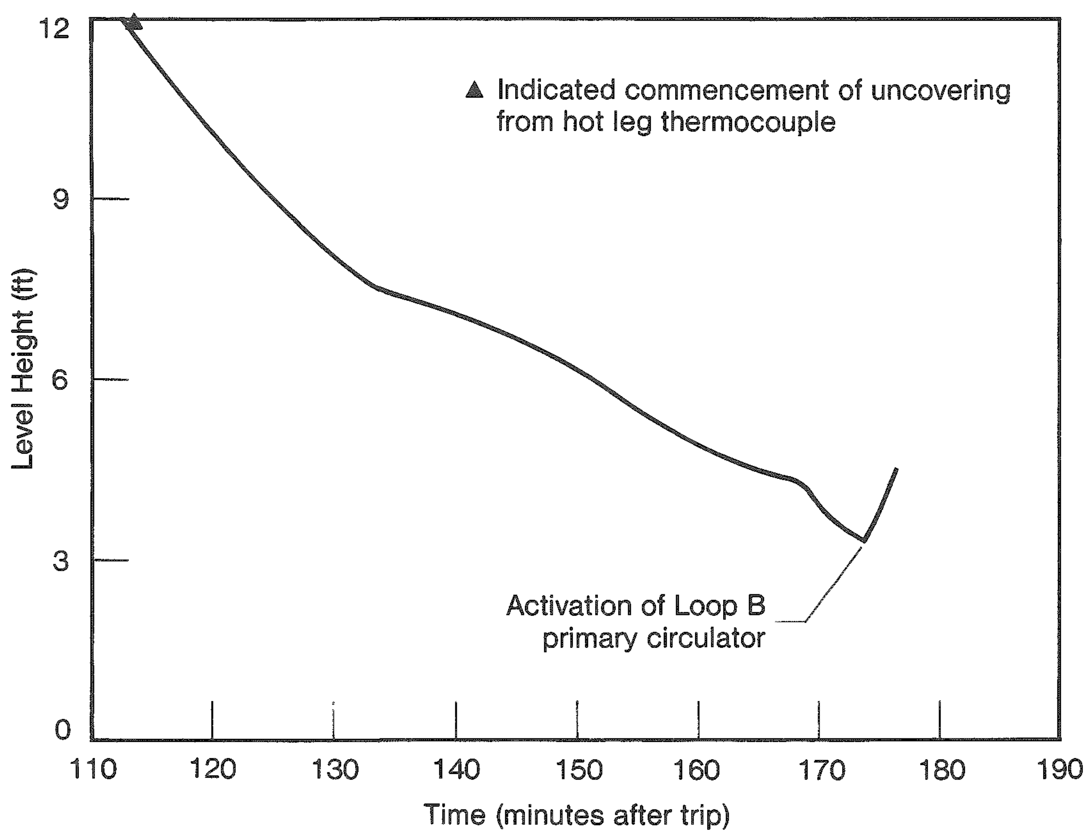


Figure CI-3. Calculated mixture level during uncovering of TMI-2 core.

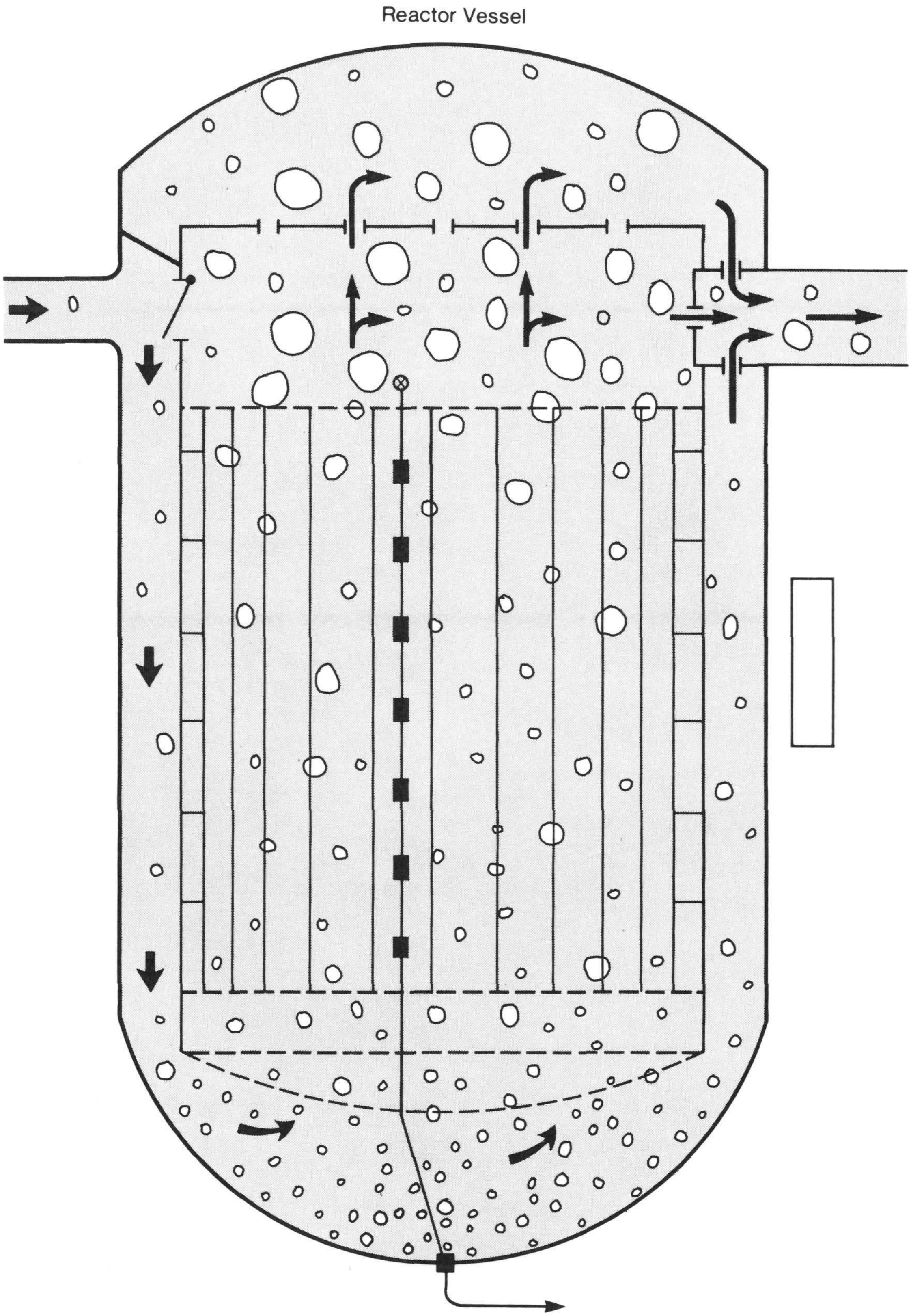


Figure CI-4. Forced core cooling; homogeneous void distribution.

Reactor Vessel

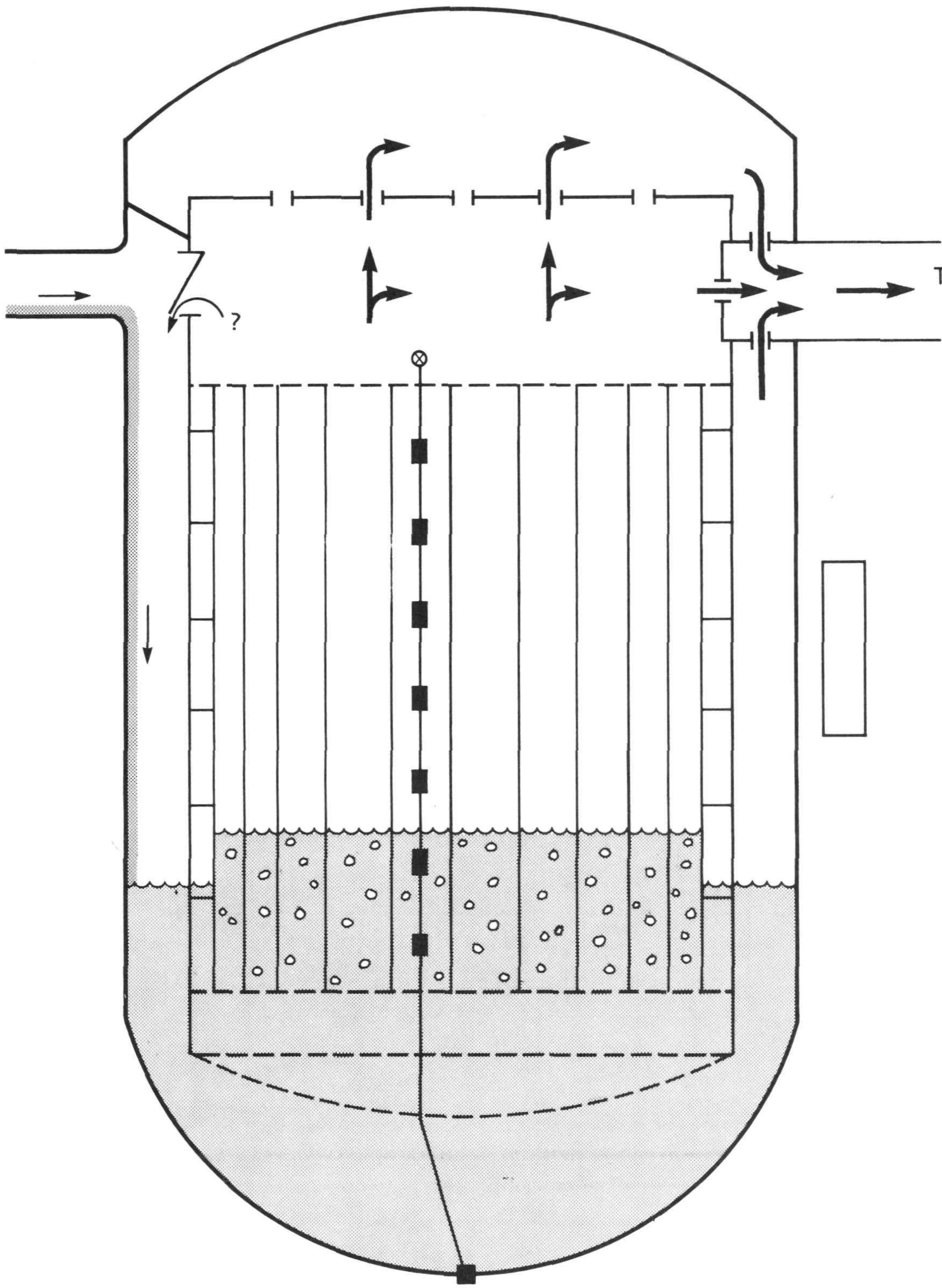


Figure CI-5. Core water level — time approximately 160–174 minutes after turbine trip.

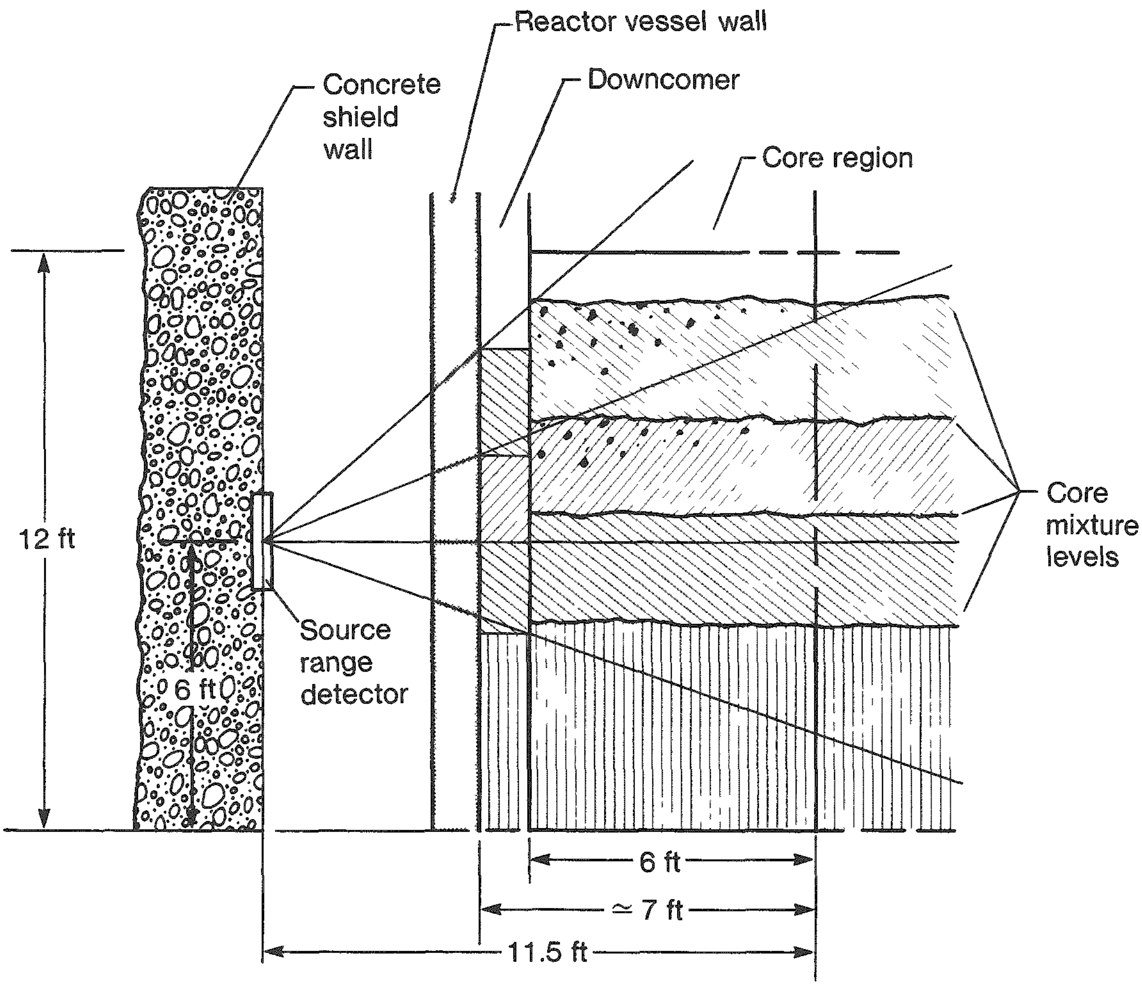
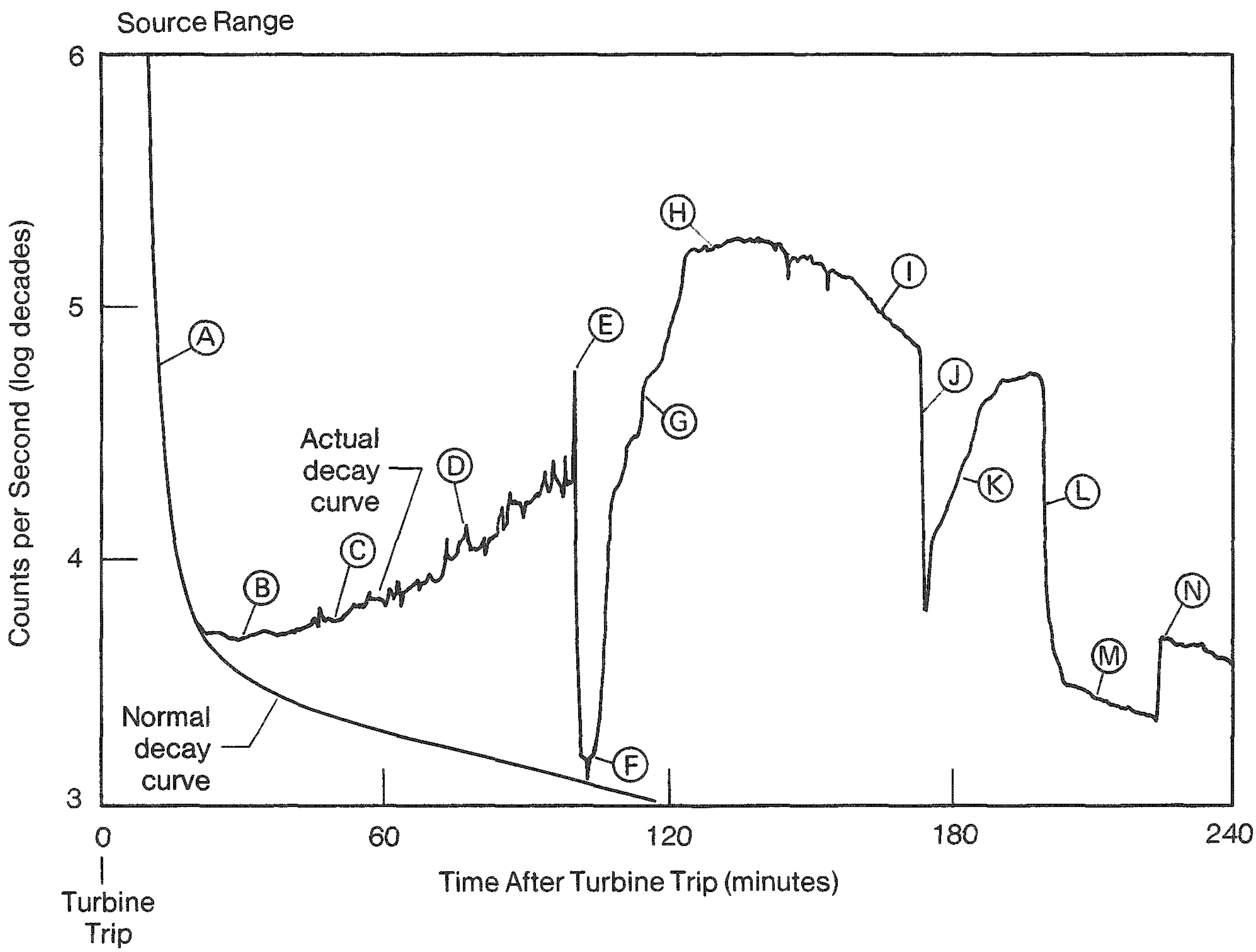
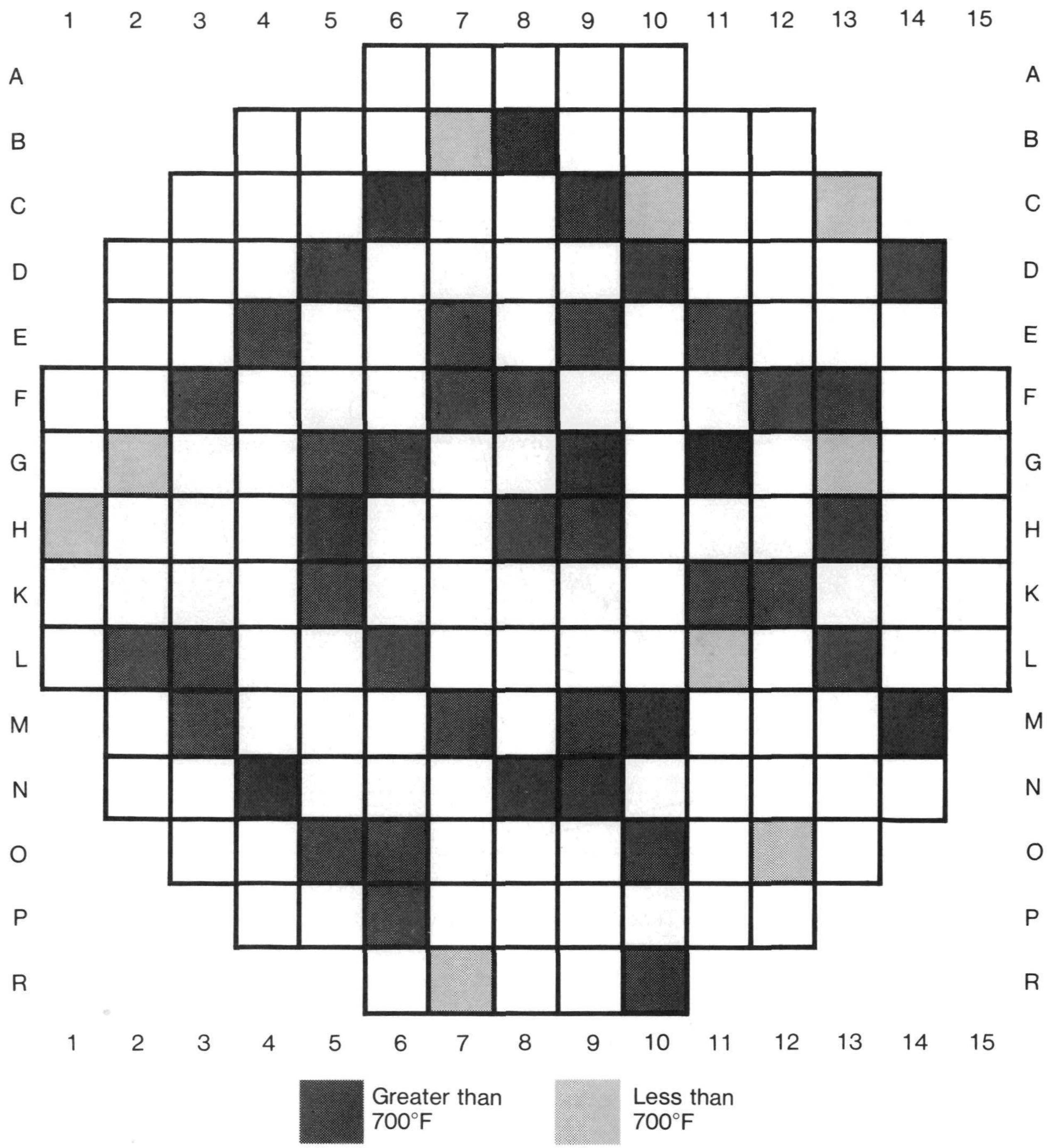


Figure CI-6. Source range neutron detector field of view versus core mixture level.



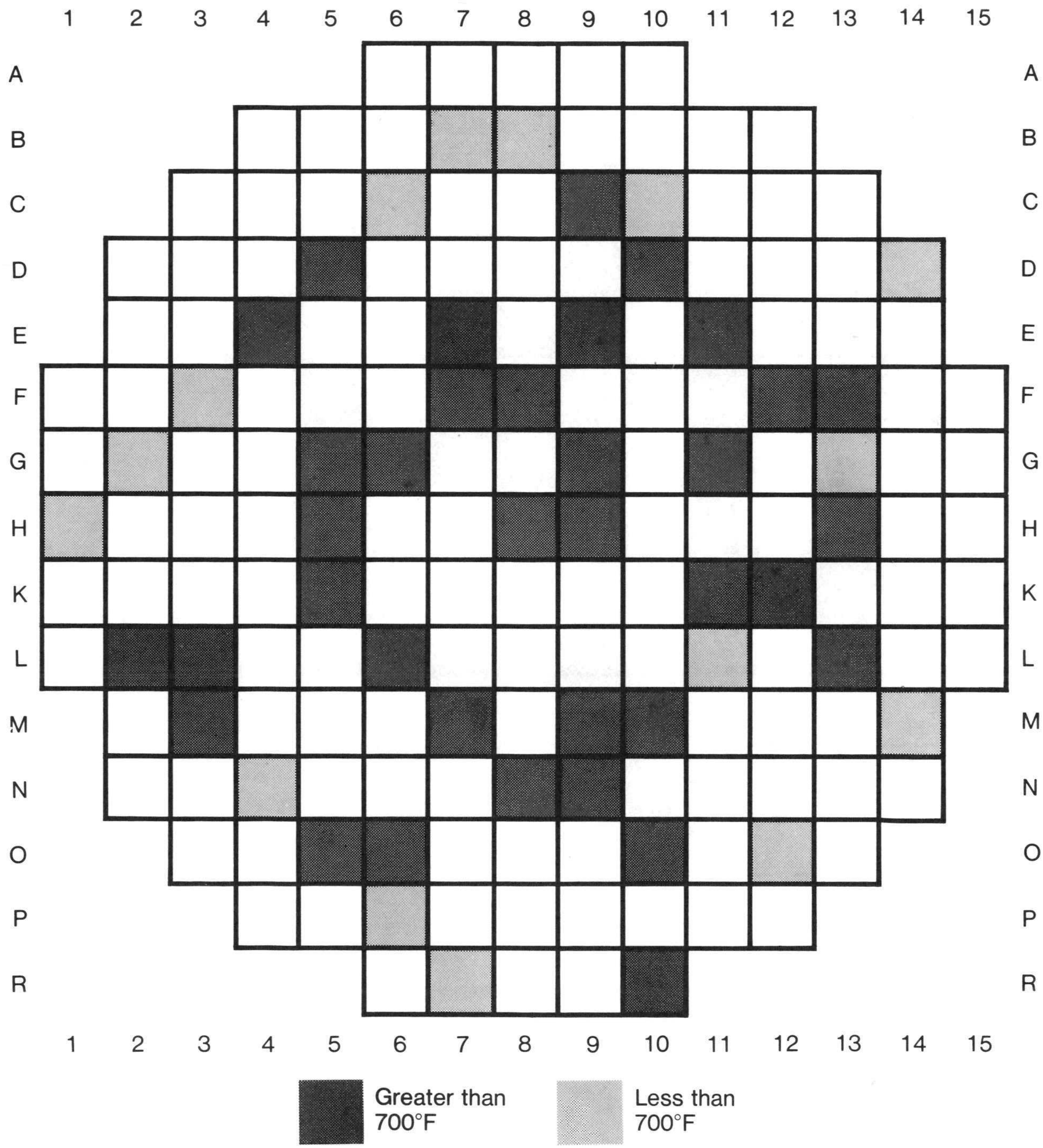
Source Range
6
5
4
3
Counts per Second (log decades)
0
60
120
180
240
Turbine Trip
Time After Turbine Trip (minutes)
A
B
C
D
E
F
G
H
I
J
K
L
M
N
Actual decay curve
Normal decay curve

Figure CI-7.



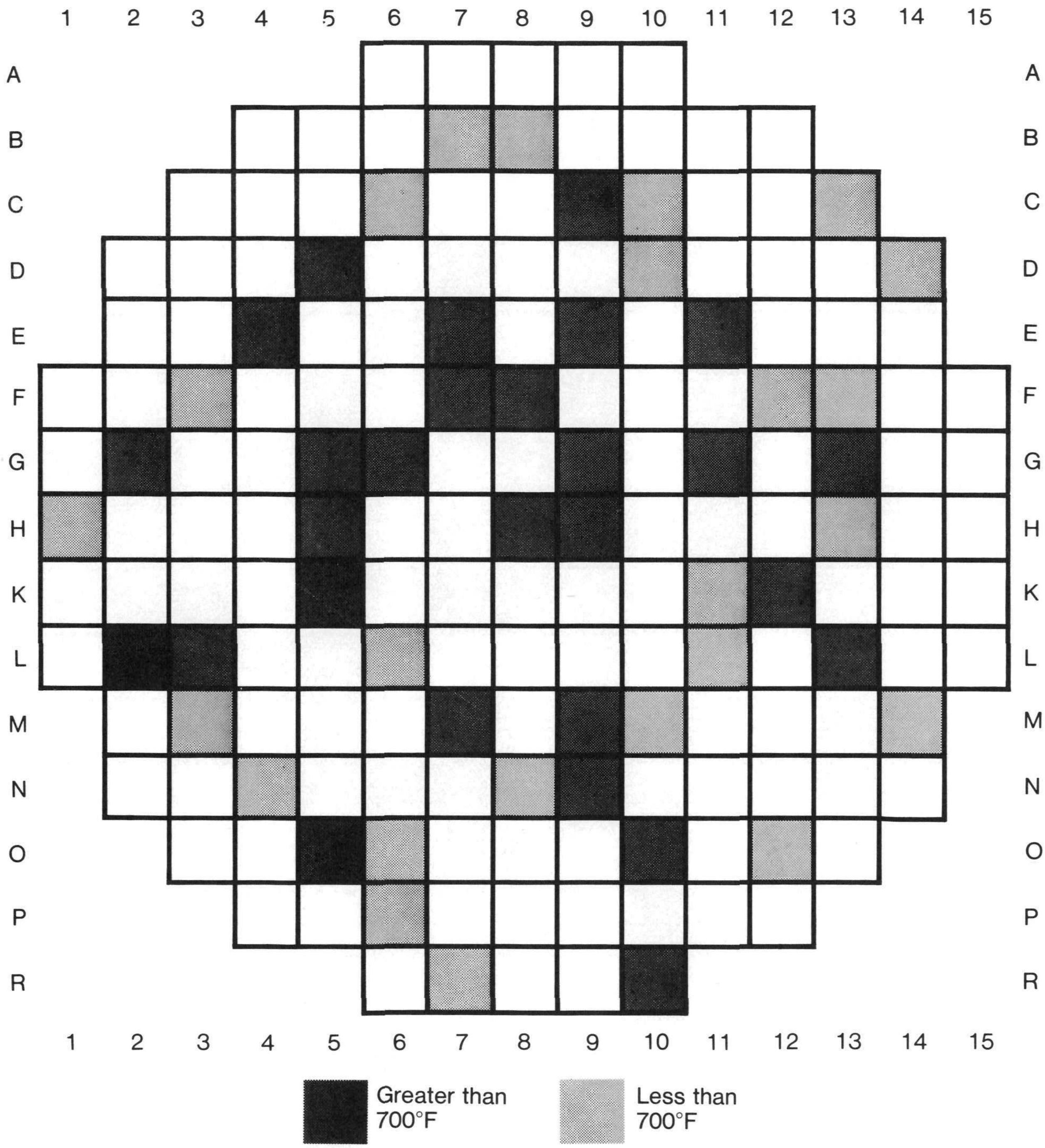
(Time: 180 minutes)

Figure CI-8.



(Time: 240 minutes)

Figure CI-9.



(Time: 300 minutes)

Figure CI-10.

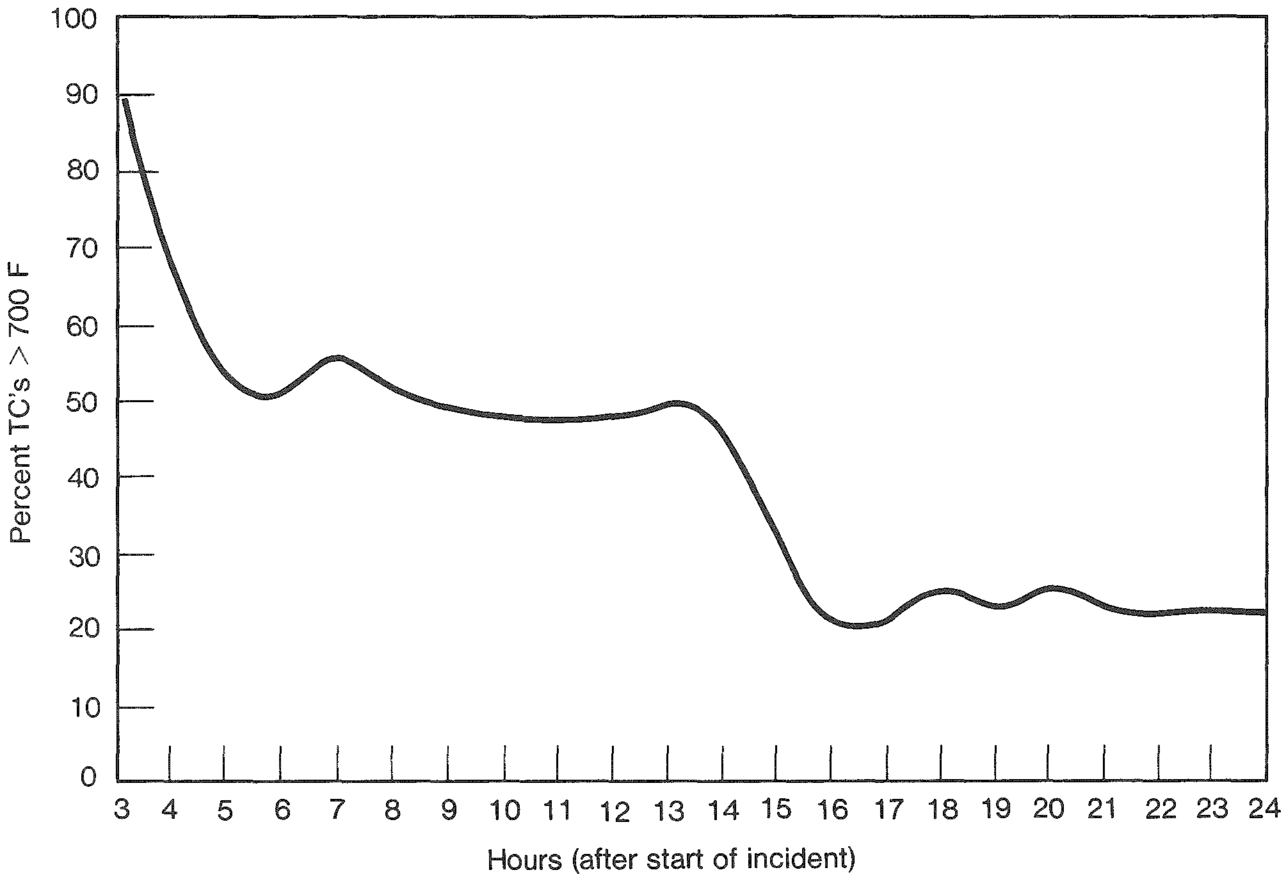


Figure CI-11. Core exit thermocouple time history.

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							281	325							
C						469			957	325					
D					1196					500				217	
E				599			1926		2580		326				
F			80				2366	2378				323	555		
G		375			2272	1774			1806		1875		234		
H	260				2452			1295	2176				1852		
K					1811						632	1760			
L		373	1566			382					296		1774		
M			325				2171		2327	348				252	
N				413				578	2167						
O					356	462				1138		308			
P						291									
R							352			475					

Figure CI-12. Map of core exit temperature (°F) 240–330 minutes.

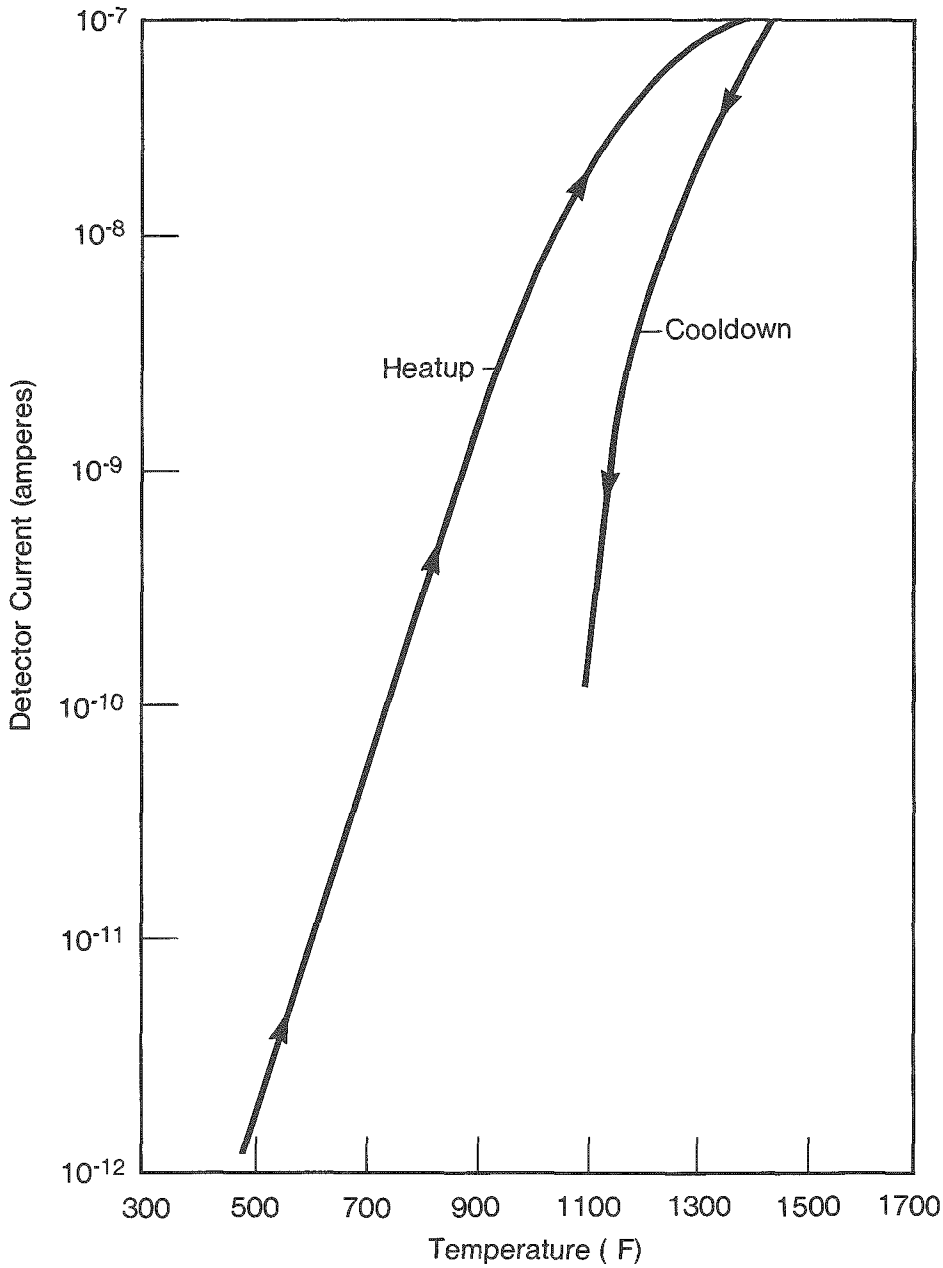


Figure CI-13. Approximate thermionic response of a self-powered neutron detector (typical).

APPENDIX ERV

ELECTROMATIC RELIEF VALVE

I. Description

The electromatic relief valve (ERV), designated RC-RV2, was manufactured by Dresser Co. of Alexandria, LA. TMI-2 uses a series 31533VX-30 valve, one of several models of flanged, pilot-operated steam relief valves with solenoid actuators. The ERV has a nominal 2-1/2 in. ID inlet with a 2500 psig ANSI standard flange and a 4" ID outlet with a 600 psig ANSI standard flange. At the set pressure of 2255 psig, the valve will relieve approximately 100,000 lbm/hr of saturated steam through a 1-5/32 in. orifice.

The ERV is in series with a block valve, designated RC-V2, just upstream of it. The block valve is remote operated under operator control only. It has no provision for automatic operation. Like the main valve, the block valve is always full open, or full closed.

2. Operation

The ERV is actuated (opened) when a reactor coolant system (RCS) pressure signal is sent to the normally de-energized solenoid operator. When energized, this operator opens a pilot valve, bleeding steam from the inner chamber of the relief valve. The steam pressure in the upper chamber of the relief valve is then able to force and maintain the valve open. The relief valve will remain open until the pilot valve closes. The relief valve actuation signal is removed on a decreasing RCS pressure of 2205 psig, de-energizing the solenoid operator, and allowing the pilot valve to close. When the pilot valve closes, steam pressure builds up in the inner chamber, overcomes the force of the steam pressure in the upper chamber, and closes the relief valve.

The valve may also be operated manually, at any RCS pressure, by manipulation of the mode switch (OFF-AUTO-MANUAL) in the control room. See also Appendix RCPCS "Reactor Coolant Pressure Control Systems."

3. History

The ERV at TMI-2 had incurred a previous stuck-open failure caused by a loss of power on the vital bus, which resulted in sending an erroneous open signal to the valve. The relief valve control system design was later corrected to assure that the valve would remain closed upon loss of power on this vital bus. TMI-2 has not reported any failures intrinsic to the valve or its operator.

The model 31533VX-30 relief valve has been installed on eight B&W reactor systems. Significant relief valve failures, in addition to the above described incident, are known. One happened at Oconnee-3, where boric acid crystallization from a leaking pilot valve stem apparently caused the pilot valve lever to bind and stick in the open position which then caused the relief valve to fail open. To prevent the pilot valve lever binding problem, Dresser has added an oilite bushing to the lever hinge to prevent excessive pilot valve leakage that could eventually cause the relief valve to open. Another happened at Rancho Seco, where excessive main seat and pilot valve leakage has required them to operate with the block valve closed. Relief valve leakage is not an uncommon problem.

Overall industry experience with the same model (31533VX-30) and similar models of Dresser pilot-operated relief valves appears to be similar to that reported by the B&W reactor systems owners. Overall industry experience with other manufacturers' relief valves (Crosby, Target-Rock) is also similar. Most of the intrinsic failures are leakage-induced. Failures in various plants have been caused by improper control circuitry and by improper mounting of the discharge pipe. Total industry statistics, other than some nuclear plant reports, are not available for the Dresser valves and for similar pilot-operated valves made by Crosby, Target-Rock, and others. Dresser has delivered more than 700 of its electromatic relief valves to industry since 1971.

4. Relief Valve Discharge Pipe and Reactor Coolant Drain Tank

The ERV and both code safety valves discharge into a common 14-inch pipe, which discharges into the reactor coolant drain tank. At the exit of each valve, before its discharge pipe joins the common 14-inch pipe, a strap-on thermocouple monitors pipe temperature as a leakage detecting measure. The 14-inch pipe discharges, inside the reactor coolant drain tank, through four perforated 8-inch pipes located 4 ft. 8 in. below the minimum tank water level.

The reactor coolant drain tank is a horizontal cylinder with hemispherical heads. Its volume is approximately 920 cu. ft. It is protected by a 6 in. relief valve set to relieve at 150 psig. Should continuous pressurizer blowdown exceed the capacity of this valve, the 18 in. manway contains a rupture diaphragm designed to burst at a pressure of 195 psig. The relief valve and the rupture diaphragm discharge to the reactor building. The contents of the reactor coolant drain tank are cooled by leakage transfer pumps which pump through two coolers and then back to the tank. The cooler effluent can be diverted to the reactor coolant drain header to reduce reactor coolant drain tank level.

While it was generally assumed that the relief valve was stuck in the full open position, it was considered advisable to verify this if possible. Two analyses to this end were made.

The first calculation of the approximate flow rate through the relief valve is based on the change in water level in the reactor drain taink. There is no recording of this water level, so the water level in the tank at time 00:00:03 is not available from records. The FSAR specifies that the level must be above 6 feet and since there was a leaky valve which had been delivering some water to the tank, the level at 00:00:03 may have been as high as 7 feet. At 00:03:13 the drain tank relief valve lifted and this indicates that the tank was full at that time.

The average flow rate to fill the tank in 190 seconds from an initial level of 6 feet is 44 lbm/sec., and from 7 feet is 10.5 lbm/sec. The output from an open valve, 20 lbm/sec. falls convincingly between these values.

The second analysis is accomplished by computing a heat balance on the drain tank during the two minutes between 00:01:26 when the temperature was recorded as 85.5°F, and 00:03:26 when the temperature was 127°F. If we estimate the level at 01:26 to be 7 feet (this is consistent with an estimate of 6.5 feet at 00:00:03 and an incoming flow of 20 lbm/sec.) we can determine the heat content (enthalpy) at 01:26. Heat (and water) are added from the pressurizer at a rate to be determined. Heat (but not water) is lost to the cooler. At the end of the 120 second interval (at 03:00:26) the total enthalpy is also determinable since the temperature was measured and the mass is a function of flow rate. This heat balance analysis leads to a flow rate out of the economizer into the drain tank of 20 lbm/sec. This analysis thus like the first is consistent with the belief that the relief valve was full open during the interval.

5. Relief Valve Discharge Pipe Temperature Indications

During the 3/28/79 incident, the following conditions were recorded for the reactor coolant system (RCS), the relief valve discharge pipe and the reactor coolant drain tank (RCDT). The expected discharge pipe temperatures calculated from open relief valve isenthalpic expansion, are shown in the last column.

Time	RCS Pressure Psig (approx.)	RCS Sat. Temp (°F) (approx.)	RCDT Press. (psia) (approx.)	Measured Discharge Pipe Temp. V1A	V1B (°F)	V2	Pipe Temperature for Isen-thalpic Discharge (°F) (approx.)
00:00:00	2155	646	17	--	--	--	291
00:00:30	1920	630	23	--	203	239	236
00:24:58	1100	556	25	263	275	285	302
01:20:31	1075	553	17	211	218	283	297
02:17:53	625	490	17	--	--	229	325

Block valve closed at approximately 02:22:00

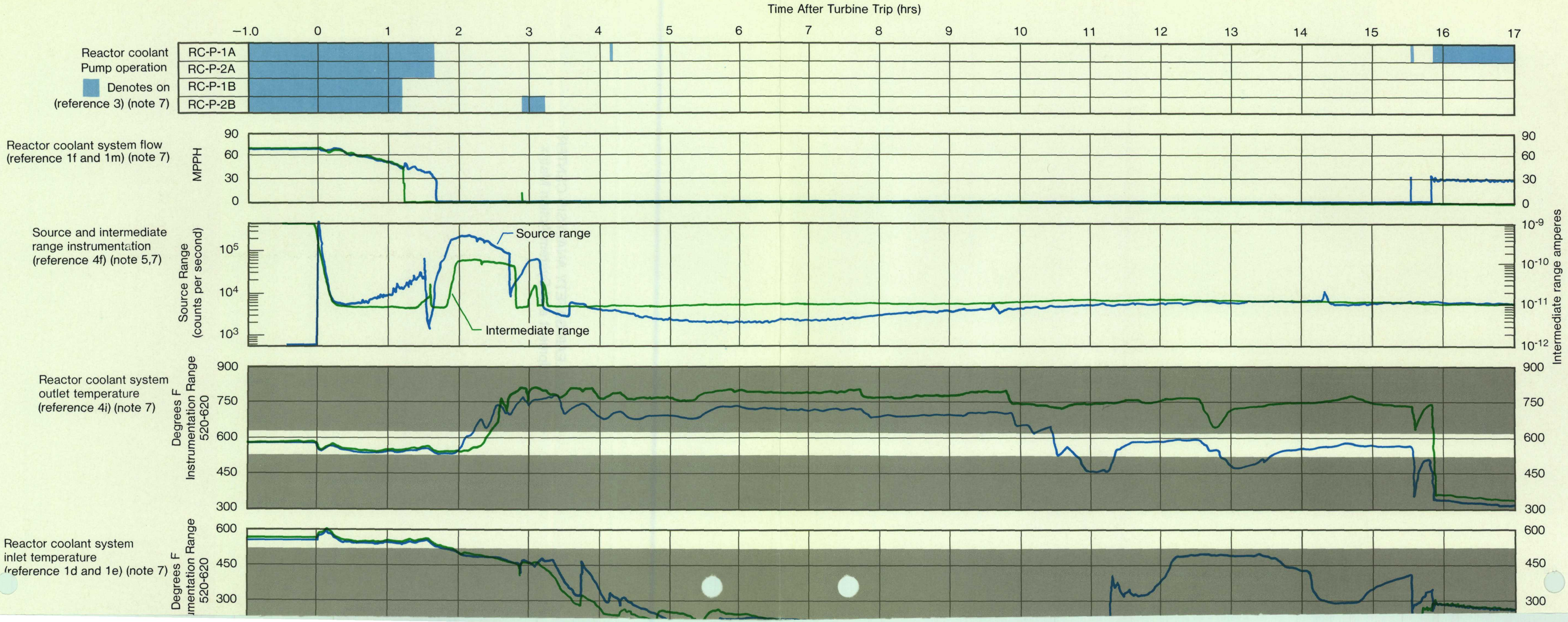
The data clearly show that the observed temperature in the discharge pipe is approximately what would be expected for an open valve. Temperatures closer to that of the pressurizer were assumed by many to be the expected indication of a stuck-open relief valve. However, isenthalpic expansion at the recorded conditions results in temperatures far below the pressurizer temperatures. The discrepancy between observed temperature and expected temperature at 02:17:53 is large enough to be significant and is not yet fully understood, but may be indicative of discharging a mixture of steam and water through the relief valve. Pressurizer levels were off scale high at about the time of this observation.

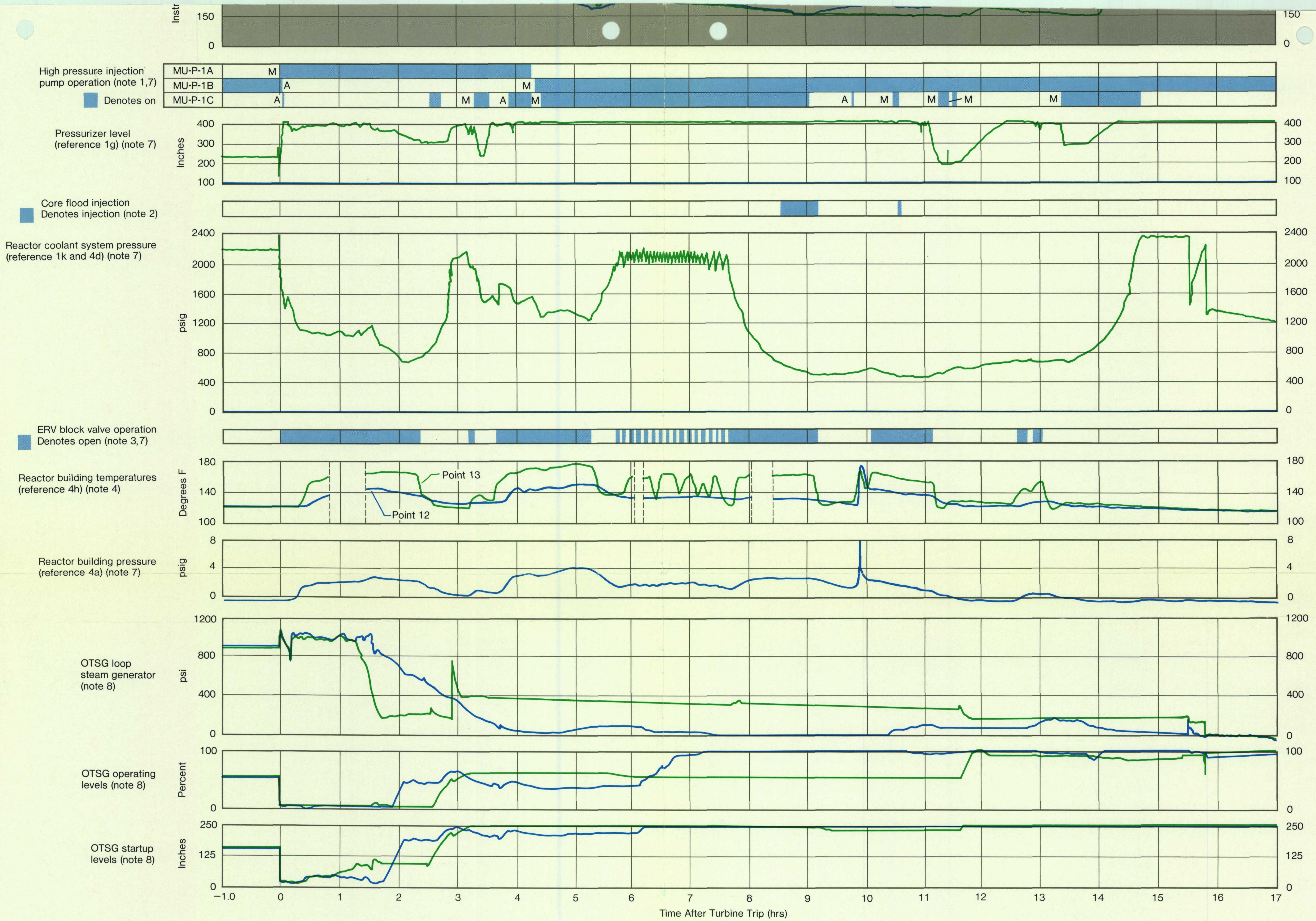
EQUIPMENT AND SYSTEM ACTION MATRIX

The Equipment and System Action Matrix is a compilation depicting significant interrelated plant parameters and equipment maneuvers pertinent to evolutions occurring during the incident at TMI-2. The arrangement was selected to best show direct relationships of equipment operation and system responses. References listed indicate the source of the input data and can be found in NSAC-1, July 1979 report. The matrix is intended to be used as a quick reference aid during review of the various parts of this report. Forthcoming expansion will display these and additional plant parameters as the operators see them. Attention will be given to how these parameters are displayed (direct or trend indication) and accessed (upon request or remote location).

NUCLEAR SAFETY ANALYSIS CENTER
Equipment and System Action Matrix

Equipment and System Action Matrix





Notes:

1. High pressure injection pump starts as shown are based on plant computer data (ref. 3) and analysis of system responses. Letter designations prior to and following each pump evolution as depicted represent the mode of control where known. M indicates manual operator action, and A indicates automatic (engineered safeguards) action. All pump trips are manual evolutions with the only exception being pump 1B at time 00:02:01.

2. Core flood injection is inferred based on the recorded reactor coolant system pressure history.

3. Darkened areas denote both ERV and ERV block valve open. ERV operation is inferred based on analysis of various plant parameters recorded during the transient.

4. This panel mounted instrumentation is located at the rear of the control room out of the direct line of sight of the operators. The reactor building temperature plot was developed from a multipoint strip chart with a sampling interval of 6 minutes. Point 12 represents tempertures at elevation 353 (six feet above the operating floor and is indicative of the highest ambient temperatures recorded within the reactor building. Point 13 represents the ambient temperatures at elevation 330 and is believed to be located in the direct path of steam discharged from the reactor coolant drain tank. The plot is discontinuous over three periods where the chart malfunctioned, overprinting continuously in one spot.

5. Source and intermediate range plots are offset approximately 6 minutes due to recording pen displacement.

6. Strip chart reference points are subject to error as chart annotation was not sufficiently precise and in some cases was accomplished a number of days following the event.

7. Information available to the operator on the main control console.

8. Refer to Appendix OTSG (once-through steam generator) for level comparison information.

Loop A

Loop B

APPENDIX ESF

ENGINEERED SAFETY FEATURES

1.0 DESIGN OBJECTIVES

The Engineered Safety Features (ESF) design objectives are to:

A. Prevent core meltdown

B. Maintain the integrity of the reactor building to ensure that the exposure of the public to radiation will be below the limits of 10 CFR 100.

The ESF systems are provided with sufficient redundancy of components and power sources to meet these objectives under the conditions of the most severe loss of coolant accident (LOCA).

1.1 EMERGENCY CORE COOLING SYSTEM

Core cooling is provided through the use of the Emergency Core Cooling Systems described below and in Figure ESF-1.

1.1.1 Core Flooding System

The description of the Core Flood System is included in Appendix CF.

1.1.2 High Pressure Injection System (HPI)

The description of the High Pressure Injection System is included in Appendix HPI.

1.1.3 Low Pressure Injection

The decay heat removal system serves an engineered safety feature function by providing "Low Pressure Injection" of borated water into the reactor core during emergency conditions, and to accomplish long term core cooling during post accident conditions, by recirculation of water to the reactor vessel from the reactor building sump. Borated water from the borated water storage tank (BWST) and sodium hydroxide from the sodium hydroxide tank are injected by the decay heat removal pumps into the reactor vessel after the reactor coolant pressure has fallen below the maximum discharge pressure capability of the pumps (approximately 180 psig). When the BWST level has been reduced to its minimum level, the suction to the decay heat removal pumps is aligned by automatic valve actuation to recirculate the water in the reactor building sump back to the reactor.

The decay heat removal system is comprised of two parallel and independent circuits, either of which will satisfy the low pressure injection requirements imposed by a LOCA. Each of the two decay heat removal pumps discharges through the tube side of its associated circuit cooler and into the reactor vessel through the two independent core flooding nozzles. During recirculation of the reactor coolant from the reactor building sump to the reactor vessel, the coolant temperature is reduced by transferring the decay heat of the reactor core to the atmosphere via the decay heat closed cooling water and the nuclear services river water systems.

1.2 REACTOR BUILDING INTEGRITY AND RADIATION EXPOSURE PROTECTION

The design and use of the reactor building structure, its isolation system, its heat removal systems, and a combustible gas control system assure reactor building integrity and low radiation exposure to the public.

1.2.1 Reactor Building

The Reactor Building is a cylindrical reinforced concrete structure resting on a flat foundation mat. The cylindrical walls and the domed roof are lined with a carbon steel liner. The structure provides biological shielding under normal and accident conditions. The steel liner encloses the reactor and the reactor coolant systems and ensures that an acceptable upper limit of leakage of radioactive material will not be exceeded under LOCA conditions.

1.2.2 Reactor Building Isolation Valves

With the exception of the reactor building sump piping, isolation valves are used to provide a double barrier so that no single, credible failure or malfunction of an active component can result in leakage from the reactor building. Where piping systems containing radioactive fluids are located both inside and outside the reactor building, isolation valves are located on each side of the reactor building penetration. In general, Train A valves are outside the reactor building and Train B valves are inside the reactor building. The reactor building sump piping uses a jacketed pipe and valve design in lieu of an isolation valve inside the reactor building.

1.2.3 The Reactor Building Spray System

The reactor building spray system serves as an engineered safety feature system and has no normal operating function. The reactor building spray system consists of two pumps, two spray headers, and one sodium hydroxide tank. The system also shares the BWST with the decay heat removal system in taking suction from the tank. The suction for the spray pumps will automatically transfer to the building sump when the tanks are at a minimum level. This system will furnish 100 percent of the design cooling capacity required during a LOCA with both of the spray flow paths in operation. Both of the spray paths operate independently.

In the event of a loss-of-coolant accident (LOCA) with the resulting pressure increase in the reactor building, sodium hydroxide and borated water are automatically combined in the vicinity of the tanks and supplied to the spray pumps. This alkaline solution of borated water is sprayed into the reactor building atmosphere through spray nozzles which are located at the top of the building. This process reduces the temperature and consequently the pressure of the post-LOCA reactor building atmosphere. The addition of boric acid insures an adequate shutdown margin. The sodium hydroxide is added to the solution to reduce the airborne concentration of radioactive iodine. The result of reducing pressure and airborne radioactive iodine concentration is a reduction in the potential leakage of radioactivity to the site and surrounding area.

The reactor building spray system was originally designed to spray a mixture of borated water, sodium hydroxide, and sodium thiosulfate. Both sodium hydroxide and sodium thiosulfate were considered desirable for removal of airborne radioactive iodine. In November 1976, to provide satisfactory chemical performance, the system was modified to operate as a sodium hydroxide spray system only, requiring a mixture of borated water and sodium hydroxide. As a result of this change, the sodium thiosulfate tank and the building spray system piping and valves between the sodium thiosulfate tank and spray pump are no longer used.

1.2.4 Reactor Building Air Cooling System

The reactor building air cooling system consists of five units connected to a common system of ductwork for air distribution. Each cooling unit consists of finned, water type cooling coils and an electric motor driven axial flow fan. The five units are assembled in a common metal housing with suitable divider plates and back draft dampers to permit shutdown of selected fans without affecting the airflow from the others.

For normal operation, cooling water is supplied to the cooling coils from the evaporative coolers with four of the five air cooling unit fans in operation. For LOCA conditions, at least two fans per train are operated at fastspeed, and the cooling water supply is drawn from the nuclear services river water system. This LOCA cooling water flow is provided by two of the four reactor building emergency cooling booster pumps. These pumps take suction from either of the redundant nuclear services river water circuits and discharge through the cooling coils to the river.

The containment LOCA heat removal requirement will be satisfied by the operation of (1) both reactor building spray loops, or (2) one reactor building spray loop and two of the five reactor building air cooler units with fans at fast speed.

1.2.5 Combustible Gas Control System

Following a LOCA, the containment atmosphere will be continuously monitored to determine hydrogen concentration. If the hydrogen concentration becomes excessive, approximately 10 days following a design basis LOCA, a thermal recombiner system, located outside of the containment will be placed into service by the operator. The reactor building atmosphere is drawn through either of the two reactor building purge outlet penetrations, to the suction of the recombiner fan. The gases discharged from the recombiner are then returned to the reactor building through either of the two reactor building purge inlet penetrations.

A backup means of hydrogen control is available by purging the reactor building atmosphere to the environment. The reactor building atmosphere can be passed through a filter unit consisting of a prefilter, a HEPA filter, an activated charcoal filter and a second HEPA filter before being discharged by means of a blower to the unit vent. To replace the gases exhausted from the reactor building, a throttling valve is provided to admit a controlled amount of outside air to the building.

2.0 ENGINEERED SAFETY FEATURES ACTUATION

The ESF systems are actuated by low reactor coolant system pressure or high reactor building pressure, as follows:

ENGINEERED SAFETY FEATURES ACTUATION CONDITIONS

Action	Trip Condition	Normal Steady State Range	Trip Setpoint psig
Start Emergency Core Injection	Low Reactor Coolant Pressure or	2120-2250 psig.	1,600
	High Reactor Building Pressure	Atmospheric	
Isolate Reactor Building, Start Reactor Building Emergency Cooling, Open Reactor Building Spray Valves	High Reactor Building pressure	Atmospheric	4
Start Reactor Building Spray Pumps	High-High Reactor Building Pressure	Atmospheric	30

A 2-out-of-3 trip logic is used for each pressure sensed.

3.0 PRESSURE SENSING DEVICES

3.1 REACTOR COOLANT PRESSURE SENSING

Reactor coolant pressure is sensed by three separate pressure transmitters with a sensing range of 0-2500 psig. Two of the three pressure transmitters are located on reactor coolant loop A and one is lcoated on reactor coolant loop B.

3.2 FUNCTIONS OF THE REACTOR COOLANT PRESSURE SIGNAL

Each transmitter feeds a separate engineered safety features instrumentation cabinet providing the following contact outputs:

- When the reactor coolant system pressure decreased to ≤1640 psig the instrumentation output supplies an actuation signal to the ESF actuation relay cabinets and a channel status signal to the computer. The computer also monitors an auxiliary relay informing the operator that the ESF relays received the signal from the pressure instrumentation.

- When reactor coolant system pressure increases to ≥1845 psig, the ESF actuation relay logic will automatically reset and an auxiliary relay provides a status signal to the computer.

One A loop reactor coolant pressure transmitter provides a signal to the non-nuclear instrumentation which provides the following:

- Control room recording of wide range reactor coolant system pressure (0-2500 psig) on a strip chart recorder.

- An alarm signal if the core flood tank isolation valve is in an abnormal position.

Each pressure transmitter output signal is sent to the computer for wide range reactor coolant system pressure monitoring.

One A loop and one B loop reactor coolant system pressure transmitter signal is used to close the decay heat removal system isolation valves, DH-V1 or DH-V-171 and DH-V2, whenever the reactor coolant system pressure is greater than 320 psig. The first two valves are in parallel; only one may be selected to operate while the other is closed. Therefore, only the selected valve will get a close signal. There is control room panel status indication of each reactor coolant system pressure.

3.3 REACTOR BUILDING PRESSURE SENSING

Reactor building pressure (as monitored for reactor building isolation and cooling) is monitored by six pressure switches with a range of 0-10 psig actually set to actuate at 3.58 psig. Functions of the reactor building isolation and cooling pressure switches are as follows:

- The control status output of each switch is sent through separate channels 1 through 3, to each of the two separate and independent ESF actuation logic trains, A and B.

- An auxiliary relay contact status is monitored off the reactor building isolation and cooling pressure switches for output to the computer.

- There is control room panel status indication for each channel of each reactor building isolation and cooling actuation.

Reactor building high-high pressure (as monitored for reactor building spray) is monitored by six pressure switches with a range of 0-100 psig actually set to actuate at 28 psig. The functions of the reactor building spray pressure switches are as follows:

- The control status output of each switch is sent through separate channels, 1 through 3, to each of two separate and independent reactor building spray activation trains, A and B.

- An auxiliary relay contact status is monitored off the reactor building spray pressure switches for output to the computer.

There is control room panel status indication for each channel of each reactor building spray actuation system. There are three reactor building pressure sensing penetrations. Each of the penetrations is monitored by a pair of reactor building isolation and cooling switches and a pair of reactor building spray switches.

3.4 ENGINEERED SAFETY SYSTEM REACTOR BUILDING PRESSURE INDICATION

The reactor building is equipped with two pressure transmitters, one with a range of 0-100 psig and the other with a range of -5 to 10 psig. The signals from these transmitters are displayed on redundant dual indicating recorders, thus providing analog trend data of reactor building pressure.

4.0 AUTOMATIC ACTUATION LOGIC

4.1 SAFETY INJECTION

There are two actuation trains, A and B, for safety injection. Initiation of an actuation train requires that 2 out of 3 reactor coolant system pressure instruments, set to trip no lower than 1640 psig, be tripped. When this occurs, the components listed on Table ESF-1 will automatically go to their engineered safety features position, thus providing high pressure injection, low pressure injection, and the support systems for these modes. The tables show three specific groups of equipment associated with each safety injection actuation train. The three groups in each train are designed to facilitate on-line testing.

There is control room panel status indication of each actuation group. When two out of three channel trip logic is satisfied, the actuation group lights will indicate tripped in addition to the individual component lights indicating their ESF status.

The computer monitors the contacts from the actuation groups and alarms when the two out of three logic has been satisfied.

4.2 REACTOR BUILDING ISOLATION AND COOLING

There are two actuation trains, A and B, for reactor building isolation and cooling. To initiate an actuation train, two out of three reactor building isolation and cooling pressure switches must be tripped, signifying high reactor building pressure. Because actuation of reactor building isolation and cooling also actuates safety injection, the components listed in Tables ESF-1 and ESF-2 will assume their ESF position, since actuation of reactor building isolation and cooling actuates safety injection. This results in reactor building isolation via closure of automatic valves, reactor building cooling via start of reactor building emergency booster pumps and associated units, start of high pressure injection pumps and opening of high pressure injection valves, start of decay heat removal pumps, opening of reactor spray valves and actuation of the required support systems. The tables show three specific groups associated with each reactor building isolation actuation; this is done to facilitate on-line testing.

4.3 REACTOR BUILDING SPRAY

There are two actuation trains for reactor building spray, A and B. For initiation of an acuation train to occur, 2 out of 3 reactor building spray pressure switches must have tripped, signifying high-high reactor building pressure. Actuation of reactor building isolation and cooling must precede actuation of reactor building spray so that the pump discharge valves will be opened. Actuation of reactor building spray starts the reactor building spray pumps, BS-P-1A and BS-P-1B, respectively associated with actuation trains A and B, and provides reactor building spray flow. There are control room panel status indications of each reactor building spray actuation pressure switch, two out of three channel logic for initiation of reactor building spray pumps and reactor building spray pump status indicating lights.

The computer monitors the contacts from each pressure switch and the two out of three actuation logic matrix. It will alarm when a pressure switch trip as well as the two out of three logic is satisfied.

5.0 SPECIAL OPERATIONS OF THE ENGINEERED SAFETY FEATURES SYSTEM

5.1 MANUAL INITIATION

There are means by which the operator may depress pushbuttons on the control room console to actuate the ESF system. Two pushbuttons, one per actuation train, allow the operator to initiate safety injection only for that actuation train, are available to initiate reactor building isolation and cooling, and safety injection for that actuation train. Reactor building spray pumps are started by operating individual hand switches.

There is control room panel status indication of each group of components to verify that all three groups actuated.

The computer monitors the contacts of one manual initiation relay per group and alarms when initiated. When the system has been manually initiated the computer alarms that it is in the TEST mode.

5.2 SAFETY INJECTION BYPASS

This pushbutton allows the operator to bypass the safety injection channels when the reactor coolant system pressure is less than 1820 psig. This also allows the operator to take manual control of the pumps and valves after safety injection has been initiated. The bypass function is automatically reset when reactor coolant system pressure exceeds 1845 psig, if it has not been manually reset by that time.

At the control room console there is an indication as to when the safety injection channel can be bypassed, when it is bypassed, when it can be reset, and when it has been reset automatically.

The computer monitors the bypass signals for each channel and will alarm when the channel has been "bypassed" and when the "bypass" has been reset.

5.3 REACTOR BUILDING ISOLATION AND COOLING PUSHBUTTON

This pushbutton allows the operator to defeat, or bypass, a reactor building isolation and cooling channel, when that particular channel and one other channel have tripped. This also allows the operator to take manual control of the affected valves and pumps after the reactor building has been isolated. Upon reactor building pressure decreasing to or below 3.58 psig the reactor building isolation and cooling system must be manually reset if it is desired that the system trip automatically again upon reactor building pressure exceeding 3.58 psig.

At the control room console there is indication as to when the reactor building isolation and cooling channel may be defeated, if it is defeated, and if it has been reset. The computer monitors the defeat relay contacts for each channel and will alarm when the channel has been defeated and when the defeat has been reset.

The term defeat, as it pertains to reactor building isolation and cooling, is synonymous with the term bypass, as it pertains to high-pressure injection.

TABLE ESF-1

SAFETY INJECTION ACTUATION

ACTUATION A

SAFETY INJECTION GROUP 1

Equipment		ESF Position
DFX-1A		ON
DC-P-1A		ON
*G2-12	(Note 1)	CLOSED
MU-P-1A		ON
NR-P-1A	(Note 3)	ON
NS-P-1A	(Note 4)	ON
*T1E-2E2		OPEN
*T3E-4E2		OPEN
*T11E-21E2		OPEN
DC-V96A		CLOSED

SAFETY INJECTION GROUP 2

Equipment		ESF Position
NR-V9A		CLOSED
DH-VRA		OPEN
DH-V5A		OPEN
DH-V8A		OPEN
DH-V100A		CLOSED
DH-V102A		OPEN
MU-P-1B	(Note 2)	ON
NR-V40A		OPEN
NS-P-1C	(Note 4)	ON
*T12-22E-2		OPEN
MU-V28		CLOSED

ACTUATION B

SAFETY INJECTION GROUP 1

Equipment		ESF Position
DFX-1B		ON
DC-P-1B		ON
*G22-12	(Note 1)	CLOSED
MU-P-1B	(Note 2)	ON
NR-P-1C	(Note 3)	ON
NS-P-1	(Note 4)	ON
*T2E-1E2		OPEN
*T4E-3E2		OPEN
*T21E-11E2		OPEN
DC-V96B		CLOSED

SAFETY INJECTION GROUP 2

Equipment		ESF Position
NR-V9B		CLOSED
DH-V4B		OPEN
DH-V5B		OPEN
DH-V8B		OPEN
DH-V100B		CLOSED
DH-V102B		OPEN
MU-P-1C		ON
NR-V40B		OPEN
NS-P-1C	(Note 4)	ON
*T22E-12E-2		OPEN
HY-V55		CLOSED

*See TABLE ESF-3.

TABLE ESF-1 (continued)

ACTUATION A			ACTUATION B		
SAFETY INJECTION GROUP 3			SAFETY INJECTION GROUP 3		
Equipment		ESF Position	Equipment		ESF Position
MU-V36		CLOSED	MU-V37		CLOSED
MU-V16A		OPEN	MU-V16C		OPEN
MU-V16B		OPEN	MU-V16D		OPEN
DH-P-1A		ON	DH-P-1B		ON
NR-P-1B	(Note 3)	ON	NR-P-1C	(Note 3)	ON
*T31E-41E		OPEN	*T41E-31E-2		OPEN
NS-V84B	(Note 5)	CLOSED	NS-V84A	(Note 5)	CLOSED
NS-V42A		OPEN	NR-V42B		OPEN
			NS-V32		CLOSED
			NS-V67		CLOSED

NOTES

NOTE 1: Diesel generator breaker will be closed only if normal power is lost.

NOTE 2: Normally MU-P-1A and MU-P-1C will be on; MU-P-1B will start only if the pump that is selected for backup (MU-P-1A or MU-P-1C) fails to start or is inoperable.

NOTE 3: The NR pump in each header selected for ESF or standby will start if a pump is not operating in that header; otherwise the operating pump will remain in service.

NOTE 4: Normally NS-P-1A and 1B will start; however, NS-P-1C will start if either NS-P-1A or B (depending upon which pump it is selected to backup) fails to start or is inoperable.

NOTE 5: Valves NS-V84A and NS-V84B will normally remain open after an ESF actuation. However, if the heat sink associated with a valve is lost that valve will close.

*See Table ESF-3.

TABLE ESF-2

REACTOR BUILDING ISOLATION AND COOLING ACTUATION

ACTUATION A R.B. ISOLATION AND COOLING GP. 1 Equipment	 ESF Position	ACTUATION B R.B. ISOLATION AND COOLING GP. 1 Equipment	 ESF Position
DH-V8A	OPEN	DH-V8B	OPEN
AH-V81	CLOSED	AH-E-11D	ON
AH-V101	CLOSED	AH-V80	CLOSED
AH-V102	CLOSED	AH-V103	CLOSED
AH-V105	CLOSED	AH-V104	CLOSED
AH-V107	CLOSED	AH-V106	CLOSED
AH-V144	CLOSED	AH-V108	CLOSED
AH-V3	CLOSED	CF-V115	CLOSED
NM-V52	CLOSED	DH-V2	CLOSED
NR-V51A	CLOSED	NM-V104	CLOSED
RR-V2A	OPEN	NR-V51B	CLOSED
RR-V2B	OPEN		
RR-V5A	OPEN	RR-V2C	OPEN
RR-V5B	OPEN	RR-V2D	OPEN
SV-V5C	OPEN	RR-V6C	OPEN
SV-V55	CLOSED	RR-V6D	OPEN
WDL-V1095	CLOSED	RR-V6E	OPEN
DC-V114	CLOSED	SV-V54	CLOSED
		WDL-V1092	CLOSED
		DC-V103	CLOSED
		DC-V115	CLOSED

TABLE ESF-2 (continued)

ACTUATION A R.B. ISOLATION AND COOLING GP. 2 Equipment	ESF Position
AH-E-4A	ON
AH-E-11A	ON
RR-V25C	CLOSED
BS-V1A	OPEN
CA-V10	CLOSED
CA-V4A	CLOSED
CA-V9	CLOSED
RR-P-1B	ON
WDG-V199	CLOSED
WDL-V1126	CLOSED
WDL-V22	CLOSED
**AH-D4092A & D	CLOSED
**AH-D4092 B, C, & E	OPEN
**ED-4098	CLOSED
AH-V125A	VENT

ACTUATION B R.B. ISOLATION AND COOLING GP. 2 Equipment	ESF Position
AH-E-4B	ON
AH-E-11C	ON
RR-V25C	CLOSED
BS-V1B	OPEN
CA-V1	CLOSED
CA-V3	CLOSED
CA-V4B	CLOSED
CA-V8	CLOSED
CA-V6	CLOSED
RR-P-1D	ON
WDG-V2	CLOSED
WDL-V1125	CLOSED
2DL-V271	CLOSED
**AH-D4092A & D	CLOSED
**AH-D4092B, C, & E	OPEN
**ED-4098	CLOSED
AH-V125B	

**See Table ESF-3.

TABLE ESF-2 (continued)

ACTUATION A

R.B. ISOLATION AND COOLING GP. 3

Equipment	ESF Position
IC-V2	CLOSED
IC-V5	CLOSED
MU-V2A	CLOSED
MU-V2B	CLOSED
MU-V377	CLOSED
NS-V72	CLOSED
NS-V81	CLOSED
NS-V83A & V83B	OPEN
NS-V215 & V216	CLOSED
NR-P-2A	ON
NR-V144A	OPEN
RR-P-1A	ON
RR-V25A	CLOSED
RR-V25B	CLOSED
IC-P-1A	OFF
AH-C-8A	ON
AH-3-11B	ON
AH-E-11C	ON
AH-P-1A	ON
AH-V1A	CLOSED
AH-V1B	CLOSED
AH-V4A	CLOSED
AH-V4B	CLOSED
AH-V5	CLOSED
AH-V60	CLOSED
AH-V62	CLOSED
AH-V72	CLOSED
AH-V74	CLOSED
AH-E-12A & B	OFF
AH-E-19A & 19B	OFF

ACTUATION B

R.B. ISOLATION AND COOLING GP. 3

Equipment	ESF Position
IC-V3	CLOSED
IC-V4	CLOSED
MU-V376	CLOSED
MU-V18	CLOSED
MU-V25	CLOSED
NS-V100	CLOSED
NR-P-2B	ON
NR-V-144B	OPEN
RR-P-1C	ON
RR-V25D	CLOSED
RR-V25E	CLOSED
AH-C-8B	ON
AH-E-11E	ON
AH-P-1B	ON
AH-V2A	CLOSED
AH-V2B	CLOSED
AH-V3A	CLOSED
AH-V3B	CLOSED
AH-V6	CLOSED
AH-V61	CLOSED
NS-V83A & B	OPEN
NS-V215 & V216	CLOSED
AH-V63	CLOSED
AH-V71	CLOSED
IC-P-1B	OFF
AH-E-12A & 12B	OFF
AH-E-19A & 19B	OFF

TABLE ESF-3

DESIGNATOR KEY

1. First Character Designator

SYSTEM PREFIXES

AH	Air Handling Equipment	HP	Health Physics
AM	Ammonia	HV	Heater Vents
AS	Start-up, Auxiliary Steam	HY	Hydrogen CO_2
BS	Reactor Building Spray	IA	Instrument Air
CA	Chemical Addition/Sampling	IC	Intermediate Closed Cooling Water
CF	Core Flooding	LO	Lube Oil-Turbine
CL	Chlorine System	LR	Leak Base Test
CO	Condensate	MO	Moisture Separator
CW	Circulating Water	MS	Main Steam
DC	Decay Heat-Closed Cooling Water	MU	Make Up & Purification
DF	Diesel Fuel Starting Air Lube Oil Jacket Cooling and Exhaust	NB	Nitrogen Blanketing
		NM	Nuclear Plant Nitrogen Manifold
DH	Decay Heat Removal	NR	Nuclear Services - River Water
DO	Domestic Water		
DW	Demineralized Water	NS	Nuclear Services - Closed Cooling Water
EF	Emergency Feedwater		
ES	Engineered Safety Feature	PC	Penetration Cooling
EX	Bleed Steam	PP	Penetration Pressurization
FH	Fuel Handling	RB	Reactor Building Normal Cooling
CRD	Control Rod Drive		
EB	Environmental Barrier	RC	Reactor Coolant
HR	Hydrogen Recominder	RD	Rod Drive Mechanisms
FS	Fire Service	RO	Reactor Coolant Pump Motor Oil Drains
FW	Feed Water		
GS	Glad Steam	RR	Reactor Bldg. Emergency Cooling River Water
HD	Heater Drains		

SA	Station Service Air	SS	Secondary Plant Sampling System
SC	Secondary Services - Closed Cooling	SV	Steam Generator Secondary Side Vents & Drains
SD	Sump Drains	SW	Screen Wash Water (River Water Pump House)
SE	Sewer	VA	Condenser Air Extraction
SF	Spent Fuel Cooling	SGC	OTSG-Chemical Cleaning System
SN	Sampling Nuclear Systems	WDL	Liquid Radwaste
SO	Seal Oil System - Generator	WDG	Gaseous Radwaste
SR	Secondary Services River Water		

2. Second Character Designator

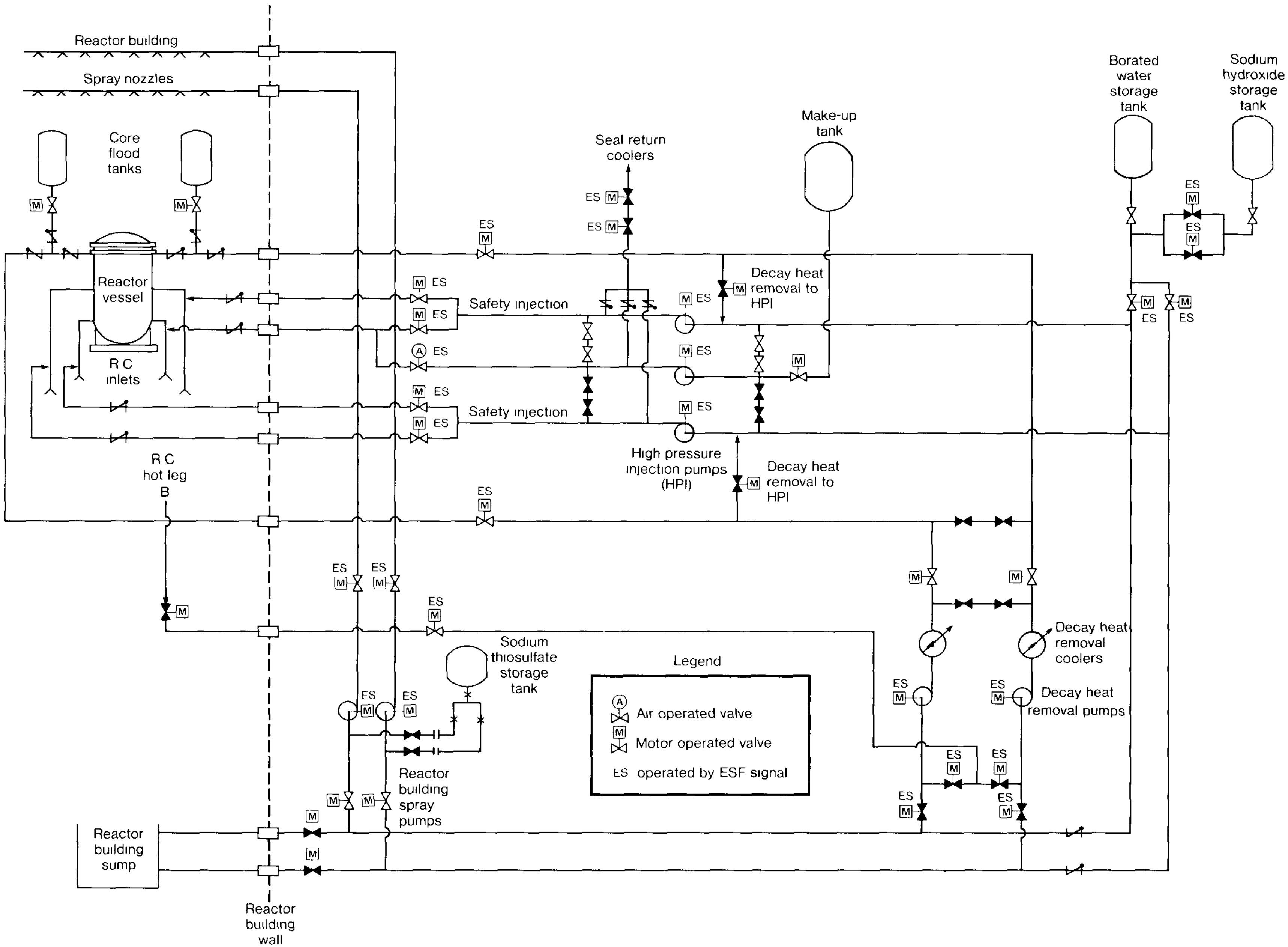
P - Pump
V - Valve
E - Electrical Motor (e.g., Fan Motor)
C - Chiller Unit

3. Special Notations

 * - Electrical Breaker designator
** - Heating and Ventilation damper designator

<u>Example:</u>

DH-P-1A   Decay Heat Removal Pump 1A



Reactor building
Spray nozzles
Core flood tanks
Reactor vessel
R C inlets
R C hot leg B
Seal return coolers
Make-up tank
Borated water storage tank
Sodium hydroxide storage tank
Safety injection
Safety injection
Decay heat removal to HPI
High pressure injection pumps (HPI)
Decay heat removal to HPI
Decay heat removal coolers
Decay heat removal pumps
Sodium thiosulfate storage tank
Reactor building spray pumps
Reactor building sump
Reactor building wall
Legend
Air operated valve
Motor operated valve
ES operated by ESF signal
ES
M
A

Figure ESF 1.

APPENDIX HPI

HIGH PRESSURE INJECTION, MAKEUP, AND PURIFICATION SYSTEM

1. FUNCTIONS

The high pressure injection makeup, and purification system is designed to provide the reactor coolant system with the following functions:

- o Provide makeup to the reactor coolant system for normal inventory control.
- o Provide injection water to the reactor coolant pump seals.
- o Remove corrosion and fission products from the reactor coolant system during purification operations.
- o Control the boron concentration of the reactor coolant.
- o In conjunction with the pressurizer, accommodate temporary changes in reactor coolant volume due to small temperature changes.
- o Maintain proper concentration of hydrogen and corrosion inhibiting chemicals in the reactor coolant.
- o Supply borated makeup water to the core flood tanks. (Refer to Appendix CF)
- o Provide makeup to the reactor coolant system for leakage and small breaks in the reactor coolant pressure boundary.
- o Provide high pressure emergency core cooling upon safety features actuation.

The high pressure injection makeup, and purification system is designed to control and maintain the reactor coolant inventory and to control the boron concentration of the reactor coolant system through the processes of makeup and letdown of the reactor coolant. The makeup portion provides normal makeup to the reactor coolant system through the reactor coolant pump seals and via the normal makeup line. (See Figure HPI-1).

The purification portion removes corrosion and fission products by purification of the letdown fluid. Two letdown coolers are provided to remove heat from the reactor coolant prior to its entering the purification demineralizers and filters. A letdown flow rate of 45 gpm will permit recirculation of approximately one reactor coolant system volume through the purification train during a 24-hour period. The maximum letdown flow rate is about 140 gpm. This flow rate permits changing the boron concentration during xenon peaking by bleeding borated coolant from the reactor coolant system and adding non-borated reactor grade water. This is done to compensate for the negative reactivity addition resulting from xenon peaking.

Each of the letdown coolers, purification demineralizers, and makeup filters is sized for one-half the maximum letdown flow rate. The makeup tank prefilters are sized for the maximum letdown flow rate.

The letdown and makeup process also accommodates for thermal expansion and contraction of the reactor coolant system during startup and shutdown.

In addition, the high pressure injection portion provides high pressure coolant injection following design basis accidents including a loss of coolant accident. This function is discussed later in this Appendix.

Performance data for the high pressure injection system is shown in Table HPI-1.

2. SYSTEM DESCRIPTION AND MAJOR COMPONENTS

The high pressure injection makeup, and purification system is shown schematically in Figure HPI-1. Component data for the major components listed below is shown in Table HPI-2.

Letdown Cooler - The letdown cooler reduces the temperature of the letdown flow from the temperature of the reactor coolant system to the temperature suitable for demineralization. Heat in the letdown coolers is rejected to the intermediate closed cooling water systems.

Letdown Flow Control - The normal letdown flow rate at reactor operating pressures is controlled by a fixed block orifice. A parallel, normally closed, remotely operated valve can be opened to obtain flow rates up to the maximum letdown capability. This valve is also used to maintain the desired letdown rate at reduced reactor coolant pressures.

Letdown Flow Radiation Monitoring - An in-line letdown radiation monitor is incorporated down-stream of the letdown filter to allow early detection of a failed fuel condition.

Makeup and Purification Filter - Two parallel makeup and purification filters are designed to remove particulate matter from the letdown stream prior to entering the purification demineralizers. These filters minimize accumulation of radioactive crud in the demineralizer resin and in the downstream piping of the purification system and the waste disposal system.

Makeup and Purification Demineralizers - The mixed-bed demineralizers are boric acid saturated and are used to remove reactor coolant impurities other than boron. At the normal letdown rate, each demineralizer can process one reactor coolant volume in 24 hours. Since the reactor coolant may be contaminated with fission and corrosion products, the resins will remove certain radioactive impurities.

Makeup Filters - Two makeup filters (prefilters) are installed in parallel to remove particulates from the effluent of the purification demineralizers to prevent solids from entering the makeup tank.

Makeup Pumps - Three parallel makeup pumps are designed to return the purified letdown fluid to the reactor coolant system and to supply seal water to the reactor coolant pumps. One pump is normally in service.

Makeup Tank - The makeup tank serves as a receiver for letdown, seal return, chemical addition and system makeup; it also provides NPSH for the makeup pump and accommodates temporary changes in system coolant volume.

Seal Injection and Return Filters - Filters are provided in the reactor coolant pump seal injection line to remove particulates which could enter the pump seals and result in increased wear. A filter is provided in the reactor coolant pump return line to prevent solids from entering the makeup tank.

3. MODE OF OPERATION

The makeup system is operated during all phases of the nuclear steam supply system operating life, including startup, power operation, and shutdown. During normal nuclear steam supply system operation, one makeup pump continuously supplies high pressure water from the makeup tank to the seals of each of the reactor coolant pumps and to the reactor coolant system thorugh the makeup line. Makeup flow to the reactor coolant system is regulated by the makeup control valve which operates on signals sensing pressurizer level.

The rate of reactor coolant pump seal injection flow is automatically controlled. A portion of the water supplied to the pump seals leaks off as controlled bleedoff and returns to the makeup tank after passing through the seal return cooler. The remainder of the water is directed into the reactor coolant system as reactor coolant system makeup.

Seal water entering the reactor coolant system makes necessary a continuous letdown flow of reactor coolant to maintain the desired coolant inventory balance. Letdown flow is also required for removal of impurities and boric acid from the reactor coolant. The letdown flow is cooled by one of the letdown coolers, reduced in pressure by the letdown block orifice, and then passed through a filter and purification demineralizer to a three-way valve which directs the coolant to the makeup tank or to the boron recovery system.

Normally, the three-way valve is positioned to direct the letdown flow to the makeup tank. However, if the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to divert the letdown flow to the boron recovery system where the boric acid is removed, or to the waste disposal system. Demineralized water is then added to the reactor coolant via a flow integrator to the makeup tank. During normal operation, the flow integrator, the control rod drive interlock or the operator will terminate dilution. Through the use of boric acid control, the operator may accommodate xenon transients resulting from unit load variations.

The makeup tank also receives chemicals for addition to the reactor coolant. A hydrogen overpressure is maintained in the makeup tank in order to insure that a predetermined amount of dissolved hydrogen remains in the reactor coolant for oxygen control. Chemicals in solution are injected into the makeup tank, which serves as a final mixing location.

System control is accomplished remotely from the control room. The letdown flow rate is set for flow rates other than normal by remotely positioning the letdown flow control valve to pass the desired flow rate. The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves.

Emergency operation of the makeup system is described later in this Appendix.

Normal Systems Functions

Fill - Prior to its heatup and pressurization, the reactor coolant system is filled with borated coolant. When the fill operation is completed, makeup and inventory control is established using the makeup pump.

Reactor Coolant Inventory Regulation - The makeup flow control valve is the primary device which controls the inventory of the reactor coolant system throughout most phases of operation. In addition to the coolant added to the system through the makeup control valve, a large portion of normal makeup water is added through each reactor coolant pump seal. The makeup flow control valve regulates flow by sensing changes in pressurizer level.

During power operation, the makeup portion of the system supplies all inventory regulation of the reactor coolant system. A constant inventory of the reactor coolant system is maintained by the combination of makeup flows through both the makeup flow control valve and the reactor coolant pump seals. Total makeup flow to the system equals letdown flow plus reactor coolant pump seal return plus any system leakage.

During plant shutdown and cooldown, the temperature of the reactor coolant system is decreased, thus causing contraction of the total reactor coolant inventory. Because the volume of the makeup tank is not sufficient to replace the contraction volume of the reactor coolant system during cooldown, additional sources of makeup are required. Other sources of borated water are the chemical addition and boron recovery system and the borated water storage tank.

Reactor Coolant Purification - One of the major functions of the purification system is to purify the reactor coolant system during various phases of reactor coolant system operation.

During periods of heatup leading to power operation, reactor coolant purification is performed through the normal letdown flow path (Figure HPI-1).

It is expected that the greatest amount of crud (contaminant deposits) will be released from the reactor coolant system surfaces during periods of large temperature transients such as heatup and cooldown. For this reason, and in order to prevent highly radioactive contaminants from accumulating in the demineralizer resin beds, the coolant which is to be purified is first passed through letdown filters. Letdown fluid continues to the mixed bed purification demineralizers which are located downstream of the filters. Their function is to remove ionic corrosion products and certain fission products, specifically Mn-54, Mn-56, Co-58, Fe-59, Co-60, Cr-51 and radioiodines. The mixed bed purification demineralizers are boric acid saturated and will remove reactor coolant system impurities other than boron. The effluent from the purification demineralizers is then filtered through additional filters so that all solid impurities are removed.

The letdown filters, the purification demineralizers, and the various other filters, provide purified water of reactor grade for injection into the reactor coolant system both as regular makeup and as reactor coolant pump seal injection water.

Chemical Adjustment for Corrosion Control - The makeup system is used for adding corrosion inhibiting chemicals to the reactor coolant system. All of the chemicals which are added as solutions are delivered to the makeup system upstream of the pre-filters for the makeup tank by the chemical addition and boron recovery system. The two main functions which the chemicals perform are those of oxygen concentration control and pH control. In order to provide oxygen and pH control, specific chemicals are added and adjusted during all phases of reactor operation, including startup, normal operation, shutdown and refueling.

Hydrazine is added to the reactor coolant for oxygen scavenging when the temperature within the reactor coolant system is less than 400°F and there is no significant neutron or gamma flux in the reactor.

For the periods of time when the temperature is above 400°F and there is a significant neutron or gamma flux in the core, oxygen scavenging is effected by maintaining the correct partial pressure of hydrogen in the makeup tank. The specified equilibrium concentration of dissolved hydrogen is thereby maintained in the reactor coolant.

The chemical employed for pH control is lithium hydroxide. Lithium is also produced in the core region of the reactor due to irradiation of the boron dissolved in the reactor coolant. The concentration of boric acid in solution determines the initial pH of the reactor coolant.

<u>Reactor Coolant Pump Seal Injection</u> - The makeup system provides a continuous and dependable supply of clean high pressure water to the reactor coolant pump seals. Seal injection water is supplied at all times when the reactor coolant pumps are operating so that the mechanical seals are subjected to a nearly constant temperature environment. In-line filters are provided in the seal injection line to prevent particulate matter from entering the seal cavity. Individual flow control valves regulate flow to each pump seal.

The reactor coolant pumps may be operated without sealing water, or without intermediate closed cooling water, but not without cooling water supplies. A drop in seal inlet flow to any reactor coolant pump in conjunction with a loss of intermediate closed cooling water, will automatically close the seal return valve for that particular pump. In this way, flow reversal through the reactor coolant pump seal is prevented.

Makeup for the Core Flood Tanks - The core flood tanks, lines, and check valves can be checked for operability during scheduled shutdowns. The water which is permitted to flow out of the core flooding tank during the test is replenished by use of the makeup lines coming from the discharge header of the makeup pump before the reactor is made critical again.

Abnormal Makeup - The makeup (high pressure injection) system provides makeup to the reactor coolant system to replenish inventory loss caused by a small leak in the reactor coolant system pressure boundary. The makeup flow control valve senses a decrease in pressurizer level and positions itself to maintain level. A high flow alarm will occur in the event of excessive makeup flow. If excessive makeup demand results in low makeup tank level, an alarm will result. Additional supplies to the makeup pumps are provided through connections from the borated water storage tank and the sodiuim hydroxide storage tank.

High Pressure Injection System - The high pressure injection, makeup, and purification system serves an engineered safety features function by using the makeup pumps to inject high pressure water into the reactor coolant system in the event of a loss of coolant accident. The safety features function of the system is provided by the injection of high pressure water from the borated water storage tank, and sodium hydroxide storage tank into the reactor coolant system upon Engineered Safety Features actuation (either reactor coolant system pressure less than 1640 psig or reactor building pressure greater than 4 psig). Because of the high discharge pressure capability of the makeup pumps, the borated water can be supplied to the reactor while the reactor pressure is high, and before operation of the decay heat removal system in the low pressure injection mode. In the event of a small reactor coolant system leak where the reactor pressure decreases slowly and the supply of water in the borated water storage tank is reduced to its lower level before the decay heat removal system pumps can overcome the reactor pressure, the makeup pumps will continue to function by taking suction from the outlet of the decay heat removal coolers. In this case, the decay heat removal pumps provide the required net positive suction head for operation of the makeup pumps by recirculating water from the reactor building sump.

Note: The mode of operation of the high pressure injection system described above is referred to by different names within the industry. Therefore, high pressure injection (HPI) in the engineered safeguards (ES) mode, safety injection (SI), emergency core cooling (ECC) and high pressure safety injection (HPSI) all refer to the engineered safety feature (ESF) of the high pressure injection system.

TABLE HPI-1

HIGH PRESSURE INJECTION SYSTEM PERFORMANCE DATA

Normal letdown flow, gpm	45-70
Maximum letdown flow, gpm	140
Total seal flow to each reactor coolant pump, gpm	8-10
Seal inleakage to reactor coolant system per reactor coolant pump, gpm	7- 9
Temperature to seals, °F	140
Purification letdown fluid temperature, °F	120
Makeup tank normal operating pressure, psig	15
Makeup tank water volume, nominal, ft^3	400

TABLE HPI-2

HIGH PRESSURE INJECTION, MAKEUP, AND PURIFICATION SYSTEM COMPONENT DATA

MAKEUP PUMP	
Number	3
Type	horizontal, multistage, centrifugal, mechanical seal
Rated capacity, gpm	300
Rated head, ft.	5545
Motor horespower, hp	700
Pump material	Stainless steel (wetted parts)
Design pressure, psig	3000
Design temperature, °F	200
Seismic category	I
LETDOWN COOLER	
Number	2
Type	shell and tube
Heat transferred Btu/h	16.1×10^6
Letdown flow lb/h	3.5×10^4
Letdown cooler inlet/outlet temperature, °F	55/120
Material, shell/tube	carbon steel/stainless steel
Design pressure (shell/tube), psig	200/2,500
Design temperature (shell/tube), °F	360,600
Code (shell/tube)	ASME VIII/ASME III-C
Seismic category	I

TABLE HPI - 2

(continued)

MAKEUP AND PURIFICATION DEMINERALIZER	
Number	2
Type	mixed bed, boric acid saturated
Material	stainless steel
Resin volume, ft^3	40
flow, gpm	70
Vessel design pressure, psig	150
Vessel design temperature, °F	200
Code	ASME III-2
Seismic category	I
MAKEUP TANK	
Number	1
Volume, ft^3	600
Design pressure, psig	100
Design temperature, °F	200
Material	stainless stee
Code	ASME III-2
Seismic category	I

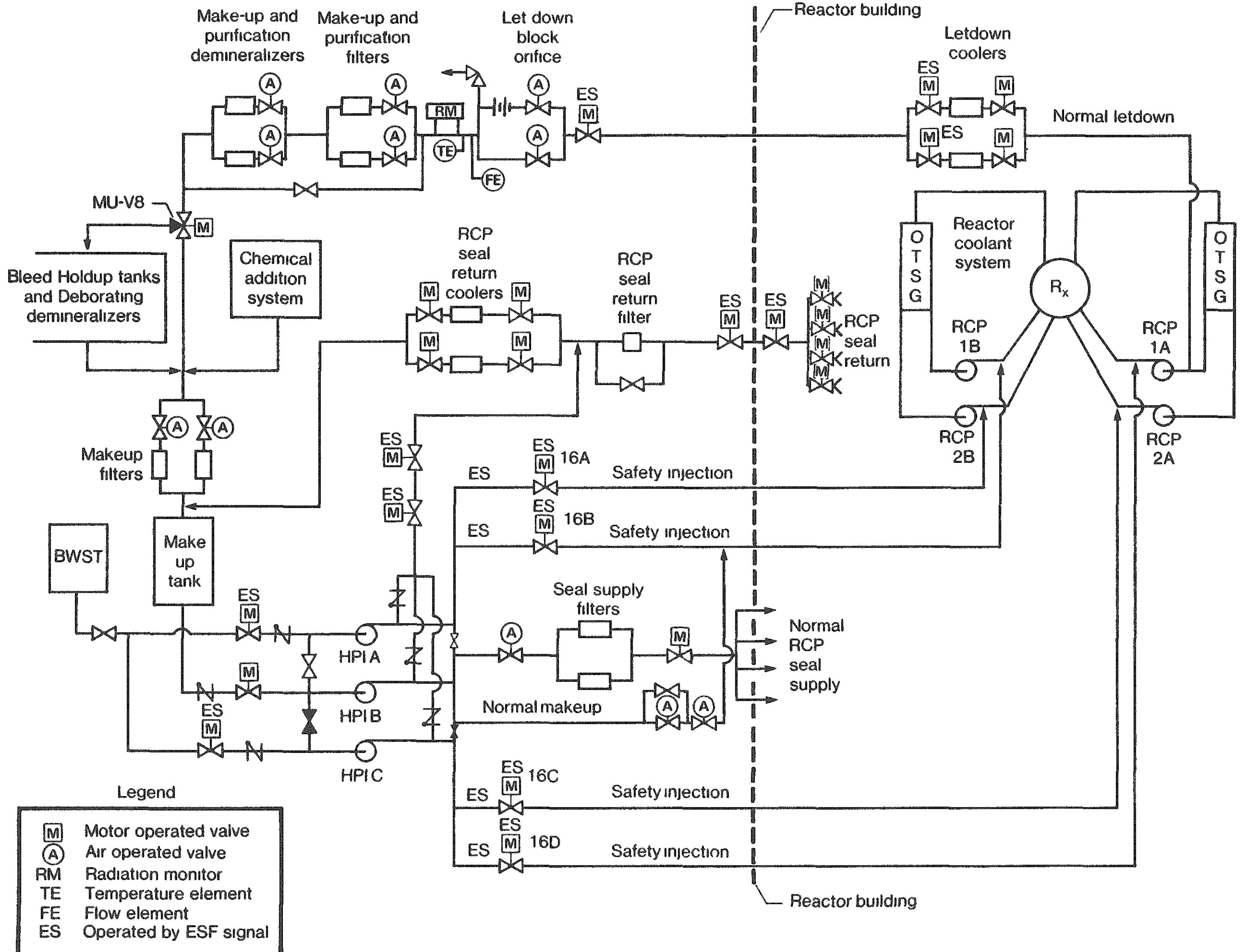

Make-up and purification demineralizers
Make-up and purification filters
Let down block orifice
Reactor building
Letdown coolers
ES
M
A
RM
TE
FE
Normal letdown
MU-V8
Bleed Holdup tanks and Deborating demineralizers
Chemical addition system
RCP seal return coolers
RCP seal return filter
RCP seal return
Reactor coolant system
OTSG
R_x
RCP 1B
RCP 1A
RCP 2B
RCP 2A
Makeup filters
16A
16B
16C
16D
Safety injection
BWST
Make up tank
Seal supply filters
Normal RCP seal supply
HPI A
HPI B
HPI C
Normal makeup
Legend
Motor operated valve
Air operated valve
RM Radiation monitor
TE Temperature element
FE Flow element
ES Operated by ESF signal

Figure HPI-1.

APPENDIX HYD

1. INTRODUCTION

1.1 Purpose and Scope

The issue causing the most concern and public apprehension during the incident at TMI involved hydrogen and the hydrogen bubble. The erroneous assumption that the accumulation of hydrogen within the primary system was or could become explosive led to speculations of a massive spread of contamination and consequent damages to the general population. As was later confirmed publicly, these speculations and fears about the "bubble" were totally unfounded. The presence of even small amounts of free hydrogen prevents accumulation of oxygen and thus any possibility of hydrogen/oxygen explosion. However, the amount of hydrogen produced was sufficient to cause legitimate concerns about core cooling and flammability in the reactor building atmosphere. An ignition, as measured by pressure and temperature spikes, did occur about 10 hours into the incident. Although equipment may have been damaged, the integrity of the reactor building was by no means challenged. The purpose of this appendix is to quantify the extent of the hydrogen generation and discuss possible scenarios.

2. SOURCES OF HYDROGEN IN LIGHT WATER REACTORS

Hydrogen may be produced by a number of mechanisms in light water reactor systems. Among these are the radiolytic decomposition of water, metal/water reaction at elevated temperature, oxidation of materials of construction, and decomposition of organic materials. Significant amounts of hydrogen may be produced only by radiolysis and the zirconium/water reaction.

2.1 Radiolysis of Water

Absorption of energy from ionizing radiation will cause the decomposition of water by a somewhat complicated mechanism to form primarily hydrogen and oxygen.

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

(Hydrogen peroxide is formed as an intermediate which is very rapidly decomposed to water and oxygen above about 200° F.) The yield of this reaction is dependent upon the energy absorbed, the nature of the radiation, temperature, reaction products residence time, etc.

For example, boiling increases the radiolysis rate by 30 times over non-boiling. Gamma radiation from cobalt-60 in boiling water generates about 0.45 molecules of hydrogen per 100 ev of energy absorbed(1). Neutrons, however, in boiling water generate about 1.1 molecules per 100 ev absorbed. The presence of 0.04 ppm of dissolved hydrogen totally suppresses radiolysis at 77°F in non-boiling systems while 0.7 ppm stops radiolysis in boiling water at 525°F(2). Standard practice in pressurized water reactors is to add about 1 ppm of hydrogen to the water in the reactor coolant system (RCS) to prevent radiolysis.

Throughout the incident at TMI-2, the dissolved hydrogen levels in the RCS were considerably above 1 ppm. Thus, radiolysis in the RCS was a source of neither hydrogen nor oxygen. (Radiolysis may have produced inconsequential quantities of hydrogen in the reactor building sump where dissolved hydrogen levels may have been considerably lower.)

2.2 Zirconium/Water Reaction

Above 1600°F zirconium alloys react with water to form hydrogen and zirconium oxide.

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

The reaction rate increases with temperature and is very rapid above 2700°F. Stoichiometrically, about 8 standard cubic feet of hydrogen are produced per pound of zirconium oxidized.

3. HYDROGEN COMBUSTION

3.1 Limits of Combustion

Near room temperature in air mixtures containing between 1 ppm and 10% moisture, the lower limit of upward flammability is 4% hydrogen. The lower limit for downward flame propagation is 8% and the lower limit of detonability is 19%(3). Combustion of hydrogen mixtures between 4% and 9% is often incomplete, leaving about 4% unburned(4).

When the temperature is elevated and water content is high, the oxygen content is often the limiting factor. Existing data indicate that between 300°F and 500°F with at least 50% steam the lower limit of flammability is about 4% oxygen with detonability repressed below 9% oxygen(5).

3.2 Pressure Increase from Hydrogen Combustion

For H_2 concentrations between 4% and 9%, combustion in large containers results in pressure increases that depend upon:

1. the temperature of the pre-combustion mixture.
2. the ignition mechanism - multiplicity and energy.
3. the H_2 concentration.

4. the vessel wall material.
5. the vessel size and shape.

Theoretical calculations, based on the energy of combustion and assuming that all hydrogen is burned, provide an upper bound to the pressure increase. Between 8% and 10% hydrogen, the transition region for downward flame propagation[4 & 7], the reaction may go to completion. This low concentration combustion exhibits slow flame propagation (1 to 10 m/sec.)[8]. Large containers and ignition multiplicity increase the flame velocity. This type of combustion yields a maximum pressure rise factor of about 8.

When the concentration of hydrogen reaches about 19% in air with normal humidity, a change in combustion phenomena occurs with the possible transition to supersonic combustion (detonation)*. This transition requires either a detonative ignitor or the shock wave reflection from at least two hard surfaces[9]. For example, detonation nearly always occurs for ignition of these compositions in pipes while in free balloons it almost never occurs.

Figure 1 illustrates the relationship between hydrogen concentration and pressure rise. Theoretical and experimental data are shown.

The pressure rise because of a detonation (hydrogen concentrations between 19% to 70%) is about 17 times the initial pressure[10]. This does not include consideration of reflection amplification or the pre-detonation compression observed in piping systems.

* With bone dry air (< 1 ppm water), detonations have been observed as low as about 14% hydrogen.

4. HYDROGEN PHENOMENA AT TMI-2

Figures 2 and 3 show a pressure pulse of approximately 28 psi that occurred at 1350 on March 28, 1979. (Figure 3 shows this pulse as an apparent decrease in steam pressures.) The reactor building temperature measurements indicated a nearly uniform increase of about 40°F (Figure 4) at the same time. No similar indications were found at any other time during the event.

According to Figure 1, a pressure rise of 28 psi will result from ignition of about 8% hydrogen. This is also consistent with the rate of pressure increase(7).

Hydrogen generation in the containment by radiolysis may be estimated from Regulatory Guide 1.7. A conservative analysis can only account for less than about 0.1% contribution to the hydrogen concentration during the the first day of the incident. Thus, metal-water reaction is the only significant hydrogen source. No measurements of the reactor building atmosphere composition were made before the pressure transient.

The earliest reactor building atmosphere measurements were made on March 31 at 0600(6). These were:

Sample 1	H_2 - 1.7%
	O_2 - 15.7%
	N_2 - 82.6%
Sample 2	H_2 - 1.7%
	O_2 - 16.5%
	N_2 - 81.8%

A series of samples taken April 1 and 2 averaged 2.1% H_2 (average of sixteen measurements) and 18.6% O_2 (eight measurements). These are documented in the secondary chemistry log book. This disconnect in oxygen measurements poses a dilemma since the oxygen depletion in the reactor building is a direct measure of the magnitude of the hydrogen generation and subsequent burn. Later reactor building measurements taken on June 1[13] (0.6% H_2, 14.5% O_2) and August 2 (0.6 % H_2, 14.1% O_2) tend to confirm the March 31 results.

The only additional data needed to close material balances are the hydrogen inventory in the reactor coolant system, both dissolved and in the bubble. Using the known temperature and pressure of the system, the dissolved hydrogen can be calculated from Henry's law constants shown in Figure 5. The volume of the hydrogen bubble used was that reported by Metropolitian Edison[14]. Figure 6 shows the bubble behavior with time both with and without accounting for effects of hydrogen solubility.

Thus, there are five time frames for material balances to be performed:

1. Just prior to the hydrogen ignition, March 28, 1979
2. March 31, 1979 [6]
3. April 1-2, 1979 (Log book data)
4. June 1, 1979 [13]
5. August 2, 1979 [15]

Note that the boundary for the material balances is the reactor building. Thus any hydrogen which escaped the reactor building will also escape the material balance. It is estimated that between 2.5 and 10 million curies of noble gas (mostly Xe-133) were released during the first few days of the accident. A corresponding amount of hydrogen can be assumed to have also escaped since the hydrogen and noble gases were released from the core at about the same time. Thus, since a maximum 10% of the noble gas released from the core was released to the environs, as much as 10% of the hydrogen released by metal water reactor in the core was also released to the environs. This produces a systematic error in the material balances of up to + 10%. Similarly, depletion of the oxygen in the air of the reactor building by other than hydrogen oxidation (e.g. corrosion) will produce errors in the negative director (i.e., less apparent zirconium/water reaction.

Basis for calculations:

Dry air contains 20.9% Oxygen by volume.

The reactor primary system volume is 1.18×10^4 cubic feet. (including the pressurizer)

The total zirconium inventory in the primary system is 53,000 lb.[11]

The reactor building free volume is 2.05×10^6 cubic feet* at 120°F, 14.7 psia, and 100% relative humidity, this corresponds to 1.5×10^6 standard cubic feet or 4200 lb moles of dry gas.

A summary of the five material balances is shown in Table 1.

4.1 Material Balance Just Prior to Hydrogen Ignition

From post-ignition analyses of the containment atmosphere it is calculated that before the ignition the hydrogen concentration was 8%. This corresponds to about 340 pound moles of hydrogen or, stoichiometrically, about 16,000 pounds of zirconium reacting in the metal/water reaction (29% of the total zirconium inventory). The primary coolant with 825 psi of hydrogen overpressure (875 psia total pressure) and at 280°F is calculated to contain about 34 lb moles of dissolved hydrogen (3% metal-water reaction). Assuming a 1400 ft^3 bubble[14] in the primary system at these conditions, (150 pound moles of hydrogen) represents another

* Burns and Roe calculated value

13% of the total zirconium-water inventory. Thus, by material balance a total of 45% zirconium-water reaction occurred.

4.2 Material Balance Using March 31 Data

As earlier noted, two samples of the reactor building atmosphere were taken at 0600 on March 31, 1979. The hydrogen concentration was 1.7% and the average oxygen concentration was 16.1 $\pm$ 0.4%. The oxygen depletion was then 4.8 $\pm$ 0.4% representing 410 $\pm$ 30 lb moles hydrogen or 35 $\pm$ 3% metal-water reaction. As in Section 4.1 (using 823 ft^3 for the bubble volume), the hydrogen inventories are 36 lb moles in the coolant and 86 lb moles in the bubble. These values correspond to 3% and 7% metal-water reaction, respectively. The residual hydrogen in the containment atmosphere is 72 lb moles representing 6% metal-water reaction. Thus the total indicated metal-water reaction based on these data is 51%.

4.3 Material Balance Using April 1-2 Data

Measurements made April 1 and 2 indicate a hydrogen concentration of 2.1% and oxygen concentration of 18.6%. As calculated above, the oxygen depletion corresponds to 200 lb moles hydrogen or 17% metal-water reaction. The 2.1% residual hydrogen represents 89 lb moles or 8% metal-water reaction. The volume of the bubble at this time was very low (probably zero) while the dissolved hydrogen in the coolant may have represented up to 3% metal-water reaction. This totals 28% metal-water reaction.

4.4 Material Balance for June 1 Data

By June 1, the hydrogen concentration had been reduced to 0.6% by operation of the hydrogen recombiners. The oxygen concentration was 14.5%. By oxygen depletion, about 540 lb moles of hydrogen were formed (47% metal water). Residual hydrogen is another 26 lb. moles (2% metal water). Dissolved hydrogen in the primary coolant is minimal (< 1 lb mole). A total amount of metal-water reaction of 49% is indicated.

4.5 Material Balance for August 2

On August 2, the hydrogen and oxygen concentrations in the reactor building were 0.6 and 14.1%, respectively. These correspond to 2% and 50% zirconium-water reaction.

5. CONTAINMENT CAPABILITY

The hydrogen combustion event produced a peak pressure of 28 psig. The containment design pressure is 60 psig; thus, there was no threat to containment structural integrity due to the combustion of the hydrogen resulting from the oxidation of an estimated 29% of the zirconium.

The possibility of the hydrogen concentration increasing substantially above the lower limit of flammability without ignition is reduced by the great abundance of ignition sources located throughout the reactor building. These sources include approximately 100 motors for valves and various rotating equipment and contacts associated with position indication and plant parameters.

Recent work has shown that even if 100% of the total zirconium in the core had reacted the reactor containment would have remained intact. Ultimate failure of the containment would occur at 150 to 190 psia via tendon failure.

TABLE 1

MATERIAL BALANCE SUMMARY

Pound Moles Hydrogen Equivalent

NUMBER	TIME	REACTOR BUILDING		RCS		TOTAL
		H_2	O_2	BUBBLE	DISSOLVED	
1	Preburn	340	0	150	34	520
2	March 31	72	410	86	86	600
3	April 1-2	89	200	< 10	36	330
4	June 1	26	540	0	< 1	570
5	August 2	26	580	0	< 1	610

% Zirconium-Water Reaction

NUMBER	TIME	REACTOR BUILDING		RCS		TOTAL
		H_2	O_2	BUBBLE	DISSOLVED	
1	Preburn	29%	0%	13%	3%	45%
2	March 31	6%	35%	7%	3%	51%
3	April 1-2	8%	17%	< 1%	3%	28%
4	June 1	2%	47%	0%	<0.1%	49%
5	August 2	2%	50%	0%	<0.1%	52%

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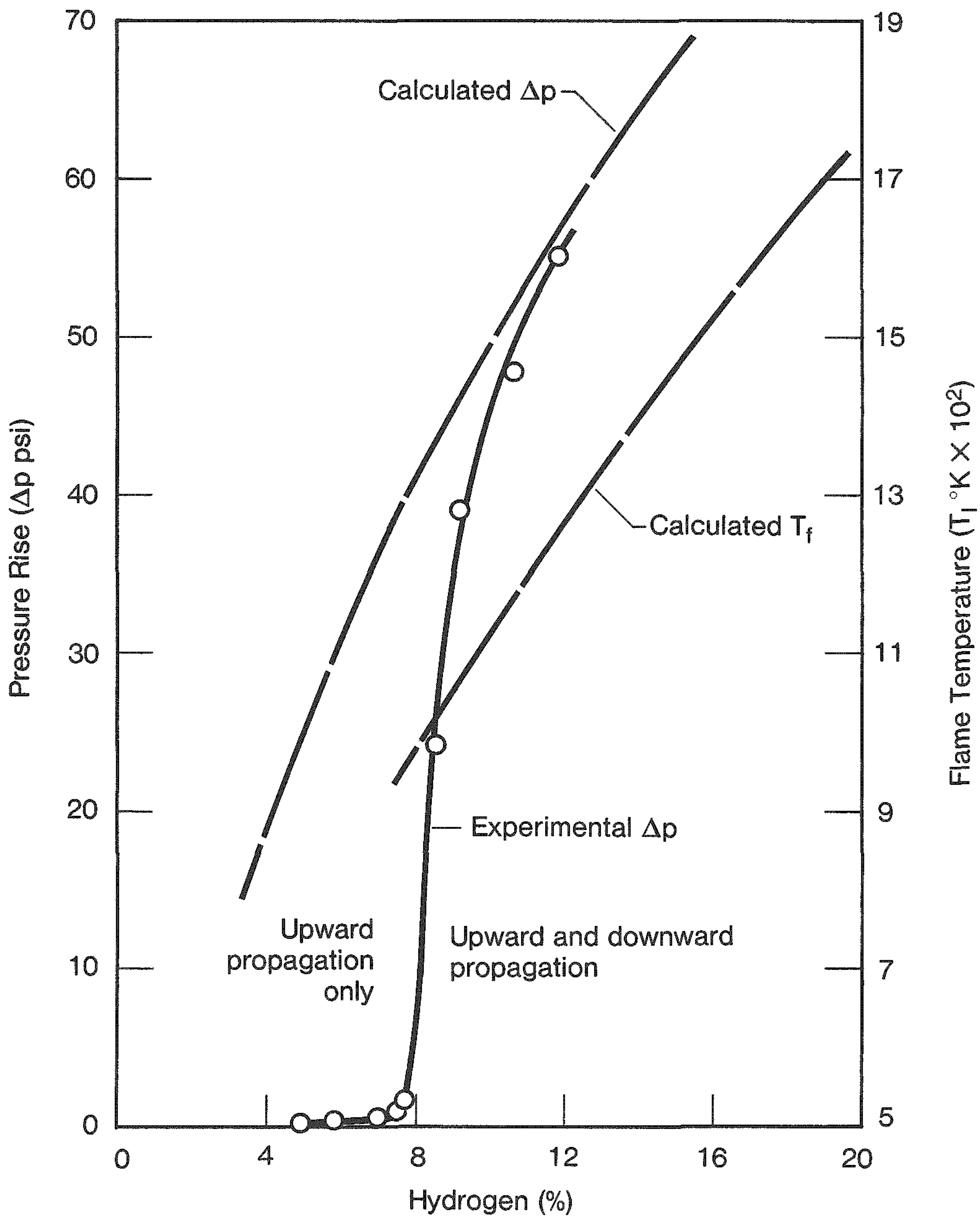


Figure HYD 1. Pressure Rises Resulting From Ignition of Hydrogen-Air in 12 Foot Diameter Sphere at 18°C

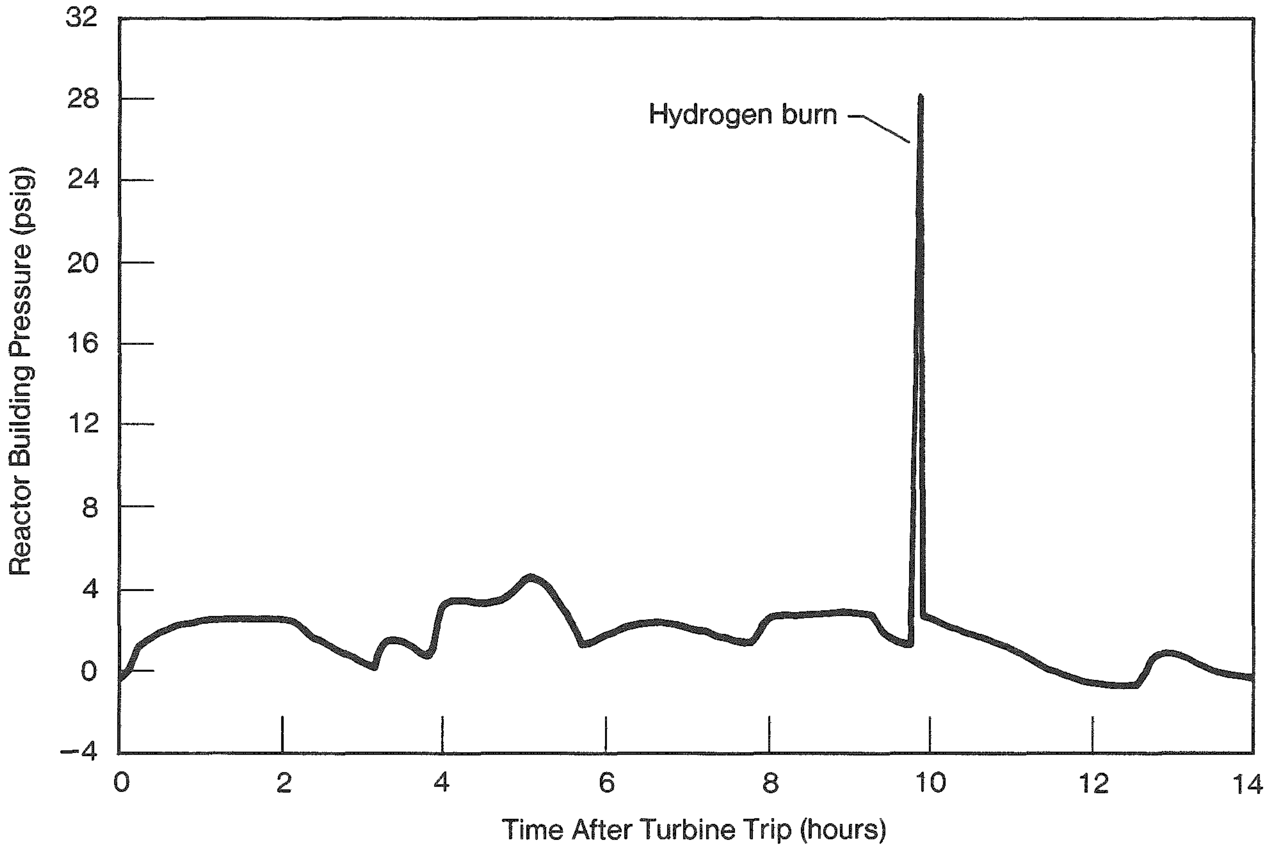


Figure HYD 2. Reactor Building Pressure Versus Time

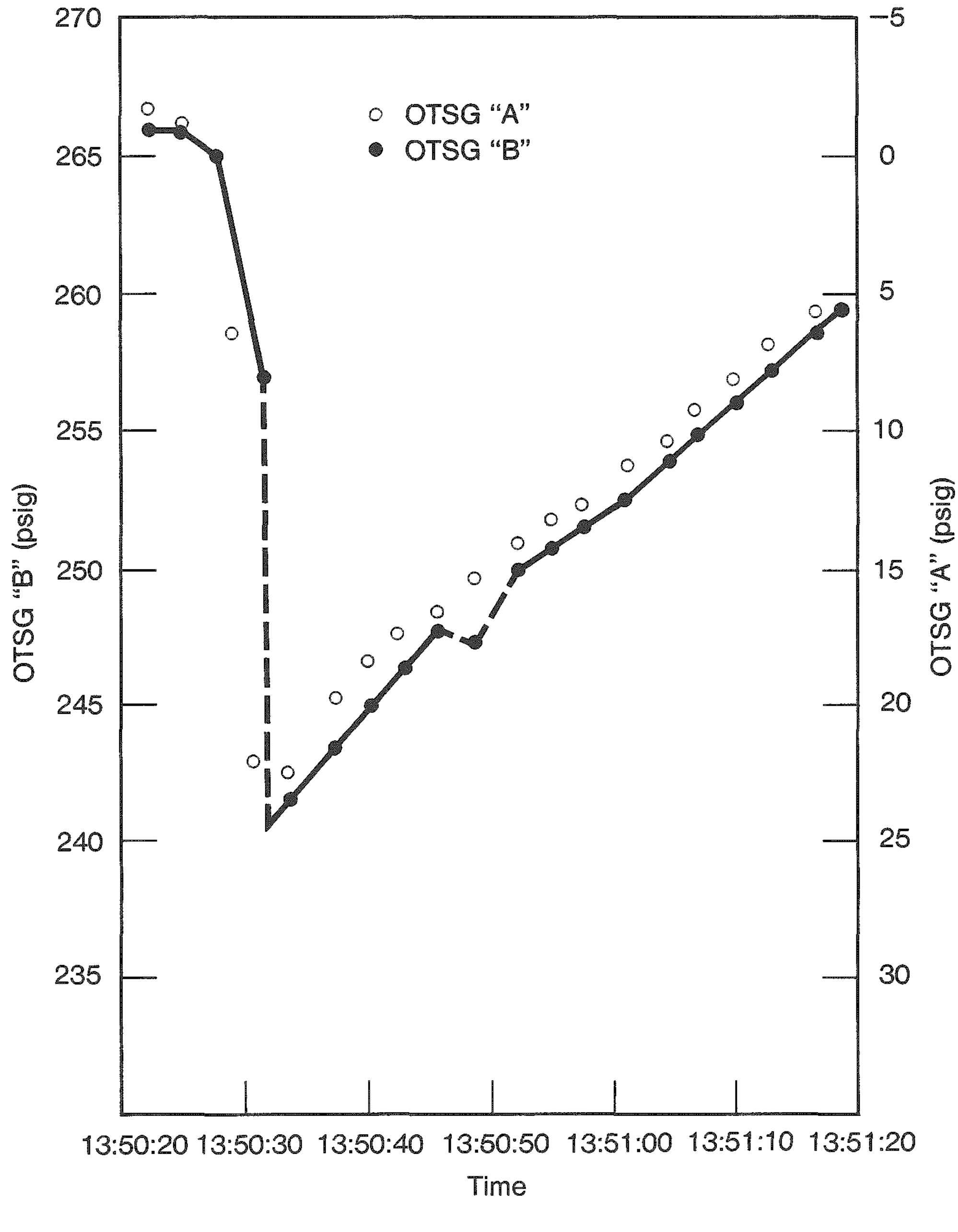


Figure HYD 3. Effect of H_2 Ignition on Measured Steam Pressures

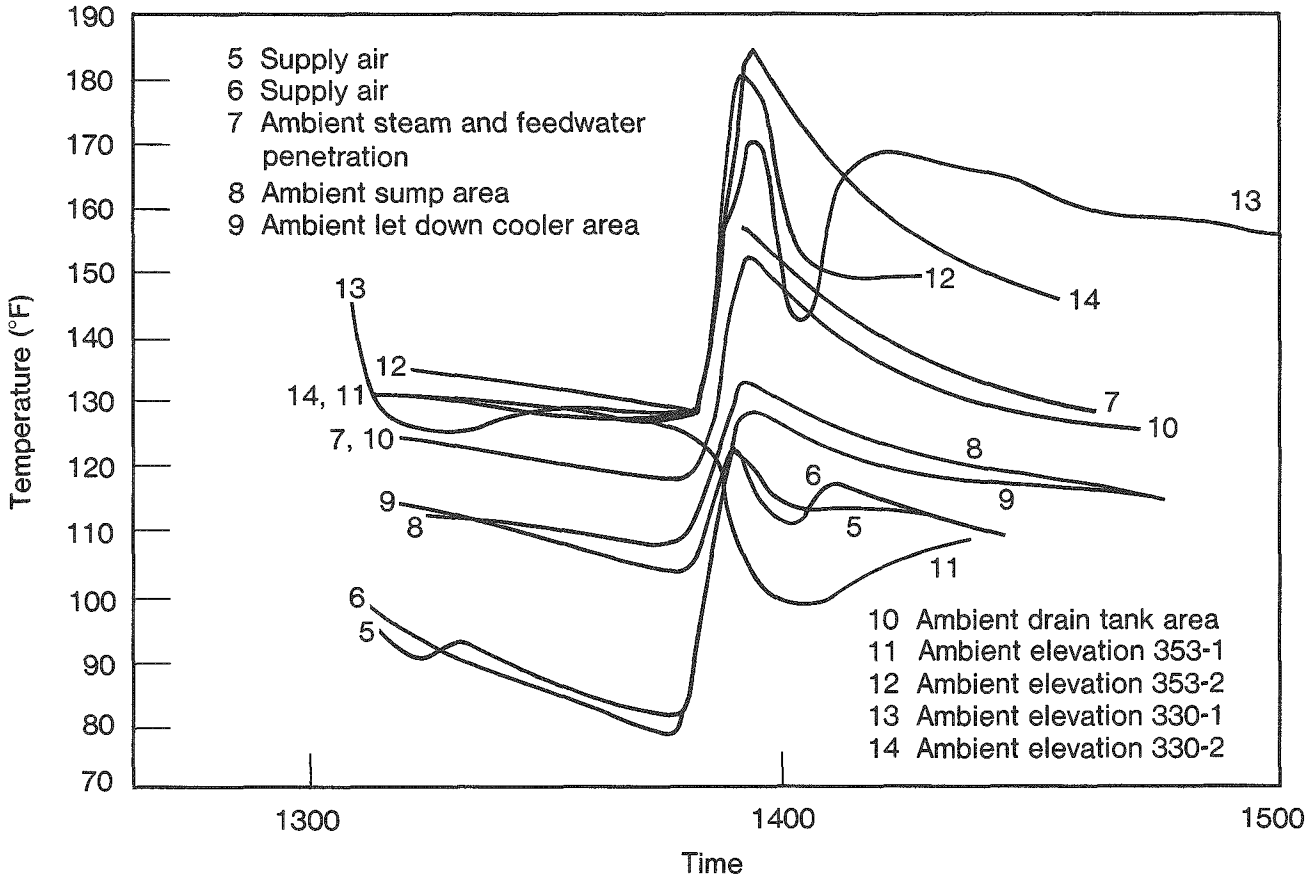


Figure HYD 4. Reactor Building Temperatures During Hydrogen Ignition

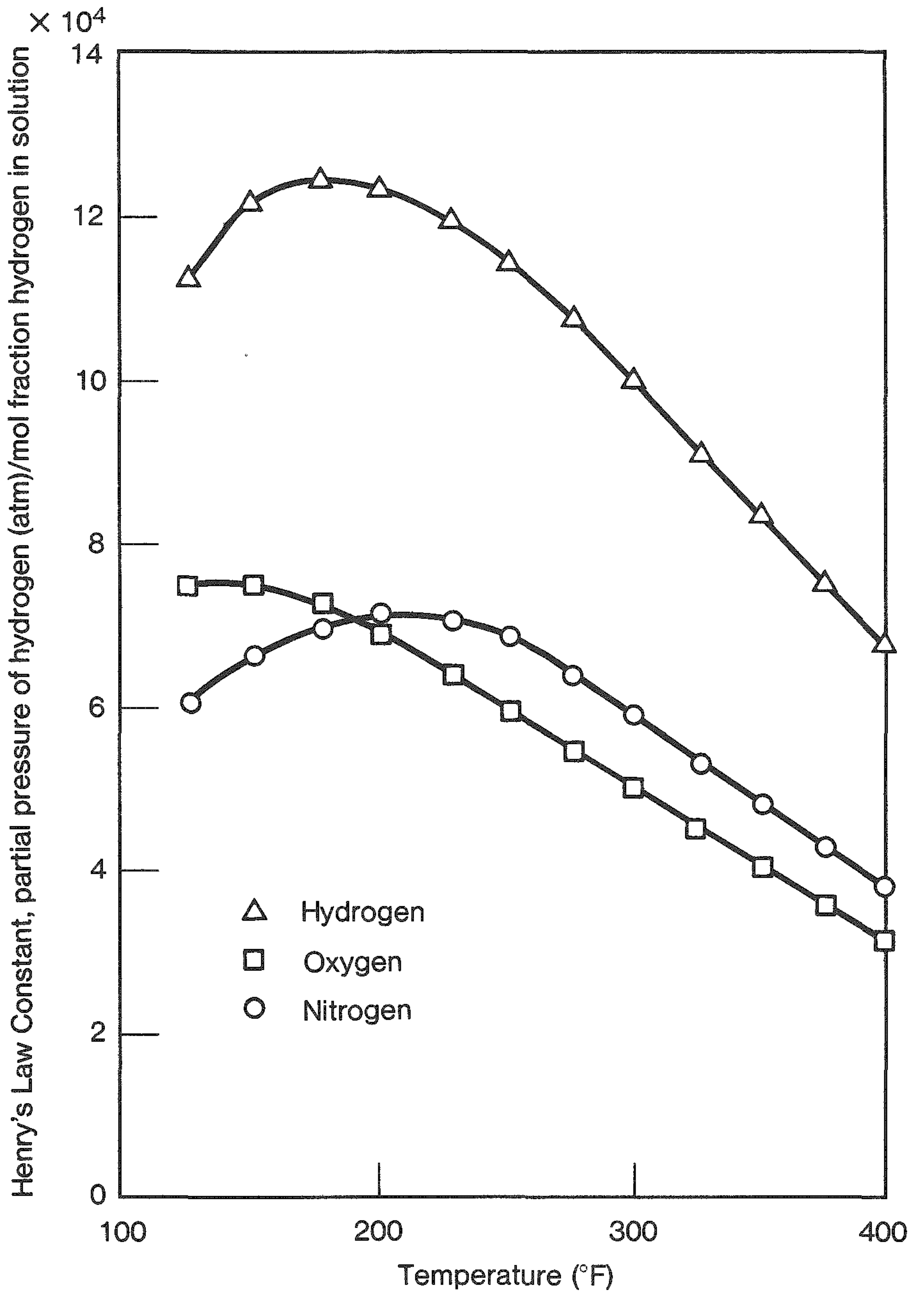


Figure HYD 5. Henry's Law Constants Versus Temperature

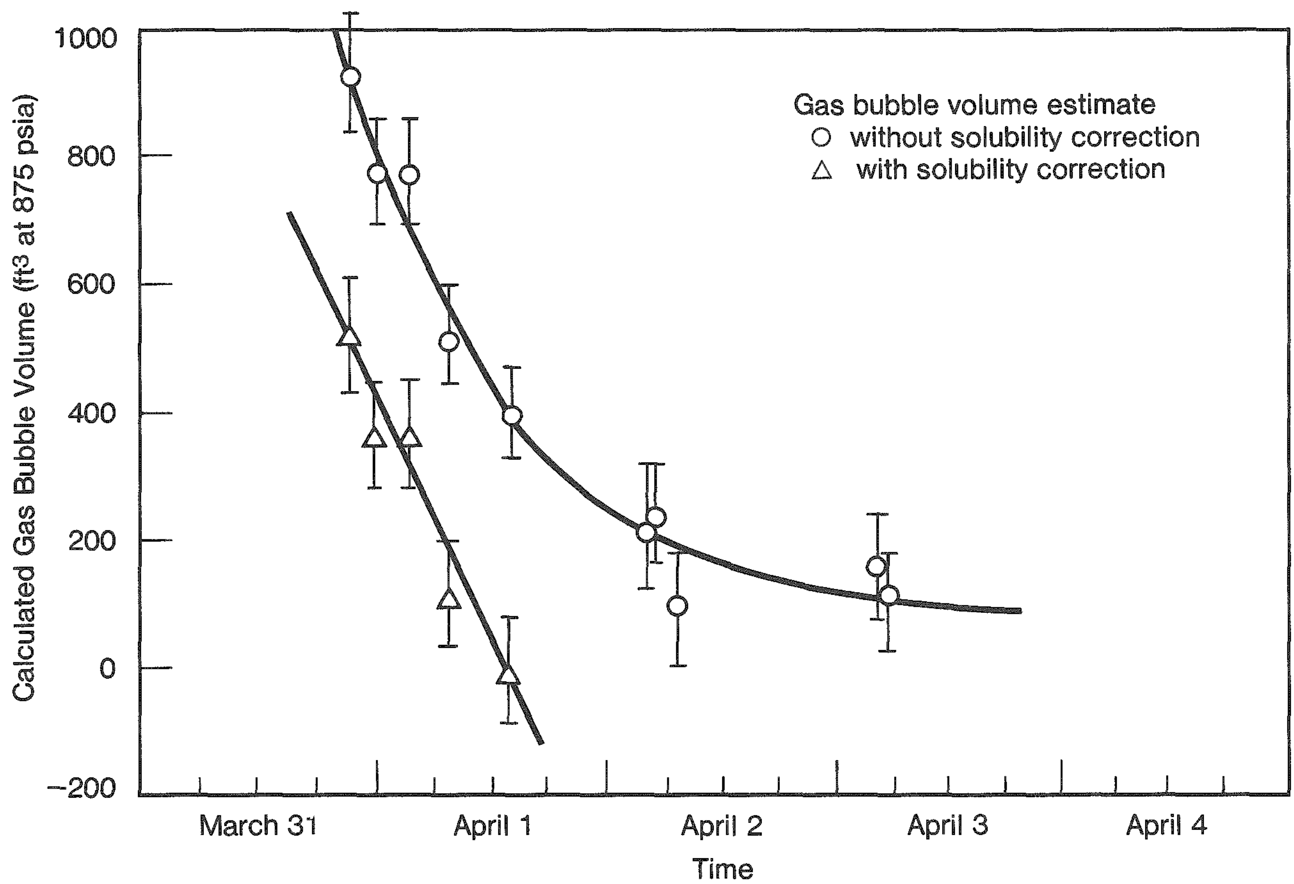


Figure HYD 6. Calculated Volume of Hydrogen "Bubble" in Pressure Vessel Versus Time

APPENDIX ICS

INTEGRATED CONTROL SYSTEM

The purpose of the integrated control system (ICS) is to maintain a match between the power produced in the reactor and the power (megawatts) generated by the turbine-generator. The ICS accomplishes this purpose by controlling the power in the reactor, the rate of steam production in the steam generator, and the megawatt output of the turbine-generator.

There are three basic ways of controlling a reactor/steam generator/turbine-generator unit. These are:

A. Reactor/steam generator following mode where a turbine load is established and the reactor/steam generator system maintains the required steam conditions.

B. Turbine following mode where the reactor/steam generator system establishes a steam output and the valves which control steam flow to the turbine maintain constant steam conditions, thus determining generator electrical output.

C. The integrated reactor/steam generator/turbine mode of control which is a combination of both A and B.

The portion of the ICS that controls reactor power uses a comparison of megawatt demand, core thermal power, and reactor coolant system temperature to produce a demand for a certain control rod position. The portion of the ICS that controls steam production in the steam generators uses a comparison of primary and secondary system parameters to produce a steam generator water level demand signal which is then used by the speed control system of the main feedwater pumps and the position control system of the main feedwater control valves. The portion of the ICS that controls the electrical generation from the turbine-generator uses a comparison of secondary system parameters and load demand to produce a demand for a certain turbine throttle valve position. The steam pressure and steam temperature to the turbine throttle are held constant.

When the ICS is in any mode of control other than the integrated mode, it shifts into a mode of operation called tracking. In this mode certain functions are constrained in amplitude or rate by certain conditions existing in the power plant. It is in this mode that the ICS provides control demands, as required, to the turbine bypass valves, the atmospheric dump valves, and the emergency feedwater valves.

In the event which occurred at TMI-2, the ICS "tracked" the loss of both main feedwater pumps and sent a runback signal to the reactor power control system, steam generator water level control systems, and the turbine control system which controls the turbine buypass valves and, when they were not available for use, the atmospheric dump valves.

Detailed study of the ICS has not been a part of the NSAC study of the Three Mile Island 2 accident. However, it appears on the basis of a general review that the ICS functioned as designed and did not contribute to the occurence or severity of the accident.

APPENDIX OPS

OPERATOR ACTION

INTRODUCTION

The Three Mile Island-Unit 2 plant operators took many significant actions during the sixteen hours covered in this report. These actions are presented here in a combination event train and narrative form. Each specific operator action identified in this event train is stated, and where applicable, a correlation is made between the operator's actions and the system's response. Some system responses were operator initiated while others derived from happenings in the plant. The event train includes the time, the system response or operator action, and remarks, for the first two hours and fifty-five minutes of the accident. Event descriptions and narrative summaries are provided at selected time intervals to indicate the condition of the plant.

References to the operator's written procedures are included to provide a basis for understanding the operator's response. It is assumed that the operators were attempting to implement procedures required by the Station Technical Specification 6.8 (Ref. 2). These originate in the guidelines set forth in NRC Regulatory Guide 1.33 (Ref. 1). The procedures which are felt to be most significant to this event are:

Loss of Steam Generator Feed (Ref. 6)
Reactor Trip (Ref. 3)
Loss of Reactor Coolant/Reactor Coolant System Pressure (Ref. 12)

Other events during the accident suggest that other emergency procedures were used. Attachment 1 provides a listing of procedures that are significant to the presentation, but this list does not represent all procedures which could have been applicable or used during the accident. Attachment II is a graphic presentation of various procedures appropriate to plant conditions during the first three hours. Attachment III contains a listing of NRC interviews with plant personnel that are referenced herein.

The event train was terminated at two hours and fifty-five minutes because plant conditions had deteriorated to the point where existing emergency procedures and training were inadequate for plant recovery. Beyond this time, the use of the event train format for detailed analysis and explanation of operator response would provide little insight into the actions taken by the operators. Beyond two hours and fifty five minutes, a closing summary is used to describe what the operators were attempting to do.

This description of operator actions has been terminated at the point where forced circulation using a reactor coolant pump, achieved a stable mode of decay heat removal. This appendix is not intended to judge operator actions or the adequacy of procedures and training.

Operator Response
Sequence of Events

Note: Asterisk (*) indicates events which were performed by the operators

TIME	EVENT/OPERATOR ACTION	REMARKS
00:00:01 (0400:36)	Condensate pump 1A (CO-P-1A) tripped.	Initiating event. It is believed that water in the instrument air lines to condensate polishing valves caused them to fail closed resulting in a low suction pressure trip of the condensate booster pumps. Through an electrical interlock, condensate pump 1A tripped. (Ref. A, p. 25)
00:00:00 (0400:37)	Feedwater pumps 1A & 1B (FW-P-1A & FW-P-1B) tripped.	The trips were caused by low feedwater pump suction pressure. Emergency Procedure for Loss of Steam Generator Feed (Ref. 6) becomes applicable.
00:00:00 (0400:37)	The main turbine tripped.	Automatic response to loss of both feedwater pumps. Abnormal Procedure for Turbine Trip (Ref. 4) becomes applicable. Operator interviews indicate required actions were implemented (Ref. C, p. 4).
00:00:00 (0400:37)	Three emergency feedwater pumps 1, 2A & 2B (EF-P-1, EF-P-2A, and EF-P-2B) start.	Automatic response to loss of both feedwater pumps. Operators verified pump operation (Ref. C, p.4).
00:00:03 (0400:40)	The pressure setpoint (2255 psig) of electromatic relief valve (ERV) (RC-R2) located on the pressurizer was exceeded.	RC-R2 opened. This is an expected response.

TIME		EVENT/OPERATOR ACTION	REMARKS
00:00:08 (0400:45)		The reactor tripped on high reactor coolant system pressure.	The nominal reactor trip setpoint is 2355 psig. Emergency Procedure for Reactor Trip (Ref. 3) becomes applicable. Alarm response procedures (Ref. 16) for all actuated alarms become applicable
00:00:08 (0400:45)	*	Pressurizer heater controls were placed in the automatic mode.	This was done to mitigate expected reactor coolant system pressure transients following any reactor trip. The Reactor Trip Procedure assumes automatic control of RCS pressure and pressurizer level (Ref. B, p. 6).
00:00:12 (0400:49)		Reactor coolant system pressure decreased below the setpoint value for ERV (RC-R2) closure.	The ERV should have reseated (closure setpoint was 2205 psig), but it remained in the full open position. It indicated closed. (Refer to Appendix ERV and TH)
00:00:12 (0400:49)		Indicated pressurizer coolant level peaked at 256 in. and began a rapid decrease.	A momentary coolant insurge followed by a rapid coolant outsurge was anticipated by the operators following a reactor trip which led to the actions below in order to maintain a coolant level in the pressurizer.
00:00:12 (0400:49) Approximate	*	Letdown flow was stopped.	These two manipulations were taken by the operators as required by steps 2.2.d and e of the Emergency Procedure for Reactor Trip (Ref. 3) which requires maintaining a pressurizer coolant level of 100" (Ref. C, pp. 2 & 3).
00:00:13 (0400:50)	*	An attempt was made to start makeup pump 1A (MU-P-1A) and open a high pressure injection isolation valve.	
00:00:13 (0400:50)		A condenser hotwell low water level alarm was received on the alarm typewriter.	Nominal alarm setpoint was 22.5 in.
00:00:14 (0400:51)		Emergency feed pump discharge pressures for the three pumps were indicated on the alarm typewriter.	This signifies pumps were in operation.

TIME		EVENT/OPERATOR ACTION	REMARKS
00:00:15 (0400:52)		Steam generator A water level indicated 74 in. on the startup range. Steam generator B water level indicated 76 in. on the startup range.	Steam generator water levels were decreasing.
00:00:30 (0401:07)		The ERV (RC-R2) and pressurizer safety valve (RC-R1B) outlet temperatures alarmed high.	RC-R2 outlet temperature indicated 239.2°F. RC-R1B outlet temperature indicated 203.5°F. Earlier in the shift, RC-R1B had alarmed at 200° F. Pressurizer safety and relief valves discharge to the reactor coolant drain tank.
00:00:30 (0401:07)		Steam generator A water level decreased to 27.3 in. on the startup range.	The operators noted decreasing steam generator levels, main feed pumps tripped, emergency feed pumps in operation and associated emergency feed control functioning (Ref. C, p. 4). The turbine trip procedure (Ref. 4, step 3.0) requires the operator to "...verify emergency feed pump started and are delivering water to the OTSG's." No direct flow measurement was available to the operators. Flow can be deduced by observing pump discharge pressure and/or OTSG's level trends.
00:00:33 (0401:10)		Steam generator B water level decreased to 28.8 in. on the startup range.	
00:00:41	*	Makeup pump 1A (MU-P-1A) was started.	Refer to 00:00:13 entry. At this point, two makeup pumps were running in an attempt to stop the anticipated decrease in pressurizer level.
00:00:48 (0401:25)		Pressurizer coolant level reached an indicated minimum of 158 in.	Indicated pressurizer coolant level began to increase. Based on past experience, operators expected a drop to approximately 90".
00:00:58 (0401:35)		A pressurizer low coolant level alarm was received on the alarm typer.	Nominal alarm setpoint was 200 inches.

- Plant Status after One Minute -

At one minute into the transient, the operators were responding to the turbine trip, the reactor trip, and complications in the condensate and feedwater systems as indicated by numerous alarms.

TIME	EVENT/OPERATOR ACTION	REMARKS
00:01:00 (0401:37)	Pressurizer safety valve (RC-R1A) outlet temperature alarmed high.	
00:01:13 (0401:50)	A condenser hotwell high water level alarm was received.	Nominal alarm setpoint was 36 inches. Operator interviews indicate their attention was diverted to hotwell level control problems which if not corrected could lead to loss of vacuum by flooding vacuum system piping (Ref. A, p. 25). In addition, concern was expressed over the potential for water hammer in the turbine bypass lines (steam dump to condenser) (Ref. P, p.8). Either of these conditions would result in the loss of the primary heat sink.
00:01:45 * (0402:22) Approximate	Both steam generators boiled dry on the secondary side.	Upon recognizing this, the operator placed the emergency feedwater control valves to the full open position (Ref. E, p. 3).
00:02:01 (0402:38) through 00:02:04 (0402:41)	Engineered safety features (ESF) for safety injection actuated automatically. Makeup pump 1B (MU-P-1B) tripped. High pressure injection pump 1C (MU-P-1C) started automatically. Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) started	The ESF actuated due to the reactor coolant system pressure decreasing below 1640 psig.

TIME		EVENT/OPERATOR ACTION	REMARKS
00:03:13 (0403:50)	*	The safety injection portion of engineered safety features was manually bypassed.	This action allows the operator manual control of pumps and valves, as necessary, to control reactor coolant system inventory, prevent makeup pump runout, and to terminate sodium hydroxide injection. At this point the operators were unaware that the ERV had not closed. An ESF actuation caused by a low reactor coolant system pressure following a reactor/turbine trip was not considered unusual, having occurred on at least two previous occasions on Unit #2 (Ref. K, pp. 22 & 25).
00:03:13 (0403:50)		The reactor coolant drain tank relief valve (WDL-R1) lifted at approximately 122 psig. Nominal setpoint is 150 psig.	
00:03:26 (0404:03)		A reactor coolant drain tank high temperature alarm was received.	Either of these last two events indicate a lifted pressurizer relief or safety valve which the operator would expect during this transient. Drain tank pressure is indicated on Panel 8A located behind the vertical panels and would not be observed unless an operator were stationed at the panel.
00:03:28 (0404:05)		A pressurizer high coolant level alarm was received.	Nominal alarm setpoint was 260 inches.
00:04:38 (0405:15)	*	The operator stopped makeup pump 1C (MU-P-1C) and throttled the high pressure injection isolation valve.	Pressurizer level was 363 inches and increasing rapidly. These actions indicate that the operators were attempting to prevent solid reactor coolant system operation and to maintain pressurizer level in the range required by Emergency Procedures for Reactor Trip and Loss of Coolant (Ref. 3 & 12) (Ref. D, pp. 14 & 25) Ref. 15 states: "The Pressurizer must not be filled with water to indicated solid water conditions (400 in.) at any time

TIME		EVENT/OPERATOR ACTION	REMARKS
			except as required for system hydrostatic tests." Ref. 13 states: "The pressurizer/ RC system must not be filled with coolant to solid conditions (400 inches) at any time except as required for system hydrostatic tests."
00:04:52 (0405:29)	*	Intermediate closed cooling pump (IC-P-1A) was started.	Intermediate closed cooling pump (IC-P-1B) had been in operation. The operation of two intermediate closed cooling water pumps indicates the operators were preparing to establish maximum letdown flow in accordance with system operating procedures (Ref. 14).
00:04:58 (0405:35)	*	An alarm was received that letdown flow was off scale.	This alarm indicates the operator was responding to the high level condition in the pressurizer by opening MUV 376, reestablishing letdown to arrest the increase in pressurizer coolant level.
00:05:00 (0405:37)		Pressurizer coolant level reached 377 in., decreased momentarily, then continued to rise.	
00:05:15 (0405:52) through	*	Condensate pump 1A (CO-P-1A) was started.	The operators were attempting to reestablish secondary plant operating conditions in accordance with the requirements of Emergency Procedure for Loss of Steam Generator Feed (Ref. 6). A condensate pump is required to establish hotwell level control (Refer to 00:01:13 entry).
00:06:29 (0407:06)	*	A condensate booster pump 2B (CO-P-2B) trip signal was received three times, followed by a trip clearing (normal) signal in each instance.	The operator's inability to start the pump was apparently caused by a low suction pressure trip when the pump started turning. The suction flowpath was blocked by closed condensate polisher isolation valves and an inoperable bypass valve (Ref. A, p. 25).

TIME		EVENT/OPERATOR ACTION	REMARKS
00:05:30 (0406:07) Approximate		The indicated reactor coolant system hotleg temperature and pressure reached saturation conditions of 582°F and 1340 psig.	
00:05:51 (0407:31)		Pressurizer coolant level indication went off scale high, greater than 400 in.	During this period, the unanticipated pressurizer level response caused the operators to question the accuracy of level indication (Ref. K, p. 30).
00:06:54 (0408:34)		A letdown cooler 1A high temperature alarm was received. (Nominal alarm setpoint was 135°F.)	Letdown flow isolated on high temperatures. This alarm could be a result of a high letdown flow condition.
00:08:18 (0408:55)	*	Emergency feedwater block valves 12A & 12B (EF-V12A and EF-V12B) were opened.	Operators verifying required system lineup discovered these normally open valves were shut. This action established feed flow to the steam generators.
00:08:58 (0409:35)		Condensate pump 1A (CO-P-1A) tripped.	
00:09:13 (0409:50)		A condensate booster pump suction header low pressure alarm was received.	
00:09:23 (0410:00) Approximate	*	The letdown flow isolation valve was opened.	The operator reestablished letdown flow.
00:10:15 (0410:52)		Pressurizer coolant level indication came back on scale, less than 400 in.	

TIME		EVENT/OPERATOR ACTION	REMARKS
00:10:24 (0411:01) through 00:11:43 (0412:20)	*	Makeup pump 1A (MU-P-1A) tripped and was restarted twice.	An operator may have initially secured the makeup pump due to concern over the high pressurizer level condition, in accordance with the requirements of Ref. 14, which states "Do not start or continue to run a makeup pump with the Reactor Coolant System (Pressurizer) in a solid water condition."
00:10:48 (0411:25)		A reactor building sump high level alarm was received	Indicated level was 4.65 feet. This indication is displayed on a panel in the Auxiliary Building.
00:13:13 (0413:50)	*	Decay heat removal pumps 1A & 1B (DH-P-1A & DH-P-1B) were shut down.	During verification of ESF actuation, the operator stopped these pumps as reactor coolant system pressure was above that required to allow low pressure injection. (Ref. B, p. 17).

- Plant Status After 13 Minutes -

The plant had just experienced a total loss of feedwater (main and emergency feed), a reactor trip/turbine trip, an ESF actuation, and was undergoing an unrecognized small reactor coolant system leak. Saturated conditions existed in the reactor coolant system, and the pressurizer level indication was abnormally high. The operators were trying to recover from the total loss of feedwater by reestablishing steam generator levels, attempting to correct the high pressurizer level condition, and simultaneously responding to numerous alarms.

TIME		EVENT/OPERATOR ACTION	REMARKS
00:14:48 (0415:25)		The reactor coolant drain tank rupture disc (WDL-U26) failed at 191.6 psig.	This information, which could be indicative of a stuck open relief valve, was not readily available to the operator (Refer to 00:03:26 entry).
00:14:50 (0415:27)		At this time, reactor coolant pump related alarms began coming in on the alarm typewriter.	These alarms could indicate abnormal reactor coolant pump operation caused by degraded conditions in the reactor coolant system.

TIME		EVENT/OPERATOR ACTION	REMARKS
00:15:43 (0416:20)		The condensate booster pumps low discharge pressure alarm was received.	Nominal alarm setpoint was 310 psig.
00:16:12 (0416:49)		A condensate booster pump suction header low pressure alarm was received.	Nominal alarm setpoint was 15 psig. These last two events signify the suction flowpath was still blocked (Refer to 00:06:29 entry).
00:19:23 (0420:00)		Reactor Building Purge Air Exhaust Duct A Monitor (HP-R-225) particulate channel count rate increased from 1×10^2 to 5×10^2 cpm. Slight increases were also indicated on the Duct B Monitor (HP-R-226) and on HP-R-222 (before the filter) and HP-R-228 (after the filter).	These indications appear to be the result of the reactor coolant drain tank rupture disc blowout. These readings are below the alarm setpoints and most likely were not noticed by the operators.
00:20:00 (0420:37)		The indicated source range neutron flux signal departed from the expected normal flux decay for a reactor trip.	These indications could have prompted the following action.
00:22:17 (0422:54)	*	The operator depressed the reactor trip pushbutton.	This operator action could indicate that Emergency Procedure for Unanticpated Criticality (Ref. 5) was implemented. An auxiliary operator was subsequently instructed to investigate a possible demineralized (non-borated) water flow path into the makeup system (Ref. L, p. 29).
00:22:44 (0423:21)		Steam generator A water level increased to approximately 30 in. in the startup range. The low level alarm cleared.	A steam generator water level control set-point of 30 in. is applicable for existing plant conditions. Sometime after this the shift supervisor left the control room to investigate problems with the secondary system (Ref. A, p. 25) (Refer to 00:06:29 Entry).

TIME		EVENT/OPERATOR ACTION	REMARKS
00:24:58 (0425:35)	*	The ERV outlet temperature was 285.4°F. Safety valves RV1A and RV1B read 263.9°F and 275.1°F respectively.	This information was requested from the computer by the operator. Valve outlet temperatures in this range were expected by the operator following a valve operation. Because of the closed indication of the ERV and the relative proximity of the valve outlet temperatures, the operator could not conclude that the ERV had not reseated (Ref. D, p. 33).
00:25:44 (0426:21)	*	An emergency feedwater pump (EF-P-1) low discharge pressure alarm was received. This could indicate that EF-P-1 was shut down.	The cause of the loss of emergency feed had been corrected. Steam generator level had been restored to the normal control range per Reference 6, and efforts were being made to stabilize level in the steam generators. The two electric-driven emergency feedwater pumps remained running.
00:26:26 (0426:21) through 00:27:51 (0427:28)		RC Loop A outlet temperature 551.9°F RC Loop B outlet temperature 550.9°F RC Loop A inlet temperature 548.1°F RC Loop A inlet temperature 547.0°F RC Loop B inlet temperature 547.0°F RC Loop B inlet temperature 546.8°F RC Loop A wide range pressure 1040 psig. RC Loop B wide range pressure 1043 psig.	This was operator requested plant status information from the computer.
00:26:46 (0427:23)		Steam generator B water level increased to approximately 28 in. in the startup range. The low level alarm cleared.	

TIME		EVENT/OPERATOR ACTION	REMARKS
00:29:23 (0430:00) through 00:49:23 (0450:00) Approximate		Reactor Building Air Sample Monitor (HP-R-227) gas channel count rate increased from 1×10^3 to 5×10^4 cpm and then decreased to 1×10^3 cpm.	This information apparently wasn't in an alarm state. The operator's attention was focused on other plant evolutions.
00:32:23 (0433:00) Approximate		Radiation readings of the following monitors increased and then leveled off: gas channel of Station Vent (HP-R-219), gas iodine and particulate channels of Fuel Handling Building Exhaust Duct (HP-R-221A, - 221B) H_2 Purge Duct particle and iodine channels (HP-R-229).	
00:36:08 (0436:08)	*	Emergency feedwater pump 2B (EF-P-2B) was shut down.	The operator was attempting to stabilize steam generator level in the normal control range per Reference 6. Steam generator A & B water levels indicated 35.3 and 40.3 in. in the startup range, respectively. One of the emergency pumps was still running, feeding both steam generators.
00:38:10 (0438:47)	*	Reactor building sump pump 2A (WDL-P-2A) was stopped.	
00:38:11 (0438:48)	*	Reactor building sump pump 2B (WDL-P-2B) was stopped.	All pumping of water from the reactor building sump to the auxiliary building was stopped because operators were previously aware that tank levels were high (Ref. O, p. 28).
00:40:00 (0440:37)		An increasing count rate continued to be indicated on the source range neutron instrumentation.	

TIME		EVENT/OPERATOR ACTION	REMARKS
00:46:23 (0447:00) Approximate		Letdown Cooler A Monitor (IC-R-1092) count rate began increasing from approximately 2000 cpm and reached over 2 x 10^4 cpm about 40 minutes later.	
00:59:12 (0459:49)		The condensate booster pump suction header low pressure alarm cleared.	The computer indication of condensate booster pump suction pressure was 89.2 psig. This occurred at the approximate time the condensate polisher bypass valve was opened by the operators. Sometime soon after this the shift supervisor returned to the control room.
00:59:21 (0459:58)		A condensate high temperature alarm was received.	The computer indication of condensate temperature was 118.5°F. This event also indicates flow through the condensate system.
01:00:49 (0501:26)	*	Condenser circulating water pumps 1B, 1C, 1D, & 1E (CW-P-1B, CW-P-1C, CW-P-1D, & CW-P-IE) were shut down.	Steam pressure control was shifted from the turbine bypass valves to the atmospheric dump valves to prevent the buildup of condenser hotwell water inventory (Ref. C, p. 20).
01:13:29 (0514:06)	*	Reactor coolant pump 2B (RC-P-2B) was stopped.	Based on the reactor coolant system pressure-temperature limitiations in the Operating Procedures for Reactor Coolant Pumps (Ref. 17) and Limits and Precautions (Ref. 15), the operator responded by stopping 1B and 2B reactor coolant pumps (Ref. J, p 25 and Ref. D, p. 37). Operator action following RC pump trip is in accordance with Emergency Procedure for Loss of RC Flow/RC Pump Trip (Ref. 7).
01:13:42 (0514:19)	*	Reactor coolant pump 1B (RC-P-1B) was stopped	

Plant Status After One Hour and Fifteen Minutes

1. The pressurizer ERV remained open.

TIME	EVENT/OPERATOR ACTION	REMARKS

2. Reactor coolant flow stopped in the "B" loop and decreasing steadily in the "A" loop.

3. Steam generator "B" level increasing inexplicably and steam generator "A" level steady at 30 $\pm$ 10".

4. Reactor building pressure increasing steadily. Note: Operators have stated that the Reactor Building Cooling System had been placed in the emergency mode of operation manually, to reduce reactor building pressure (Ref. N, p. 10).

5. Reactor building sump over-flowing.

6. Increasing radiation levels and associated alarms at various locations in the plant.

7. Pressurizer level indicating at the high end of the scale and remaining relatively constant. Note: Operators believed the plant was solid with limited pressure control, and were letting down at approximately 100 gpm with heaters on in an effort to regain a steam bubble in the pressurizer (Refer to Appendix for Reactor Coolant Pressure Control System) (Ref. P, p. 12).

8. Reactor coolant pressure relatively constant in the range of 1000 psig.

9. Reactor shutdown neutron flux levels increasing.

10. Reactor coolant inlet and outlet temperatures indicating in the range of 550°F.

11. The procedures inferred to be in effect are:

1.	Unanticipated Criticality	Ref. 5
2.	Loss of RC. Flow/RC. Pump Trip	Ref. 7
3.	Loss of Reactor Coolant/Reactor System Pressure	Ref. 12
4.	Pressurizer System Failure	Ref. 10
5.	Response To Alarms	Ref. 16

TIME		EVENT/OPERATOR ACTION	REMARKS
01:20:31 (0521:08) through 01:20:58 (0521:58)		Pressurizer Relief and Safety Valve Temperatures ERV RC R2 - 283°F Safety valve RC R1A - 211°F Safety valve RC R1B - 218°F	This was an operator requested computer print-out of relief and safety valve outlet temperatures. Temperature of the ERV was essentially the same as the reading requested the previous hour. Temperatures of the safety valves had dropped approximately 50°F. The operators continued to believe these valves were closed (Ref. P, p.10) (Refer to 00:24:58 Entry).
01:30:00 (0530:37) Approximate		Reactor out-of-core intermediate range neutron instrumentation channel NI-3 came on scale and began increasing.	This response was consistent with the steadily increasing source range count rate.
01:30:40 (0531:17)	*	There was a marked increase in secondary side steam flow from steam generator A.	These and the subsequent secondary plant evolutions may have been caused by an operator adjusting turbine bypass valve position in an attempt to establish cooldown of the reactor coolant system.
01:31:22 (0531:59)	*	Secondary side steam flow from steam generator A decreased rapidly.	
01:32:04 (0532:41)	*	Feedwater flow to steam generator B was increased.	This served no apparent purpose since flow in the reactor coolant system loop B had already been stopped.
01:32:19 (0532:56		Steam generator A indicates dryout on the secondary side.	With flow in loop B stopped the resulting increase in steam generator A heat load contributed to its dryout.
01:34:10 (0534:47)	*	Feedwater flow to steam generator A was increased.	Feedwater flow was diverted to steam generator A to correct the dryout condition.
01:34:16 (0534:53)	*	Feedwater flow to steam generator B was reduced.	The events from 01:30:40 through 01:34:16 appear to indicate operator attempts to establish heat removal via a steam generator.

TIME		EVENT/OPERATOR ACTION	REMARKS
01:40:37 (0541:14)	*	Reactor coolant pump 2A (RC-P-2A) was stopped.	See Remarks at 01:13:29
01:40:45 (0541:22)	*	Reactor coolant pump 1A (RC-P-1A) was stopped.	See Remarks at 01:13:29. Emergency Procedure for Loss of Reactor Coolant Flow/Reactor Coolant Pump Trip (Ref. 7) directs plant operation (natural circulation) with all reactor coolant pumps tripped is controlled by the Emergency Procedure for Station Blackout (Ref. 8).
01:41:00 (0541:37) Approximate		Out-of-core neutron instrumentation indicated a decreasing flux level.	This temporary decrease was caused by a coolant phase separation with the liquid filling the downcomer annulus. The phase separation is attributed to stopping the reactor coolant pumps and the resultant flow coastdown.
01:42:00 (0542:37)	*	Steam generator B was isolated on the secondary side.	Steam Generator B was isolated because of indications of a secondary leak to containment (Ref. A, p. 31).
01:42:30 (0543:07) Approximate		Out-of-core nuclear instrumentation indicated increasing flux levels.	A conflict existed between actual indicated plant parameters and expected plant parameters. A 6% shutdown margin had been calculated. After the reactor trip, additions of makeup water to the reactor coolant system (RCS) were drawn from borated water sources of $\geq$ 1000 ppm. The RCS boron concentration was approximately 1000 ppm initially. Indicated shutdown flux levels were increasing and a 700 ppm analysis result was reported. Due to the above conditions, a backup sample was requested by the operators (Ref. A p. 60 Ref. H pp. 17 - 23).

TIME		EVENT/OPERATOR ACTION	REMARKS
01:51:27 (0552:04) through 02:29:18 (0604:34)		Loop A and B hot leg temperatures were increasing and continued upward until they went off scale high, greater than 620°F.	These temperatures and the reactor coolant system pressure indicate the presence of superheated steam in the reactor coolant system. With cold leg temperatures off scale low, the average reactor coolant temperature indicated 570°F. These are narrow range temperatures available to the operators on the main control console.
02:03:57 (0604:34)	*	An operating range water level of 50% was established and subsequently maintained in steam generator A.	Efforts were being made to establish natural circulation using the steam generator in accordance with Emergency Procedure for Loss of RC Flow/RC Pump Trip (Ref. 7) (Ref. D, p. 34 and Ref. C, p. 12).
02:14:23 (0615:00) Approximate		The Reactor Building Air Sample Monitor (HP-R-227) particulate channel radiation reading increased and eventually went off scale high. The gas channel radiation reading began to increase at 0625 and then went off scale high. Iodine channel count rate began to increase about 0645 and went off scale high.	
02:17:53 (0618:30)	*	ERV (RC-R2) outlet temperature was 228.7°F.	The ERV outlet temperature was operator requested. The indicated temperature at this time is more than 50°F lower than the reading one hour previous.
02:22:00 (0622:37) Approximate	*	The ERV block valve (RC-V2) was closed. Reactor coolant system pressure began to increase.	The source of the reactor coolant system leakage was corrected (Ref. N, p. 9).
02:33:27 (0634:04)	*	The operator commenced feeding steam generator B.	Steam Generator B was unisolated and filled to the natural circulation level required by Ref. 7.

TIME		EVENT/OPERATOR ACTION	REMARKS
02:34:23 (0635:00) Approximate	*	An additional makeup pump was started.	Makeup pump 1C was started. It appears this action was taken to hasten pressure recovery to enable reactor coolant pump operation in order to reestablish forced circulation.
02:38:23 (0639:00) Approximate		The Makeup Tank Area Monitor (HP-R-206) in the auxiliary building, Fuel Handling Bridge South Monitor (HP-R-210) in the reactor building, and Reactor Building Dome Monitor (HP-R-214) radiation reading began to increase.	
02:38:23 (0639:00) Approximate		Letdown Cooler A Radioactivity Monitor (IC-R-1092) readings pegged off scale high.	
02:39:23 (0640:00) Approximate	*	Two boron analyses indicating boron concentrations in the reactor coolant system of approximately 400 ppm were received by the shift supervisor. Emergency boration of the reactor coolant system was started.	The boron sample analysis results were alarmingly low. Emergency boration is a requirement of Abnormal Procedure for Loss of Boron (Moderator Dilution) (Ref. 11). Additional efforts were instituted by the operators to locate possible sources of demineralized water entry points (Ref. H, p. 22).

- Plant Status After 2 Hours and Forty Minutes -

1. The leaking pressurizer ERV had been isolated by closing the isolation valve.
2. Forced flow stopped in both reactor coolant loops.
3. Steam generator A level in the range for natural circulation.
4. Reactor coolant outlet temperatures off scale (>620°F) increasing; inlet temperatures at approximately 450°F and decreasing.
5. Radiation alarms and increasing levels at various locations in the plant.

TIME	EVENT/OPERATOR ACTION	REMARKS

6. Pressurizer coolant level indicating steady at 300 inches.

7. Reactor coolant pressure in the range of 1000 psig and increasing.

8. Reactor shutdown neutron flux indicating steady at abnormally high levels.

9. Reactor building pressure decreasing.

10. The procedures inferred to be in effect are:

1.	Unanticipated Criticality	Ref. 5
2.	Loss of RC. Flow/RC. Pump Trip	Ref. 7
3.	Station Blackout	Ref. 8
4.	Loss of Reactor Coolant/Reactor System Pressure	Ref. 12
5.	Pressurizer System Failure	Ref. 10
6.	OTSG Tube Rupture	Ref. 9
7.	Loss of Boron (Moderator Dilution)	Ref. 11
8.	Response To Alarms	Ref. 16

TIME		EVENT/OPERATOR ACTION	REMARKS
02:46:00 (0645:37) Approximate	*	MU-P-1C was stopped.	It appears the operator stopped the pump as system pressure had recovered sufficiently to allow a reactor coolant pump to be started.
02:46:23 (0647:00)	*	The operator attempted to start reactor coolant pump 1A.	The pump did not start.
02:51:57 (0652:34)	*	The operator attempted to start reactor coolant pump 2A.	The pump did not start.

TIME		EVENT/OPERATOR ACTION	REMARKS
02:53:16 (0653:53)	*	The operator attempted to start reactor coolant pump 1B.	The pump did not start.
02:54:09 (0654:46)	*	The operator started reactor coolant pump 2B.	Flow was indicated for only a few seconds and returned to zero. The pump was subsequently stopped.

Post Two Hour and Fifty Five Minute Narrative

At time two hours and fifty-five minutes, plant conditions had deteriorated to the point that plant operations were beyond the intended scope of the station emergency procedures. Attempts to reestablish forced circulation with the reactor coolant pumps had failed. An evaluation made at this time concluded large voids existed in the reactor coolant system hot legs (Ref. F p. 23, Ref. G p. 25) and boiling was taking place in the core (Ref. H p. 11). Prompted by this evaluation, a decision was made to pump into the reactor cooling system using high pressure injection and to vent fluids out of the system through the electromatic relief valve (feed and bleed) to establish a new mode of core cooling (Ref. H p. 12, Ref. I p. 14). Subsequently, this approach was modified by cycling the ERV block valve to maintain reactor coolant system pressure at approximately 2000 psig in an attempt to remove the voids from the system. Concurrently, high radiation conditions within the plant prompted the implementation of site and general emergency plans. In addition, steam generator B was isolated in response to increased radiation levels indicated on the condenser vacuum pump exump into the reactor cooling system using high pressure injection and to vent fluids out of the system through the electromatic relief valve (feed and bleed) to establish a new mode of core cooling (Ref. H p. 12, Ref. I p. 14). Subsequently, this approach was modified by cycling the ERV block valve to maintain reactor coolant system pressure at approximately 2000 psig in an attempt to remove the voids from the system. Concurrently, high radiation conditions within the plant prompted the implementation of site and general emergency plans. In addition, steam generator B was isolated in response to increased radiation levels indicated on the condenser vacuum pump exhaust monitor (Ref. N, p. 9). This action implies the use of the Emergency Procedure for OTSG Tube Rupture (Ref. 9).

With no indication of this cooling mode being effective and concerned that a "solid" system (natural circulation) could not be attained, the plant operating staff discussed the relative merits of heat removal by either of the following methods:

1) Heat removal by forced circulation using the reactor coolant pumps;

2) Heat removal by using the decay heat removal system.

Because of the concern for ERV block valve failure resulting in loss of pressure control, a decision was made to depressurize the primary plant. During depressurization, core flood system injection would be used to confirm the core was covered (Ref. J pp. 61-65). Therefore, depressurization of the reactor coolant system was started. Noting a small injection of core flood water, the interpretation was made by the operating staff that the core was covered. Efforts to further reduce system pressure was unsuccessful.

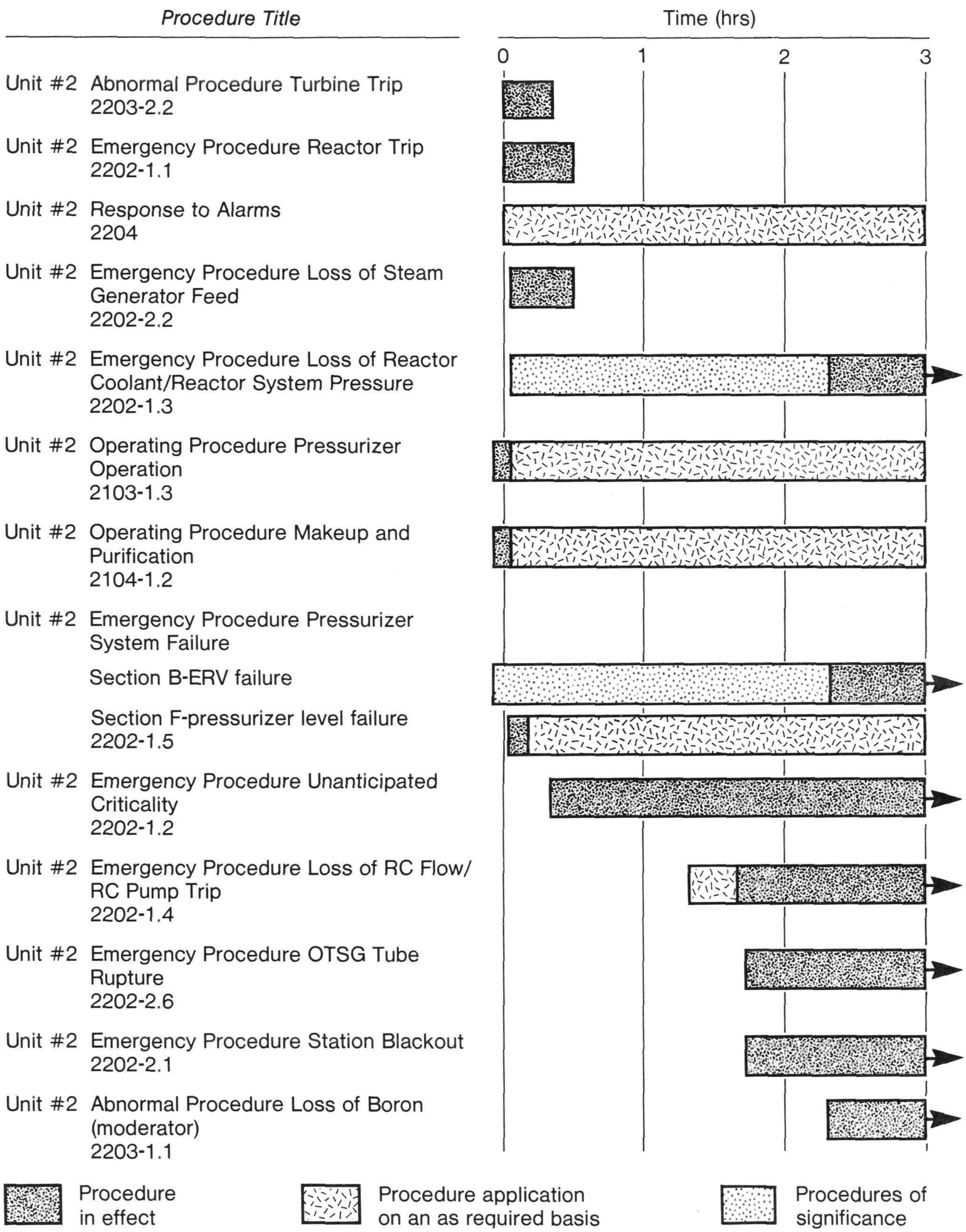
Subsequently, the decision was made to repressurize the system and to restart a reactor coolant pump (Ref. J p. 72, Ref. I p. 30). Reactor coolant system pressure was increased using an additional high pressure injection makeup pump. Upon attaining a system pressure suitable for reactor coolant pump operation, a pump was started and core cooling by forced circulation was again established which provided a stable mode of decay heat removal.

ATTACHMENT 1

Reference 1	NRC Regulatory Guide 1.33	
Reference 2	TMI Technical Specification 6.8	
Reference 3	Unit #2 Emergency Procedure Reactor Trip	2202-1.1
Reference 4	Unit #2 Abnormal Procedure Turbine Trip	2203-2.2
Reference 5	Unit #2 Emergency Procedure Unanticipated Criticality	2202-1.2
Reference 6	Unit #2 Emergency Procedure Loss of Steam Generator Feed	2202-2.2
Reference 7	Unit #2 Emergency Procedure Loss of RC Flow/RC Pump Trip	2202-1.4
Reference 8	Unit #2 Emergency Procedure Station Blackout	2202-2.1
Reference 9	Unit #2 Emergency Procedure OTSG Tube Rupture	2202-2.6
Reference 10	Unit #2 Emergency Procedure Pressurizer System Failure Section B ERV Failure Section F Malfunction in Pressurizer Level Indication or Control	2202-1.5
Reference 11	Unit #2 Abnormal Procedure Loss of Boron (Moderator Dilution)	2203-1.1
Reference 12	Unit #2 Emergency Procedure Loss of Reactor Coolant/Reactor System Pressure	2202-1.3
Reference 13	Unit #2 Operating Procedure Pressurizer Operation	2103-1.3
Reference 14	Unit #2 Operating Procedure Makeup & Purification	2104-1.2
Reference 15	Unit #2 Operating Procedure Nuclear Plant Limits and Precautions - (Babcock & Wilcox Technical Document #67-1002171-00)	2101-1.1

Reference 16	Unit #2 Response to Alarms	2204
Reference 17	Unit #2 Operating Procedure Reactor Coolant Pump	2103-1.4

Attachment II



ATTACHMENT III

Reference A..............................NRC-00-0084

Reference B..............................NRC-00-0094

Reference C..............................NRC-00-0138

Reference D..............................NRC-00-0173

Reference E..............................NRC-00-0088

Reference F..............................NRC-00-0083

Reference G..............................NRC-00-0148

Reference H..............................NRC-00-0276

Reference I..............................NRC-00-0106

Reference J..............................NRC-00-0149

Reference K..............................NRC-00-0161

Reference L..............................NRC-00-0089

Reference M..............................NRC-00-0116

Reference N..............................NRC-00-0183

Reference O..............................NRC-00-0224

Reference P..............................NRC-00-0107

APPENDIX OTSG

ONCE THROUGH STEAM GENERATOR (OTSG)

1. INTRODUCTION

The nuclear steam supply system at TMI-2 employs two once-through steam generators (OTSGs) for transferring heat from the reactor coolant system to the secondary system. The OTSG (Figure OTSG-1) is a vertical, straight tube and shell boiler in which the reactor coolant (the heat source) is on the tube side and the secondary coolant is on the shell side. The OTSG produces superheated steam which is controlled at a constant main steam pressure throughout the power operating range. The tubes and tubesheets form the boundary between the radioactive reactor coolant and the non-radioactive steam and feedwater.

Figure OTSG-1 shows the flow paths of the reactor coolant water and the feedwater through the OTSG. Reactor coolant water enters the steam generator at the upper plenum, flows down inside the tubes while transferring heat to the secondary shell-side fluid, and exits through the lower plenum. Feedwater enters through the side of the steam generator, near the middle, travels down the downcomer, the up along the outside of the tubes, and exits the steam generator as superheated steam. This flow path carries the feedwater through four heat transfer regions. Starting with the feedwater inlet, these are:

- Feedwater Heating - Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into a feedheating annulus (downcomer) formed by the shell and the baffle around the tube bundle. The steam that heats the feedwater to saturation is drawn into the top of the downcomer by the condensing action of the relatively cold feedwater.

- Nucleate Boiling - The saturated water enters the tube bundle, and the steam-water mixture flows upward on the outside of the tubes counter-current to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until departure from nucleate boiling (DNB) is reached; then film boiling and superheating occur. The quality at which the transition from nucleate boiling to film boiling occurs is a function of pressure, heat flux, and mass velocity.

- Film Boiling - Dry saturated steam is produced in the film boiling region near the middle of the tube bundle.

- Superheated Steam - Saturated steam is raised to final temperature in the superheater region, and leaves the unit as superheated steam.

The heat transfer to the secondary system is controlled by the rate of feedwater introduction to the OTSG. This also controls the area of the total tube bundle length that is exposed to liquid and low quality secondary coolant for a given reactor power level. Therefore increasing feedwater flow causes an increase in heat transfer and decreasing feedwater flow decreases heat transfer. The inherent feature of the design is high responsiveness to feedwater control which makes possible an accuracy of control that has operational and safety advantages, such as being able to ride through many secondary upsets without tripping the reactor. This increases plant availability and provides fewer challenges to the safety system.

2. WATER LEVEL INDICATION IN THE OTSG

There are three ranges of water level indication in the OTSG (Figure OTSG-2). These ranges provide indication of equivalent liquid water level over the length of the shell side of the OTSG. These indication ranges are:

- <u>Full Range</u> - (0-600") - Indication is obtained from one transmitter in each generator and is read in the control room on one indicator per generator.

- <u>Start-up Range</u> - (0-250") - Indication comes from two transmitters per generator and is read in the control room on two dual indicators. The upper tap for this instrument is located at 308 inches above the reference plane shown in Figure OTSG-2; the 250 inches is equivalent liquid water level only.

- <u>Operating Range</u> - (0-100%) - Indication is obtained from two transmitters in each generator, is temperature compensated, and is read in the control room on two dual recorders. This instrument has a common upper tap with the start-up range instrument.

As shown in Figure OTSG-2, a partial overlap exists between these operating and start-up level indications. When comparing these two level signals, temperature compensation of the operating range must be taken into account. Due to this compensation, a direct linear comparison of the overlapping regions cannot be made. At normal system temperature and pressure in the OTSG, 100 inches on the start-up range correlates to approximately 5 to 6% on the operating range.

The start-up and operating ranges are used when taking the plant from a shutdown condition to a full power operation. The full range is used primarily when the plant is being shut down and for long term shutdown conditions such as dry layup (OTSG dry) or wet layup (OTSG solid).

3. FEEDWATER SUPPLIES TO THE OTSG

Feedwater is supplied to the OTSG from the feed, condensate, and emergency feedwater systems. Depending on steam pressure in the steam generator, combinations of the condensate, condensate booster, and main feed pumps, are used to maintain water level in the OTSG. In the event of loss of the normal feedwater supply methods, emergency feedwater from three pumps can be supplied to the OTSG through the auxiliary feed nozzle.

4. ISOLATION OF THE FEEDWATER SUPPLY

Isolation of the secondary side of a steam generator includes isolating the feedwater inlet and the steam exit flow paths (see also Figure C/FWD-1). Isolation of the feedwater is accomplished by closing the valves which supply feedwater to the main and auxiliary feedwater nozzles of the steam generator. The steam flow paths are isolated by closing the main steam isolation valves to the main turbine and the turbine bypass valves. Overpressure protection of the OTSG after isolation is still provided by the atmospheric dump valve and the main steam safety valves.

5. NATURAL CIRCULATION CAPABILITY OF THE OTSG

Eight events have taken place (two were unintentional) that demonstrate the natural circulation ability of the OTSG in loop configuration identical to that at TMI-2. These tests, and eight others on a somewhat different loop arrangement, have demonstrated that natural circulation is quickly established at flow rates greatly in excess of that needed to remove the heat being generated in the reactor. At typical decay heat levels (of about 1% of rated power) the flow is about four times that required. At higher power levels this ratio is somewhat less. A comparison of the percent of design flow to the percent of design power shows ratios of 1.4 to 2.3 for the eight tests applicable to TMI-2. The results are independent of secondary water level in the steam generator within the startup range.

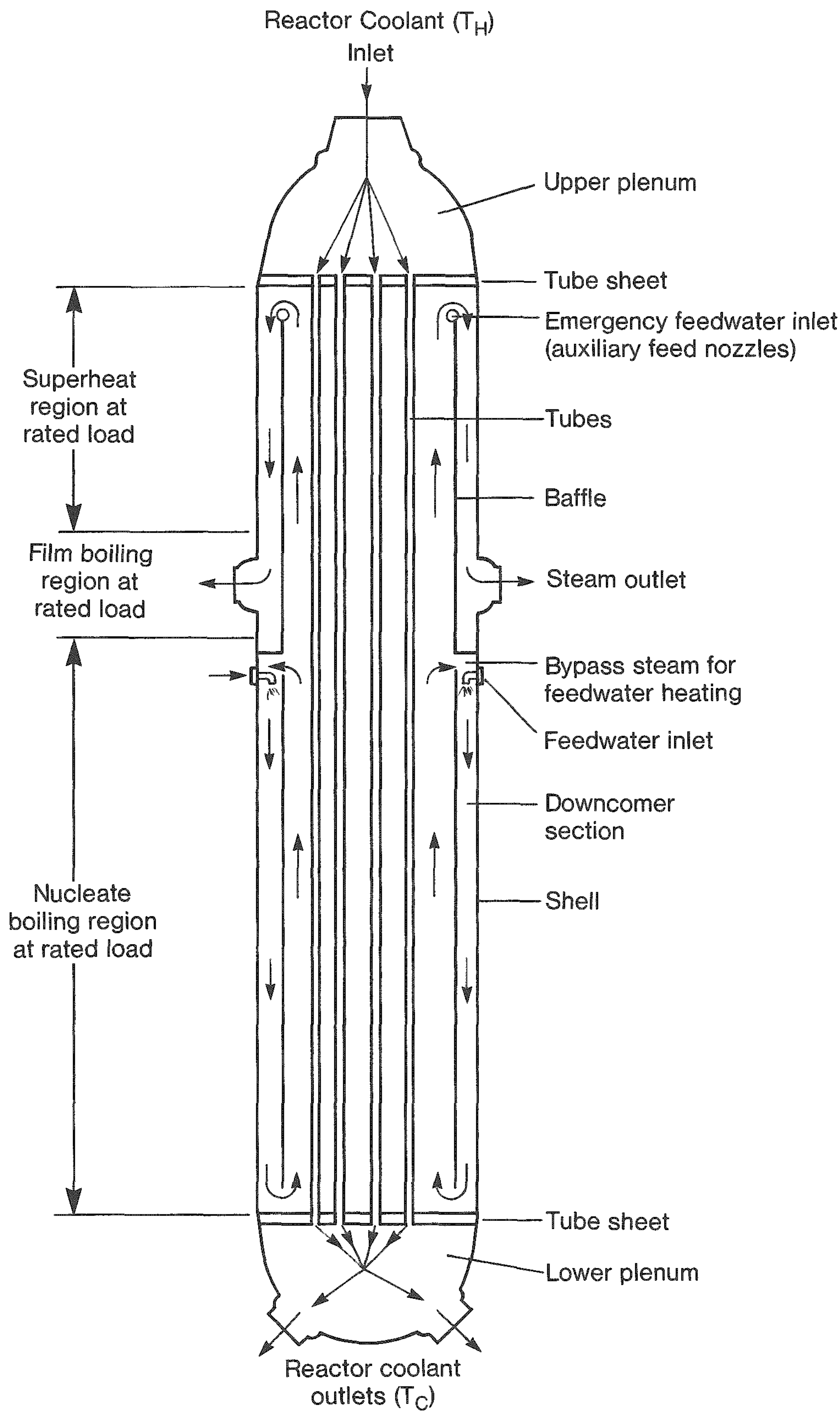


Figure OTSG-1.

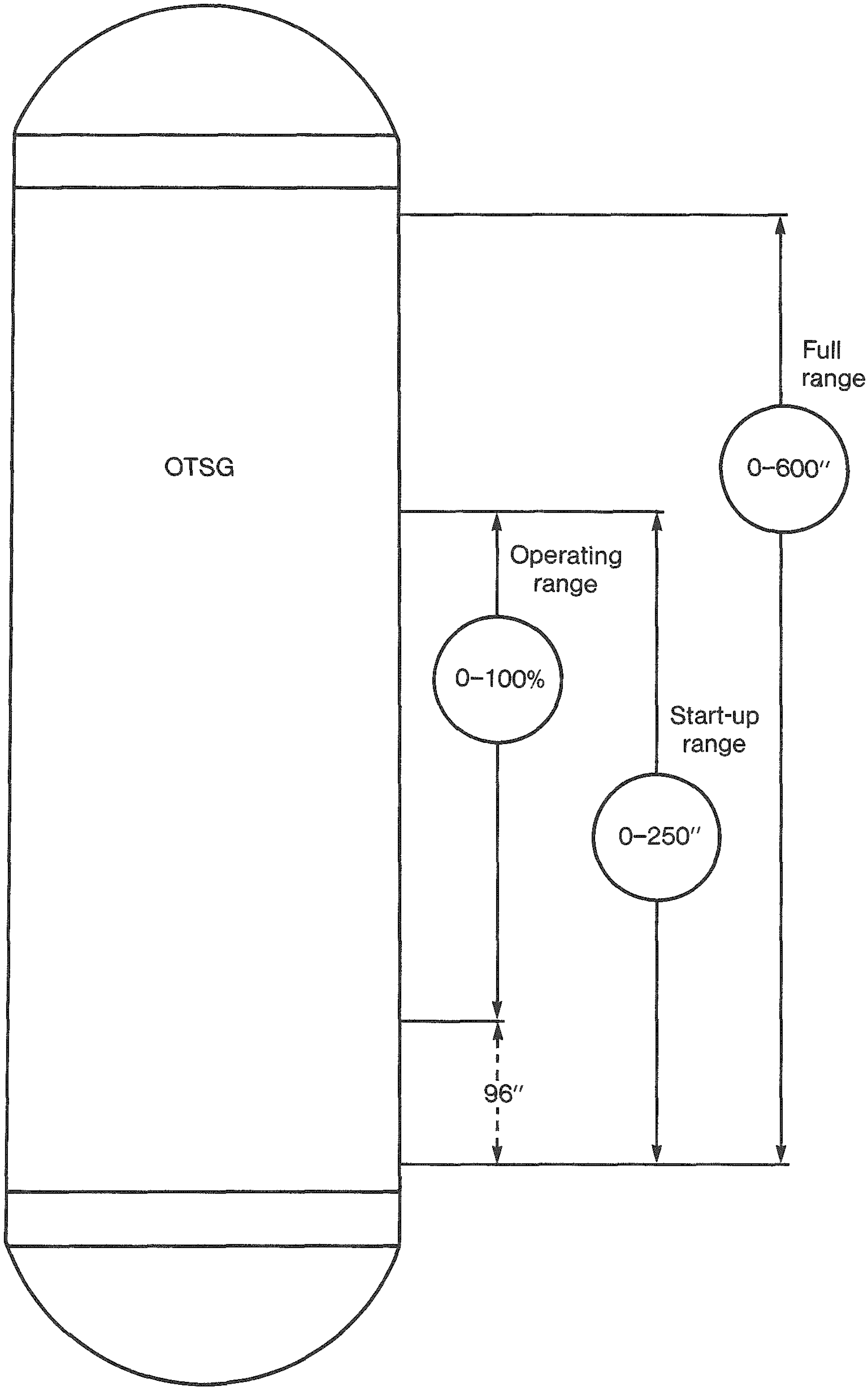


Figure OTSG-2.
OTSG Level Indication

APPENDIX PDS

PLANT DATA SOURCES

1. REACTIMETER

The reactimeter is a 24 channel data acquisition system provided by Babcock & Wilcox. Its name derives from its capability to record reactor core reactivity. This function is normally used only during reactor start-up testing. The reactimeter also records other data, and it was the availability of these data that made the reactimeter particularly valuable at TMI-2.

The 24 channels of data are recorded on magnetic tape in the form of voltage readings. These voltages are directly proportional to the parameter being monitored, e.g., pressure, temperature, flow, etc. The parameters which were being monitored at the time of the accident are listed in Table PDS-1. The data signals originate from the same detectors that provide signals for normal plant monitoring and safety systems actuation.

The reactimeter can sample each channel on any time interval from 0.2 second to 12.6 seconds. At the time of the accident it was set to sample each channel on a 3 second interval. It samples all 24 channels in 1.6 milliseconds, which is essentially simultaneously.

The reactimeter is physically located in the Unit 2 Cable Spreading Room. The only attention it normally requires is changing the magnetic tape about every 26 hours.

TABLE PDS-1

REACTIMETER LOGGED PARAMETERS

Channel	
1	Power range level--nuclear instrument-5 (0-125%)
2	Loop A hot leg temperature--narrow range (520-620°F)
3	Loop B hot leg temperature--narrow range (520-620°F)
4	Loop A cold leg temperature--wide range (50-650°F)
5	Loop B cold leg temperature--wide range (50-650°F)
6	Loop A reactor coolant flow--temperature compensated (0-90 MPPH)*
7	Pressurizer level--temperature compensated (0-400 in.)
8	Makeup tank level (0-100 in.)
9	Presurizer spray valve position (open-closed)
10	Drain tank presure (0-250 psig)
11	Loop B reactor coolant pressure--narrow range (1700-2500 psig)
12	Reactor trip (run-trip)
13	Loop B reactor coolant flow--temperature compensated (0-90 MPPH)*
14	Feedwater temperature (0-500°F)
15	Turbine header pressure--Loop A (600-1200 psig)
16	Steam generator A operate level--temperature compensated (0-100%)
17	Steam generator A start-up level (0-250 in.)
18	Feedwater flow--Loop A (0-6500 KPPH)†
19	Feedwater flow--Loop B (0-6500 KPPH)†
20	Turbine trip (run-trip)
21	Steam generator A steam pressure (0-1200 psig)
22	Steam generator B steam pressure (0-1200 psig)
23	Steam generator B operate level--temperature compensated (0-100%)
24	Steam generator B start-up level (0-250 in.)

*MPPH: Million pounds per hour.
†KPPH: Thousand pounds per hour.

The magnetic tapes that are produced by the reactimeter can be directly read by a computer which converts the voltage readings into engineering units and displays the data in the form of tables or graphs. Because of the accurate, continuous, retrievable nature of the reactimeter data, it is considered to be the most reliable data available on the parameters it was monitoring. For this reason, it is being used as a reference baseline against which to measure the accuracy of other data sources pertaining to the TMI-2 accident.

2. CONTROL ROOM STRIP CHARTS

Many of the primary and secondary plant parameters are continuously recorded on strip chart recorders located in the control room. The parameters monitored by control room strip chart recorders along with the type and panel location of the recorders are listed in Table PDS-2. Figure PDS-1 shows the panel layout in the control room. These recorders allow the operators to observe trends in the monitored parameters and they create a historical record of the trends.

There are basically two types of recorders used in the control room--pen recorders which employs an ink pen to produce a continuous line plot of the parameter's value and the multipoint recorder which monitors several parameters and prints a code number identifying each parameter, as it is scanned. The code number is printed at a location on the strip chart representing the parameter's value. Figure PDS-2 shows examples of both types of recorder traces.

Legibility is normally the biggest problem encountered in trying to extract information from strip charts. This is especially true of the multipoint recorders, when several parameter traces are printed on top of each other, and when the printed numbers are not readable. The problem of legibility is compounded by the slow

TABLE PDS-2

CONTROL ROOM STRIPCHART RECORDERS

Parameter(s) Recorded	Units	Recorder Number & Type	Recorder Location
Any 4 analog inputs to plant computer		4-single pens	Panel 2
Reactor building pressure	psig	2-dual pens	Panel 3
Makeup tank level	in.	1-single pen	Panel 3
Pressurizer level	in.	1-single pen	Panel 4
Loop A wide range pressure	psig	1-single pen	Panel 4
Loop A narrow range pressure	psig	1-single pen	Panel 4
Loop B narrow range pressure	psig	1-single pen	Panel 4
Reactor coolant outlet temperature	°F	1-single pen	Panel 4
Reactor coolant average temperature	°F	1-single pen	Panel 4
Source & intermediate range power level	CPS & AMP	1-dual pen	Panel 4
Intermediate range power level	AMP	1-single pen	Panel 4
Power range level	percent	1-single pen	Panel 4
Selected turbine header pressure	psig	1-single pen	Panel 4
Steam generator A operate level	in.	1-single pen	Panel 4
Steam generator B operate level	in.	1-single pen	Panel 4
Steam generator A feedwater flow	KLB/HR	1-single pen	Panel 4
Steam generator B feedwater flow	KLB/HR	1-single pen	Panel 4
Steam generator A & B operate level	in.	1-dual pen	Panel 5
Liquid waste discharge ΔT above river temperature	°F	1-dual pen	Panel 8
Cooling tower makeup water flow	GPM	1-single pen	Panel 8
Reactor coolant pumps seal cavity pressure	psig	1-multipoint	Panel 8
Transfer flow from reactor coolant drain tank	GPM	1-single pen	Panel 8A
Wind speed and direction	MPH & DEG	1-dual pen	Panel 10
Outside air temperature and ΔT at different elevations	°F	1-multipoint	Panel 10
Control rod drive motor temperature	°F	1-multipoint	Panel 10
Valve stem leakage thermocouples	°F	1-multipoint	Panel 10
Turbine generator temperatures	°F	1-multipoint	Panel [illegible]

TABLE PDS-2

(continued)

Reactor coolant & S/G temperatures	°F	1-multipoint	Panel 10
Radiation monitoring system	MR/HR & CPM	6-multipoints	Panel 12
Self powered neutron detectors backup	NANOAMPS	2-multipoints	Panel 14
Main tubine governor valve position	percent	1-single pen	Panel 16
Main turbine vibration	MILS	1-multipoint	Panel 16
Main turbine casing temperatures	°F	1-multipoint	Panel 16
Condenser vacuum	in. HG	1-dual pen	Panel 17
Condenser circulating water temperatures	°F	2-dual pens	Panel 17
Main feedwater pumps speed and turbine governor valve position	RPM and percent	2-dual pens	Panel 17
Main feedwater pump turbine vibration	MILS	2-dual pens	Panel 17
Main turbine header pressure	psig	2-dual pens	Panel 17
Reactor building temperatures	°F	1-multipoint	Panel 25
Reactor building ventilation flows	CFM	2-dual pens	Panel 25
Auxiliary building exhaust ventilation	CFM	1-dual pen	Panel 25
Auxiliary building supply ventilation	CFM	1-single pen	Panel 25
Fuel handling building exhaust ventilation	CFM	1-dual pen	Panel 25
Fuel handling building supply ventilation	CFM	1-single pen	Panel 25
Control building ventilation flows	CFM	1-dual pen	Panel 25
Auxiliary building temperatures	°F	1-multipoint	Panel 25
Fuel handling building temperatures	°F	1-multipoint	Panel 25
Control building temperatures	°F	1-multipoint	Panel 25

speed at which the strip charts travel (normally 1 inch/hour for the pen plotters). A large amount of data is compressed into a small linear space. Also, if the strip charts are not properly annotated when removed from the recorder, problems occur in recovering the time frame of the plots.

Differences between strip chart and reactimeter values for the same parameter indicate that strip chart data is generally less accurate than reactimeter data. However, the strip charts are calibrated periodically and have acceptable accuracy for most purposes -- especially as a source of trend information.

3. OPERATOR INTERVIEWS

On March 30, 1979, GPU conducted the first interviews of individuals who had been involved in the accident. The following categories of personnel were interviewed:

- The Unit 2 operating staff on duty at the time the accident occurred.
- Station management involved with the conduct of the Emergency Plan.
- Others who could potentially contribute information on the events during the accident.

Table PDS-3 is a list of the interviews by job position along with dates they were conducted. Table PDS-4 shows the "Interview Questions" that were used as a guide in conducting the interviews.

The operator interviews provide valuable insight on the operators' thought processes during the course of the accident, i.e., why certain actions were taken, what was perceived about the condition of the plant from the instrumentation, what tended to confuse or complicate the situation, etc. The interviews are of little use in substantiating the time intervals between events

TABLE PDS-3

TMI PERSONNEL INTERVIEWS

Individual Interviews Position	Date
Unit 2 control room operators (A, B)*	March 30, 1979
Unit 2 auxiliary operators (A, B, C, D)	March 30, 1979
Unit 2 shift foreman (A)	March 30, 1979
Station shift supervisor (A)	March 30, 1979
Unit 2 control room operators (A, B)	April 6, 1979
Station shift supervisor (A)	April 6, 1979
Unit 2 operations supervisor	April 20, 1979
B & W representative (A)	April 20, 1979
Station shift supervisor (B)	April 20, 1979
Station health physics foreman	April 20, 1979
Station shift supervisor (C)	April 20, 1979
Control room operator (C)	May 4, 1979
Group Interview	
Station health physics supervisor	April 12, 1979
Station manager	April 12, 1979
Unit 1 superintendent	April 12, 1979

TABLE PDS-3

TMI PERSONNEL INTERVIEWS (continued)

Group Discussion	
Station health physics supervisor	April 14, 1979
Unit 2 superintendent--technical support	April 14, 1979
Unit 2 superintendent	April 14, 1979
Station manager	April 14, 1979
B & W representative (B)	April 14, 1979
Unit 1 operations supervisor	April 14, 1979
Unit 1 superintendent	April 14, 1979
Statin shift supervisor (A)	April 14, 1979
Dictated Statements	
Control room operators (A, B, C)	March 29, 1979
Notes Recorded During Accident	
Unit 2 operations engineer	1315, March 28, 1979 to to 0320, March 29, 1979

*Letters distinguish individuals in positions which have more than one person with the same title.

TABLE PDS-4

INTERVIEW QUESTIONS

1. Date/time/job function.

   Location 1 hour before and during the event.

   Duration of participation (time entered site, left site, etc.)

2. Chain of command in job function during event. Where did orders come from and go to?

3. OBSERVATIONS

   Recollection of sequence of events

   - Instruments directly observed, how reliable? Cross checked?
   - Documentation developed (formal or informal roles, calculations, logs, etc.)
   - Control functions initiated by operator.
   - References used in event (i.e., technical specifications, procedures, etc.)
   - Reference material needed but not available.
   - Hardware needed but not available.
   - Direction received or given.
   - Directly observed malfunctions of equipment or procedure.
   - Recommendations received from outside organizations (state, federal or other).
   - Normal and abnormal plant conditions observed.

   Radiation Reports/Spread of Contamination

   - Sequence of radiation event--when, where, how high?
   - Data collected on sources--internal and external.
   - Who collected--how, when, and where analyzed?
   - Primary and secondary chemistry data history.

TABLE PDS-4

INTERVIEW QUESTIONS (continued)

Radiation Reports/Spread of Contamination (continued)

- o Exposure records of all involved.
- o Collection of off-site dosimetry.
- o Area meterology.
- o History of all off-site notification.
- o Data on off-site HP mobilization.

Subjective Data

- o Personal concerns with time or direction given.
- o Impressions of state of plant at various times (i.e, relative hazard, danger, actions being taken).
- o Recommendations offered during course of event. To whom were they made? What was the basis for the recommendation?
- o Adequacy of communication on and off-site.

or actions because the operator's sense of time became very distorted during the course of the accident. The early interviews were more informative than the ones at later dates. It appears that those interviewed had begun to forget some of the details in the later interviews and that their thinking was becoming colored by discussions with others, and by information gained from review of the accident data. Even in the initial interviews, recall seemed to diminish as the accident lengthened. The information from the interviews regarding the first three hours of the accident is, therefore, more complete than that relative to the later hours. During April and May, 1979 the NRC conducted approximately 200 interviews of GPU or MetEd staff personnel. The interviews covered a broad spectrum of technical disciplines within the utility organization.

4. PLANT COMPUTER

The plant computer system at TMI-2 utilizes a Bailey 855 computer linked with a smaller NOVA computer to form one integral system. The NOVA computer is an addition made by Metropolitan Edison Company to provide more capacity for balance-of-plant monitoring. The principal function of the computer system is to monitor plant parameters (approximately 3000) and to display them along with any related calculations. The parameter input signals are either analog or digital.

In performing its monitor function, the computer scans 960 digital and 80 analog inputs every second. An analog parameter may be scanned on 1, 5, 15, 30, or 60 second intervals depending on its relative importance. Each second the computer scans all the 1 second scan points, 1/5 of the 5 second scan points, 1/15 of the 15 scan points, and so on.

The computer has two output modes for the points it scans--an alarm printer and a utility printer. These are both automatic typewriters, and if either fails its output is automatically transferred to the other. A small cathode ray tube display is also provided which duplicates the output of the printers.

Alarm Printer

For all monitored parameters that have an alarm function, the alarm printer automatically prints an alarm message when the parameter has gone into an alarm condition, i.e., exceeded an alarm setpoint or changed state. Figure PDS-3 shows an excerpt from the alarm printer output.

The printed alarm time is the real clock time when the computer scanned the parameter and found it in an alarm condition. Note, that a parameter on a 60 second scan rate which exceeds its alarm setpoint immediately after a scan, will be in the alarm condition for 60 seconds before the computer records the alarm. If a parameter were to exceed its alarm setpoint and then return within the setpoint between two consecutive scans, the computer would not record the alarm condition.

The alarm inputs are stored by the computer in an alarm backup buffer until they are printed. This buffer can store up to 1365 alarm inputs before it is filled. The alarm printer can only print one alarm every 4.2 seconds. If alarms are occurring at a faster rate, the printer gets further and further behind, and the alarms may be printed minutes after they are recorded. (At one point during the TMI-2 accident the alarm printer was at least 161 minutes behind.) After the buffer is filled (i.e., 1365 alarms are waiting to be printed) the computer program is designed to print the message "Alarm Monitor Holdup" indicating that future alarms will not be stored until some of the 1365 backlogged alarms are printed. These unstored alarms will never be printed. The operator does have the option of suppressing the alarm sequence. This erases all old alarms from the computer memory and causes it to start printing new alarms which originated after the suppression.

Utility Printer

The utility printer provides output on request. The value or condition of any monitored parameter can be requested. Special subroutines allow the operator to request output values in specific preprogrammed groups called "Operator Special Summaries" (Figure PDS-4) or to trend output values in preprogrammed groups called "Operator Group Trends" (Figure PDS-5).

The computer is also programmed to record automatically all changes in state of a predesignated group of parameters called "Sequence of Events" inputs. These event inputs are stored in the computer and can be printed on request as shown in Figure PDS-6. This particular computer function does not use the scan process described above, but uses a continuous monitoring process which enables it to print the exact time that the "Sequence of Events" inputs occurred. The sequence is started by any one of the "Sequence of Events" inputs changing state and continues until printed by the operator.

Another feature programmed into the computer is the "Memory Trip Review." Triggered by a reactor or turbine trip, this routine records a set of predesignated parameter inputs for 15 minutes before and 15 minutes after the trip. This information is stored until the operator requests that it be printed. Figure PDS-7 is an example of the "Memory Trip Review" printout.

The plant computer provides the operator with an efficient means of keeping logs and showing trends on a large number of plant parameters under normal operating conditions. The computer was not designed to accommodate the data needs of the operator in an accident situation. Using the computer in an accident situation requires that the operator leave his control panels in order to request computer output; it takes the computer several seconds to supply the requested output; and, the automatic alarm printout is often several minutes behind real time. All of these tend to limit the computer's usefulness in an accident situation.

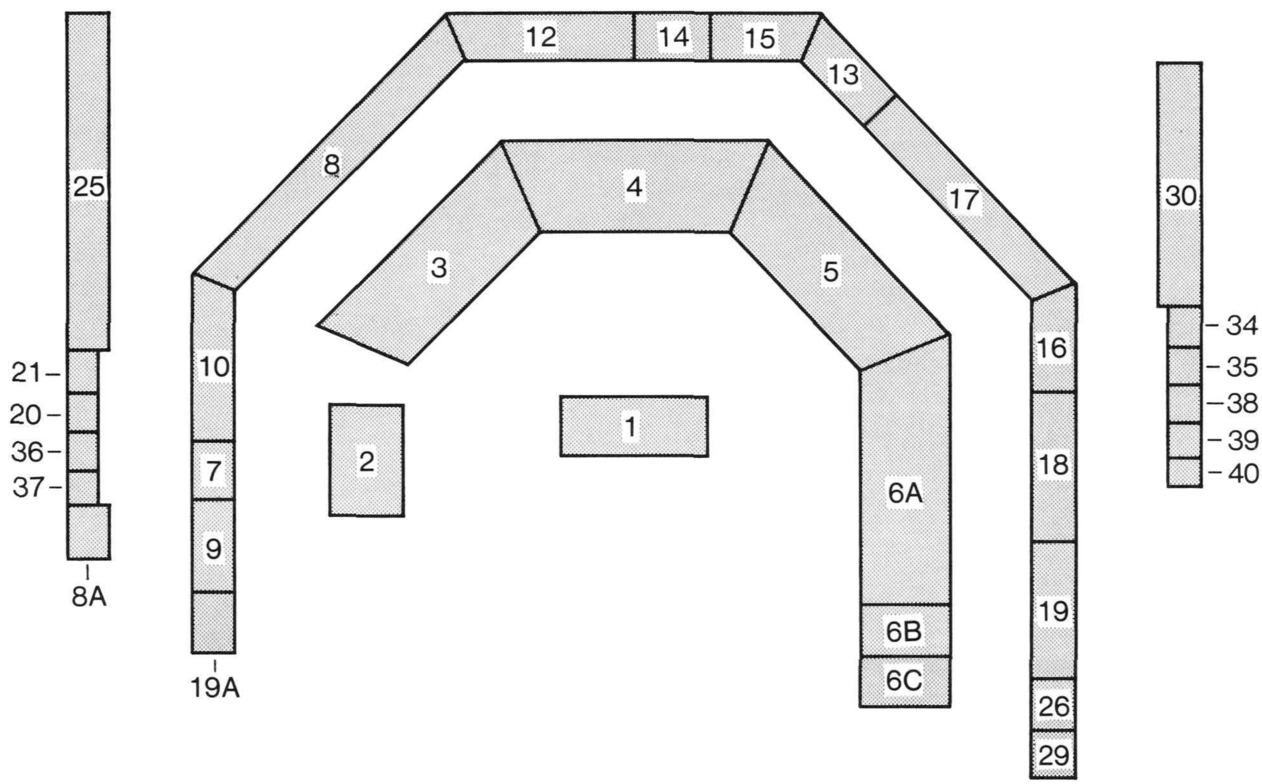
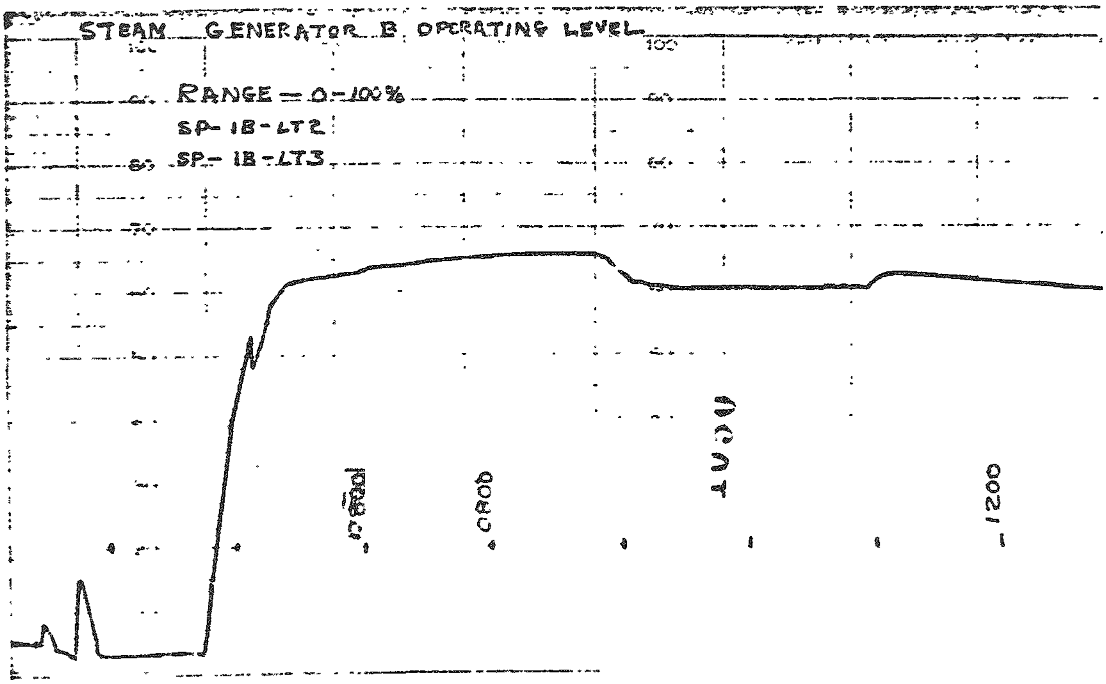
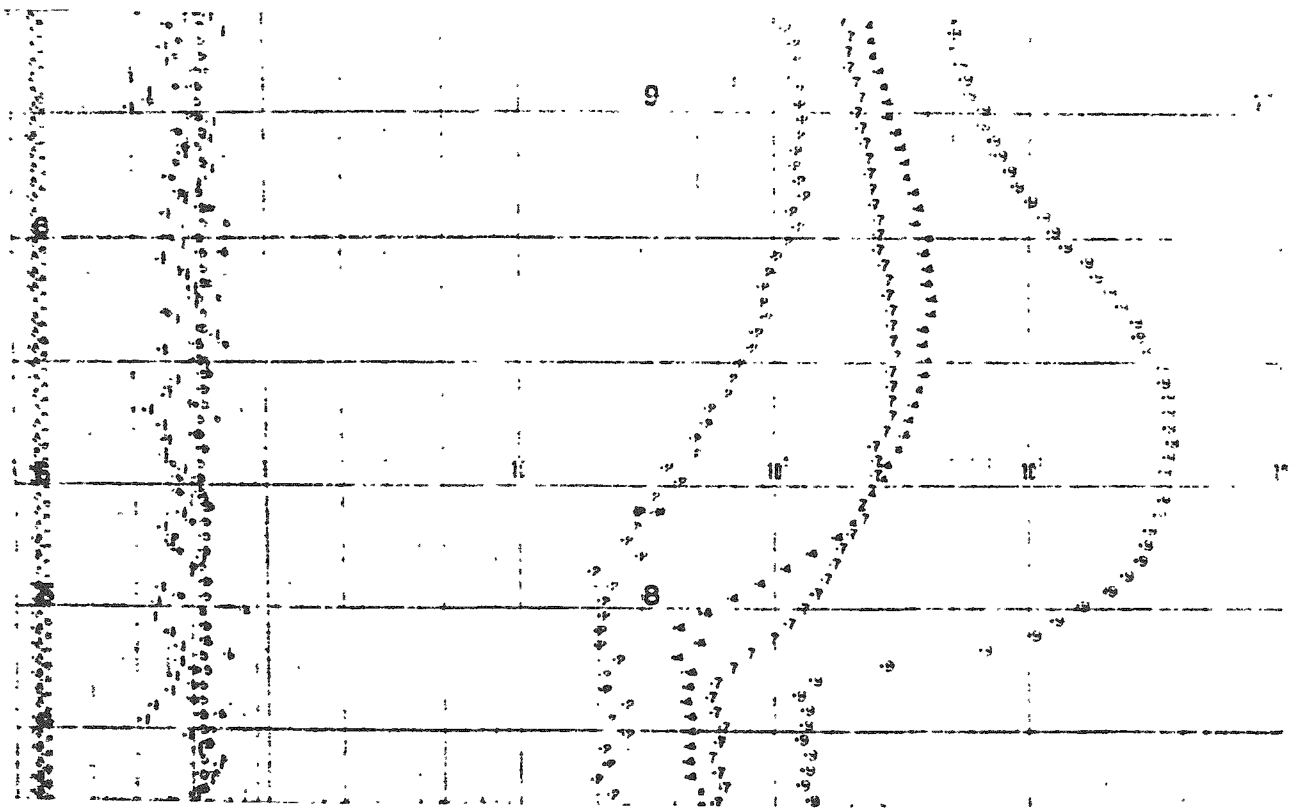


Figure PDS-1. Control Room Panel Layout

FIGURE PDS-2



A. Pen Plotter



B. Multipoint

APPENDIX PDS

FIGURE PDS-3
ALARM PRINTER OUTPUT

04:00:50	CONT	3172	RC MAKE-UP PMP 1A TRIPPED	NORM
04:00:51	CONT	2674	PRESS HTR GROUP 1	NORM
04:00:51	CONT	2675	PRESS HTR GROUP 2	NORM
04:00:51	CONT	2676	PRESS HTR GROUP 3	NORM
04:00:51	CONT	2677	PRESS HTR GROUP 4	NORM
04:00:51	CONT	2678	PRESS HTR GROUP 5	NORM
04:00:51	NORM	0117	EMER FD PMP1 DISCH PRESS	1237.
04:00:51	NORM	0118	EMER FD PMP 2A DISCH PRESS	1471.
04:00:51	NORM	0119	EMER FD PMP 2B DISCH PRESS	1445.
04:00:52	LOW	0136	FEED PUMP 1A OUTLET PRESS	795.
04:00:52	CONT	2766	11STG FDW HTR B LVL HIGH	NORM
04:00:52	CONT	2767	11STG FDW HTR A LVL HIGH	NORM
04:00:52	CONT	2881	RP SUR BYPASS (CH NI7)	NORM
04:00:52	CONT	2882	RP SUR BYPASS (CH NI8)	NORM
04:00:52	CONT	2879	RP SUR BYPASS (CH NI5)	NORM
04:00:52	CONT	2880	RP SUR BYPASS (CH NI6)	NORM
04:00:55	NORM	0024	HTR DR PMP 1A DISCH FL(KLB/H)	67.
04:00:56	BAD	0186	GENERATOR MW OUTPUT GROSS	-????.
04:00:56	CONT	2794	MS&R 2A RHTR DRAIN LVL HIGH	NORM
04:01:00	FLAG	3605	ELP1 FRROR POWER RANGE CH1	YES
04:01:00	FLAG	3606	ELP2 ERROR POWER RANGE CH2	YES
04:01:00	FLAG	3607	ELP3 ERROR POWER RANGE CH3	YES
04:01:00	FLAG	3608	ELP4 ERROR POWER RANGE CH4	YES
04:01:05	NORM	0096	COND HWL LVL (IN H2O)	26.44
04:01:06	CONT	2887	RP CH A RC HI PRESS	NORM
04:01:06	CONT	2893	RP CH C RC LO PRESS	TRIP
04:01:07	CONT	2891	RP CH A RC LO PRESS	TRIP
04:01:07	CONT	2892	RP CH B RC LO PRESS	TRIP
04:01:07	HIGH	0402	RC PRESS REL VLV RV2 OUT TEMP	239.2
04:01:07	HIGH	0404	RC PRESS REL VLV RV1B OUT TEMP	203.5
04:01:07	CONT	2894	RP CH D RC LO PRESS	TRIP
04:01:11	NORM	0186	GENERATOR MW OUTPUT GROSS	2.
04:01:15	LOW	0002	SP STM GEN A S-U RANGE LVL (IN)	23.8
04:01:16	CONT	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:01:17	LOW	0003	SP STM GEN B S-U RANGE LVL (IN)	23.7

FIGURE PDS-4

GROUP 12
11:18:53
03/28/79

OPERATOR SPECIAL SUMMARY B

0390	RC LOOP A OUTLET TEMP	-???.?
0589	RP LOOP A RC OUTLET TEMP 1 NRRW	-???.?
0590	RP LOOP A RC OUTLET TEMP 2 NRRW	-???.?
0392	RC LOOP A INLET TEMP NARROW 1	-???.?
0393	RC LOOP A INLET TEMP NARROW 2	-???.?
0391	RC LOOP B OUTLET TEMP	-???.?
0591	RP LOOP B RC OUTLET TEMP 1 NRRW	-???.?
0592	RP LOOP B RC OUTLET TEMP 2 NRRW	-???.?
0395	RC LOOP B INLET TEMP NARROW 1	-???.?
0396	RC LOOP B INLET TEMP NARROW 2	-???.?
1685	WFFA SP FDWTR FL A DP (K LB/HR)	297.
1686	WFFB SP FDWTR FL B DP (K LB/HR)	515.
0605	RP LOOP A RC DELTA PRESS (INH2O)	-???.?
0606	RP LOOP B RC DELTA PRESS (INH2O)	-???.?
0586	RP LOOP A RC PRESS 2 NRRW	2033.
1713	TOUTA AVG TEMP REACTOR OUTLET A	-333.3
1702	TINA SL REACTOR INLET TEMP A	176.0
1714	TOUTB AVG TEMP REACTOR OUTLET B	-333.3
1703	TINB SL REACTOR INLET TEMP B	185.2
1778	TSTA AVG SP MAIN STEAM TEMP A	429.7
1780	PSTA AVG STEAM PRESSURE A PSIA	16.
1779	TSTB AVG SP MAIN STEAM TEMP B	464.3
1781	PSTB AVG STEAM PRESSURE B PSIA	316.
1776	TFWA AVG FEEDWATER TEMP A	287.0
1777	TFWB AVG FEEDWATER TEMP B	414.4
1715	WRCA AVG RC FLOW A (M LB/HR)	.75
1716	WRCB AVG RC FLOW B (M LB/HR)	.75
0478	SP FDWTR FLOW A DP (IN/H2O)	2.
0480	SP FDWTR FLOW B DP (IN/H2O)	6.
0134	3STG HTR B OUTLET PRESS	166.
0135	3STG HTR A OUTLET PRESS	155.
1750	CORPW CORE POWER THERMAL (MW)	2710.

FIGURE PDS-5

14:04:14 GROUP TREND

OPERATOR GROUP C

	1032	0386	0390	0389	0468	0398	0472	0488	0469	0473	0471	0394	0463
14:05:16	200.2	50.9-	???.?	399.0	469.4	522.	439.4	417.5	219.1-	10.	405.5	192.7	335.5
14:06:33	196.1	51.3-	???.?	398.3	469.7	523.	439.3	417.3	219.0-	10.	405.3	195.3	336.9
14:08:13	193.9	52.3-	???.?	402.8	470.2	523.	439.1	417.2	219.0-	10.	405.2	197.0	338.1
14:10:13	194.5	53.9-	???.?	407.1	470.2	517.	438.8	417.0	218.9-	11.	404.9	198.0	339.2
14:12:13	195.4	55.7-	???.?	417.0	470.3	510.	438.5	416.9	218.9-	11.	404.7	197.9	340.2
14:14:13	195.8	58.4-	???.?	421.2	470.2	500.	438.1	416.7	218.9-	11.	404.4	199.7	341.0
14:16:14	195.5	61.3-	???.?	428.5	470.3	489.	437.8	416.6	218.8-	11.	404.2	199.2	341.7
14:18:14	195.0	64.6-	???.?	436.9	470.2	481.	437.6	416.4	218.8-	11.	404.0	200.3	342.3
14:20:14	194.6	67.2-	???.?	442.3	470.2	473.	437.2	416.4	219.1-	11.	403.7	201.6	342.7
14:22:12	194.4	69.2-	???.?	444.8	469.9	468.	437.0	416.3	219.2-	11.	403.6	203.1	343:3
14:24:14	194.3	71.5-	???.?	448.8	469.9	465.	436.7	416.1	219.0-	11.	403.2	203.2	343.7
14:26:13	194.0	72.5-	???.?	454.4	469.9	462.	436.4	416.1	219.3-	3.	403.3	209.6	344.0
14:28:13	194.1	68.5	560.8	452.3	469.9	450.	436.1	416.0	219.3	13.	404.7	212.9	344.2
14:30:14	193.5	69.7	548.0	451.6	469.7	443.	435.8	415.8	219.0	24.	405.5	207.3	344.4
14:32:15	193.0	73.8	548.2	451.3	469.6	437.	435.4	415.8	219.0	31.	405.9	203.0	344.8
14:34:13	191.8	76.7	567.1	451.1	469.4	423.	435.2	415.8	219.0	29.	405.8	205.8	345.2
14:36:14	190.0	80.1	612.4	449.2	469.2	409.	434.9	415.7	219.0	27.	405.3	206.2	345.6
14:38:14	190.3	67.7-	???.?	449.1	469.3	414.	434.6	415.5	219.0	25.	404.9	217.2	345.7
14:40:13	192.5	66.3	607.7	449.5	469.0	421.	434.3	415.5	219.0	25.	404.4	222.3	345.8

FIGURE PDS-6

SEQUENCE OF EVENTS REVIEW

04:00:35:833	3089	CNDS PUMP CO-P-1A	TRIP
04:00:36:729	3109	TD MAIN FW PMP FW-P-1B	TRIP
04:00:36:881	3108	TD MAIN FW PMP FW-P-1A	TRIP
04:00:36:955	3153	MAIN GENERATOR DIFF	TRIP
04:00:36:968	3113	EHC AUTO STOP TRIPPED	TRIP
04:00:36:997	3126	TURBINE TRIPPED	TRIP
04:00:37:294	3136	TURB INTERCEPT VALVE NO 4 CLOSED	CLSD
04:00:37:300	3133	TURB INTERCEPT VALVE NO 1 CLOSED	CLSD
04:00:37:305	3135	TURB INTERCEPT VALVE NO 3 CLOSED	CLSD
04:00:37:307	3134	TURB INTERCEPT VALVE NO 2 CLOSED	CLSD
04:00:37:323	3137	500 KV MN XFMR BKR B2-02	TRIP
04:00:37:349	3138	500 KV MN XFMR BKR B2-2602	TRIP
04:00:37:364	3131	TURBINE STOP VALVE NO 3 CLOSED	CLSD
04:00:37:368	3132	TURBINE STOP VALVE NO 4 CLOSED	CLSD
04:00:37:402	3130	TURBINE STOP VALVE NO 2 CLOSED	CLSD
04:00:37:444	3129	TURBINE STOP VALVE NO 1 CLOSED	CLSD
04:00:37:838	3129	TURBINE STOP VALVE NO 1 CLOSED	CLSD
04:00:44:186	3178	RP BLUE CH RC HI PRESS TRIP	HIGH
04:00:44:237	3177	RP YELLOW CH RC HI PRESS TRIP	HIGH
04:00:44:268	3176	RP GREEN CH RC HI PRESS TRIP	HIGH
04:00:44:283	3175	RP RED CH RC HI PRESS TRIP	HIGH
04:01:05:737	3181	RP YELLOW CH RC LOW PRESS TRIP	LOW
04:01:06:893	3179	RP RED CH RC LOW PRESS TRIP	LOW
04:01:06:932	3180	RP GREEN CH RC LOW PRESS TRIP	LOW
04:01:07:622	3182	RP BLUE CH RC LOW PRESS TRIP	LOW
04:01:15:152	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:01:15:191	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:01:15:301	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:02:35:500	3162	ES ACT A EMER INJ BT2 CH TRIP	TRIP
04:02:35:502	3165	ES ACT B EMER INJ BT2 CH TRIP	TRIP
04:02:38:548	3163	ES ACT A EMER INJ BT3 CH TRIP	TRIP
04:02:38:550	3166	ES ACT B EMER INJ BT3 CH TRIP	TRIP
04:02:39:506	3183	RP RED CH PRESS-TEMP TRIP	TRIP
04:02:40:157	3186	RP BLUE CH PRESS-TEMP TRIP	TRIP
04:02:40:415	3184	RP GREEN CH PRESS-TEMP TRIP	TRIP
04:02:43:925	3164	ES ACT B EMER INJ BT1 CH TRIP	TRIP
04:02:43:927	3161	ES ACT A EMER INJ BT1 CH TRIP	TRIP
04:02:49:761	3185	RP YELLOW CH PRESS-TEMP TRIP	TRIP
04:02:55:161	3159	DIESEL GEN DF-X-1B FAULT	FALT
04:05:51:761	3093	CNDS BSTR PMP CO-P-2B	TRIP
04:05:57:882	3093	CNDS BSTR PMP CO-P-2B	TRIP
04:07:00:833	3093	CNDS BSTR PMP CO-P-2B	TRIP
04:08:27:833	3115	EHC LOSS OF DC POWER	LOST
04:08:35:666	3089	CNDS PUMP CO-P-1A	TRIP
04:12:17:8[illegible]	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:12:17:878	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:12:18:296	3172	RC MAKE-UP PMP 1A TRIPPED	TRIP
04:22:17:833	3213	RC PUMP 2A OFF	OFF
04:22:17:931	3213	RC PUMP 2A OFF	OFF
04:22:17:990	3213	RC PUMP 2A OFF	OFF
04:22:18:313	3213	RC PUMP 2A OFF	OFF
04:22:18:353	3213	RC PUMP 2A OFF	OFF
04:22:18:530	3213	RC PUMP 2A OFF	OFF
04:22:18:578	3213	RC PUMP 2A OFF	OFF
04:22:18:615	3213	RC PUMP 2A OFF	OFF
04:22:18:655	3213	RC PUMP 2A OFF	OFF
04:22:19:124	3213	RC PUMP 2A OFF	OFF

MEMORY TRIP REVIEW

	577	578	579	580	581	582	583	584	392	395	589	591	585	587
03:45:45	97.4	97.1	97.4	97.2-	3.06-	3.65-	3.37-	3.31	558.1	557.0	605.6	606.0	2144.	2145.
3:46:45	97.5	97.3	97.6	97.2-	3.15-	3.71-	3.46-	3.40	558.6	557.5	606.0	606.4	2152.	2153.
03:47:45	97.0	96.8	97.0	96.9-	3.81-	4.31-	4.06-	4.03	558.5	557.7	606.2	606.5	2150.	2151.
03:48:45	97.3	97.1	97.3	96.7-	3.78-	4.28-	4.06-	4.00	557.9	557.1	605.8	606.1	2144.	2145.
03:49:45	97.4	97.2	97.4	97.1-	3.28-	3.84-	3.56-	3.50	558.2	557.2	605.8	606.1	2148.	2149.
03:50:45	97.6	97.3	97.5	97.3-	3.25-	3.78-	3.56-	3.50	558.5	557.5	606.1	606.4	2151.	2152.
03:51:45	96.8	96.6	96.8	96.8-	4.06-	4.59-	4.31-	4.28	558.4	557.5	606.2	606.5	2150.	2151.
03:52:45	97.4	97.1	97.3	97.1-	3.31-	3.87-	3.59-	3.56	558.1	557.2	605.7	606.0	2145.	2146.
03:53:45	96.9	96.6	96.9	97.2-	3.37-	3.90-	3.68-	3.59	558.6	557.7	606.1	606.4	2152.	2153.
03:54:45	96.8	96.6	96.8	96.7-	4.00-	4.53-	4.28-	4.25	558.2	557.6	606.1	606.3	2147.	2148.
03:55:45	97.3	97.1	97.3	97.1-	3.34-	3.87-	3.62-	3.56	558.0	557.1	605.6	606.0	2145.	2146.
03:56:00	97.3	97.1	97.3	97.1-	3.34-	3.90-	3.59-	3.59	558.0	557.1	605.8	606.2	2149.	2149.
03:56:15	97.4	97.2	97.4	97.1-	3.34-	3.90-	3.59-	3.59	558.3	557.3	605.8	606.2	2149.	2149.
03:56:30	97.4	97.2	97.4	97.2-	3.34-	3.90-	3.62-	3.59	558.3	557.3	605.9	606.3	2152.	2152.
03:56:45	97.5	97.2	97.5	97.2-	3.34-	3.90-	3.62-	3.59	558.5	557.5	605.9	606.3	2152.	2152.
03:57:00	97.5	97.2	97.5	97.3-	3.37-	3.93-	3.65-	3.59	558.5	557.5	606.1	606.5	2154.	2155.
03:57:15	96.9	96.7	96.9	97.3-	3.37-	3.93-	3.65-	3.59	558.9	557.7	606.1	606.5	2154.	2155.
03:57:30	96.9	96.7	96.9	96.7-	4.09-	4.65-	4.37-	4.31	558.9	557.7	606.1	606.4	2149.	2150.
03:57:45	96.8	96.5	96.7	96.7-	4.09-	4.65-	4.37-	4.31	558 4	557.5	606.1	606.4	2149.	2150.
03:58:00	96.8	96.5	96.7	96.6-	4.09-	4.62-	4.34-	4.34	558.4	557.5	605.9	606.1	2145.	2146.
03:58:15	96.7	96.5	96.7	96.6-	4.09-	4.62-	4.34-	4.34	558.0	557.2	605.9	606.1	2145.	2146.
03:58:30	96.7	96.5	96.7	96.5-	4.09-	4.62-	4.34-	4.31	558.0	557.2	605.6	605.9	2143.	2144.
03:58:45	97.1	96.9	97.1	96.5-	4.09-	4.62-	4.34-	4.31	558.1	557.2	605.6	605.9	2143.	2144.
03:59:00	97.1	96.9	97.1	97.0-	3.62-	4.15-	3.90-	3.84	558.1	557.2	605.7	606.1	2147.	2148.
03:59:15	97.2	97.0	97.1	97.0-	3.62-	4.15-	3.90-	3.84	558.3	557.3	605.7	606.1	2147.	2148.
03:59:30	97.2	97.0	97.1	97.0-	3.59-	4.15-	3.90-	3.81	558.3	557.3	605.7	606.2	2149.	2150.
03:59:45	97.2	97.0	97.2	97.0-	3.59-	4.15-	3.90-	3.81	558.4	557.4	605.7	606.2	2149.	2150.
04:00:00	97.2	97.0	97.2	97.1-	3.59-	4.15-	3.87-	3.84	558.4	557.4	605.9	606.2	2149.	2150.
04:00:15	97.2	97.0	97.2	97.1-	3.59-	4.15-	3.87-	3.84	558.3	557.3	605.9	606.2	2149.	2150.
04:00:30	97.2	97.0	97.2	97.0-	3.59-	4.15-	3.90-	3.84	558.3	557.3	605.8	606.2	2140	2150.
**														
04:00:45	94.9	94.6	94.4	92.2-	8.31-	9.25-	9.06-	8.62	558.3	557.3	607.1	607.4	2332.	2335.
04:01:00	94.9	94.6	94.4	92.2-	8.31-	9.25-	9.06-	8.62	558.3	557.3	607.1	607.4	2332.	2335.
04:01:15	1.3	1.3	1.4	1.3-	.25-	.18-	9.06-	8.62	576.3	574.4	607.1	607.4	2332.	2335.
04:01:30	1.3	1.3	1.4	1.3-	.25-	.18	.18-	.40	576.3	574.4	581.4	579.7	1868.	1869.
04:01:45	.5	.5	.6	.6-	.18-	.12	.31-	.34	574.8	573.3	581.4	579.7	1773.	1774.
04:02:00	.5	.5	.6	.6-	.18-	.12	.31-	.34	574.8	573.3	577.3	576.0	1773.	1774.
04:02:15	.2	.3	.4	.3-	.18-	.12	.34-	.34	575.8	573.6	577.7	576.0	1700.	1701.
04:02:30	.2	.3	.4	.3-	.18-	.12	.34-	.34	575.8	573.6	577.7	576.2	1700.	1701.
04:02:45	.1	.1	.2	.2-	.18-	.12	.40-	.34	577.0	574.8	578.7	577.1-	????.-	????.
04:03:00	.1	.1	.2	.2-	.18-	.12	.40-	.34	577.0	574.8	578.7	577.1-	????.-	????.
04:03.15	.0	.1	.2	.1-	.15-	.09	.40-	.34	577.6	576.0	578.3	577.8-	????.-	????.
04:03:30	.0	.1	.2	.1-	.15-	.09	.40-	.34	577.6	576.0	578.3	577.8-	????.-	????.
04:03:45	.0	.0	.1	.1-	.15-	.09	.40-	.31	577.5	576.5	578.3	578.1-	????.-	????.
04:04:00	.0	.0	.1	.1-	.15-	.09	.40-	.31	577.5	576.5	578.3	578.1-	????.-	????.
04:04:15	.0	.0	.1	.1-	.15-	.09	.40-	.31	577.7	576.8	578.5	578.3-	????.-	????.
04:04:30	.0	.0	.1	.1-	.15-	.09	.40-	.31	577.7	576.8	578.5	578.3-	????.-	????.
04:04:45	.0	.0	.1	.1-	.15-	.12	.43-	.31	577.9	577.0	578.6	578.3-	????.-	????.
04:05:00	.0	.0	.1	.1-	.15-	.12	.43-	.31	577.9	577.0	578.6	578.4-	????.-	????.
04:05:15	.0	.0	.1	.1-	15-	.09	.43-	.34	578.1	577.2	578.7	578.4-	????.-	????.
04:05:30	.0	.0	.1	.1-	.15-	.09	.43-	.34	578.1	577.2	578.7	578.5-	????.-	????.

APPENDIX PE

PRECURSOR EVENTS

Prior to the March 28, 1979 accident at TMI-2, several events had occurred at other B&W plants or at TMI-2 itself which could be regarded as precursors of the TMI-2 event in that they contained the same key elements. Only at TMI-2, however, did events proceed to the point of core damage. These precursor events are described in NUREG-0560 (Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company). A brief description and discussion of them is presented in this Appendix.

Key elements in the TMI-2 event were a transient in the feedwater system which led, as designed, to opening of the Electromatic Relief Valve (ERV), followed by failure of the valve to close upon reduction of pressure, with knowledge of that failure not directly available to the operator. Numerous feedwater transients have occurred in both B&W and other PWRs, and in many of them the Power Operated Relief Valve (termed the ERV at TMI-2) opened. B&W is reported to have estimated that, up to the time of the TMI-2 accident, about 150 such openings had occurred. In three of these the valve failed to reseat. The three events were:

June 13, 1975 Oconee 3	<u>Reactor power being reduced from 100% to 15%</u>. PORV stuck open following system transient, and rupture disc gave way in quench tank. 1500 gallons of water released to containment sump. Transient was terminated by operators' actions, including closing the PORV block valve. No core damage.

September 14, 1977 Davis-Besse 1	Reactor power at 9%. Following feedwater system trip, PORV opened and failed to close. HPI initiated and quench tank rupture disc gave way. Operator terminated transient by closing PORV block valve. No core damage.
March 28, 1979 Three Mile Island 2	Reactor at 98% power. Following feedwater transient, PORV opened and failed to close. Quench tank rupture disc gave way. Operators eventually terminated accident but only after severe core damage. Several hundred thousand gallons of water released to containment.

To these should be added a post-TMI-2 event at Crystal River 3, reference Florida Power Corporation Notepad entry to NSAC, February 29, 1980:

February 26, 1980	Reactor at 99% power. Following a transient induced by interruption of power to an instrument bus, PORV opened and failed to close when pressure dropped. HPI initiated. Quench tank rupture disc gave way. Transient was terminated by operator action. No core damage. About 43,000 gallons of water released to containment.

While the above brief descriptions of these events suggest generic deficiences in the PORVs, it is important to note that in the Davis-Besse case, a seal-in relay which should have been present in the valve control system was absent, leading to open/close oscillation of the valve and finally causing it to jam. In the Oconee case, malfunction of the valve was due to mechanical problems, including heat expansion, boric acid crystal buildups, rubbing of the valve lever on a bracket and bending of the solenoid spring bracket. The cause of PORV failure to close at TMI-2 is unknown. The cause of PORV failure to close at Crystal River was the instrument power failure. These findings emphasize that PORV malfunction may be due more to incorrect electrical installation or to extraneous electrical failures than to intrinsic mechanical faults in the valve itself, although both types of causes have occurred.

In all four cases cited above, no means of positive indication of the position of the PORV had been provided. Consequently, the operators had to deduce, from indirect observations, that the valve was open.

The above findings also emphasize the necessity for careful study of unintended occurrences in nuclear power plants, even though those occurrences may not have led to significant damage or lost production. Such study is necessary even if the unintended occurrence is in a system which is not one of the formally designated "safety systems," since in the cases cited no "safety system" malfunctioned--the malfunctions were in equipment and systems clearly designated as "non-safety" since they were regarded as unnecessary to safe shutdown of the plant in any "design basis accident." Clearly also, it is important that the study of unintended occurrences not be confined to one plant. The studies must be done on the occurrences as a group if generic problems are to be identified.

APPENDIX RB

REACTOR BUILDING COOLING SYSTEM

Description

The Reactor Building Cooling System consists of five air cooling units. They are arranged in a common housing and connected to a common system of ductwork for air distribution. These units (AH-E-11A, B, C, D, and E) are located outside the secondary shielding at elevation 305' in the south section of the reactor building. Each cooling unit contains a two-speed fan, which is rated at 42500 CFM during normal operation and at 32500 CFM during peak postaccident pressure conditions, and a series of 6 finned cooling coils. Figure RB-1 shows the major components of the system.

Air discharges from the fan coil units to a common plenum below the units. At each end of the plenum, one large supply duct is routed to an area where the steam generator and reactor coolant pump are located within the secondary shielding. The ducting on the west side of the reactor building serves steam generator (RC-H-1A), the reactor coolant pumps, (RC-P-1A and 2A) and the pressurizer (RC-T-2). The temperature of this air supply is monitored at point 6 on the reactor building temperature recorder. The other large supply duct, which is located on the east side of the building, serves steam generator (RC-H-1B) and reactor coolant pumps (RC-P-1B & 2B). It is monitored at point 5 on the reactor building temperature recorder.

Smaller sized ductwork provides cooled air to various equipment areas located below floor elevation 305'. The supply to the reactor coolant drain tank area is taken from the west end of the plenum. The supply to the letdown cooler area, reactor building sump pump area and the impingement barrier area for steam and feed piping is taken from the east end of the plenum. Air exhausting from these equipment areas returns to the fan inlets through two stairwells and various floor gratings and openings. The power for reactor building cooling fans AH-E-11A and 11B is supplied from the 480 V substation bus 2-11E (red) that is provided for engineered safety features. Fans AH-E-11D and 11E are supplied from bus 2-21E (green). Fan AH-E-11C can be powered from either of these two buses.

Reactor building temperatures are monitored on the reactor building temperature recorder (AH-XMTR-5017) located on Panel 25. This panel is located behind the main control panel. Table RB-1 lists the locations monitored by this recorder.

During normal plant operation, four of the five air cooling units are in use. Water is provided to the cooling coils by one of two full capacity reactor building normal cooling water pumps (RB-P1A or 1B) at the rate of 2400 gpm. Heat is rejected to the atmosphere by two half-capacity evaporative coolers (RB-Z-1A & 1B). During accident situations, the reactor building cooling system will transfer to emergency operation if the reactor building pressure reaches 4 psig. (see Appendix ESF) In this mode, the cooling water supply shifts to the nuclear services river water system. This is effected by closing each fan coil outlet valve that returns water to the cooling system outside of the reactor building. The transfer also causes the emergency cooling water supply valves to fan coolers (RR-V-5A, B, C & RR-V-6C, D, E) to open. The reactor building emergency cooling booster pumps (RR-P-1A, B, C, D) are also started and pump discharge valves (RR-V-2A, B, C, D) are opened. Each booster pump is rated at 3000 gpm. Each set of two pumps supplies flow to a train consisting of two fan coil units through a common discharge header. The cooler associated with fan AH-E-11C can be supplied from either of the two headers. The booster pumps receive electrical power from the same 480 V buses as their corresponding fan coil units. Pump suction is taken from the nuclear services river water system and discharges through the air cooling coils to the river via the mechanical draft cooling tower.

The reactor building pressure signal that triggers at 4 psig, also switches the air cooling fans to fast speed and opens two quick action valves which divert a portion of the air supply from the steam generator areas to an area above the operating floor.

System Response

Prior to the turbine trip at the onset of the accident, plant strip charts show that the reactor building was at a subatmospheric pressure of approximately -2 psig and an average temperature of approximately 116°F. Unit technical specifications require the building pressure to be maintained between -2 and +3 psig and the temperature not to exceed 130°F as determined by the arithmetical average of temperature recorder points 11-16 (Table RB-1).

It is assumed that the reactor building cooling system was in normal operation. This would call for operation of four of the five reactor building air cooling fans.

The turbine tripped at 0400:37. At approximately 0404 the reactor coolant drain tank relief valve lifted. This valve discharges into a funnel drain adjacent to the tank. At the time, a distinct temperature spike (from 87°F to 153°F) was noted on temperature recorder point 10. This point is taken from a temperature element located on the middle of the south wall of the drain tank room. Room temperature then decreased rapidly over the next 6 minute interval to 127°F. The drain tank rupture disc failed at 0415:25 and discharged to a point outside the room in the vicinity of stairwell #1 on the west side of the building. This stairwell opens onto floor elevation 305', immediately adjacent to column R5. A temperature element (recorder point 130) is affixed at elevation 326' of this column. From a pretrip reading of 120°F, the temperature at this recorder rose rapidly to 153°F shortly after 0415. This temperature rise is closely tracked by a comparable temperature rise in the west side supply temperature (recorder point 6). This suggests that the greater portion of the steam relief from the drain tank was pulled in by the two fan coil units AH-11D & 11E on the west side of the reactor building.

At the early stages of the transient, the operators reported that the reactor building fire alarms came on and that they noticed some of the RB temperatures increasing. Simultaneously, they also noticed an increase in reactor building pressure (panel 3). In response, they elected to start the reactor building emergency cooling booster pumps and to switch all 5 reactor building cooling fans to high speed. This appears to have occurred at approximately 0430 because the recorded reactor building pressure, which had previously increased rapidly to about 1.8 psig, slowed its rate of increase. At about the same time, the temperature of the east side air supply, which had been rising, peaked at 117°F and began to fall. Also, the temperature of the west side air supply, which had reached 134°F, began increasing at a slower rate.

Reactor building emergency cooling can be manually initiated in either of two ways: (1) By placing the engineered safety features "Reactor Building Cooling & Isolation Actuation" system into the "Manual Actuation & Test" position; or (2) By conducting a number of discrete component control manipulations at panels 3, 15 &25. The former does not appear to be the means of initiation the operators chose since reactor building isolation and safety injection were not indicated on the computer printout.

For a period of 60 minutes, starting at approximately 0433, the reactor building temperature history was lost because the recorder printed in one position.

At 0542 the B steam generator was isolated. Operators reported that the reactor building pressure stopped going up and started a slight downward trend. This is supported by the recorded reactor building pressure history. At about this time, reactor building pressure peaked at approximately 2.8 psig and reactor building temperatures began decreasing from a high of 151°F.

At about 0622:37 the ERV block valve (RC-V2) was closed. At this point, reactor building pressure began a steady decrease from 2.2 psig to 0.3 psig over the next 50 minutes. The reactor building west side temperature at elevation 326' (point 13) decayed rapidly, tracked rather closely by the supply air temperatures, with the west side (point 6) showing the more rapid decay. Thereafter, the temperatures plotted at points 6 & 13 exhibited significant swings, depending on the pressurizer ERV block valve position.

In the operator inverviews, it was stated that one reactor building emergency booster cooling pump was found not running. It has proved difficult to determine the actual system valve line-up that the plant operators had selected. However, the temperature of the air supply to the two side of the building was quite different indicating an imbalance in the amount of cooling. Later, following automatic initiation of reactor building emergency cooling, the two supply air temperatures exhibit a much closer relationship.

At approximately 0713 the operator opened the ERV block valve (RC-V2) again. Reactor building pressure which had decreased to approximately 0.3 psig, started rising, reaching a peak of approximately 1.6 psig at 0730 when the ERV block valve (RC-V2) was reclosed.

At approximately 0741 the operator reopened the ERV block valve (RC-V2). The reactor building pressure rose rapidly to approximately 3.2 psig and both engineered safety features channels actuated on reactor building high pressure. This actuation caused automatic initiation of reactor building isolation and full reactor building emergency cooling. This is substantiated from the data which shows that reactor building pressure flattened out at approximately 3.2 psig over the next 40 minutes. The two reactor building air supply temperatures started decreasing and converged to within 10°F of each other where previously they had been separated by 20°F to 30°F.

At about 0830 reactor building pressure started to increase, reaching a high of approximately 4.3 psig at 0905.

At 0918 the operator closed the ERV block valve. Reactor building pressure and temperatures then dropped rapidly. At 0940 reactor building pressure leveled out at approximately 1.6 psig.

During the next 2 hours, while the operators were maintaining reactor coolant system pressure by cycling the ERV block valve, reactor building temperatures stabilized at approximately 140°F (elevation 353'). Reactor building pressure rose slowly to approximately 2.3 psig and then commenced a decrease. The ERV block valve maneuvers were followed closely by temperature point 13 and the two supply air temperatures, with the west side supply responding more rapidly.

At 1139 the ERV block valve was again opened. Depressurization of the reactor colant system was begun. Reactor building pressure started rising and reached a value of 2.6 psig at 1200, where it remained constant for the next hour. Reactor building temperatures remained relatively constant, with the exception of point 13 which rose rapidly from 123°F to 165°F and then leveled off. At approximately 1310, the ERV block valve was closed again and reactor building pressure began to decrease.

At 1350 reactor building pressure had decayed to approximately 1.3 psig when an instantaneous pressure spike (approximately

28 psig) caused by a hydrogen burn was recorded. This pressure spike caused the reactor building spray pumps to start and spray was initiated into the reactor building. Simultaneously, all recorded reactor building temperatures, with one exception, spiked upward with the highest reading reaching 182°F. The exception was temperature element (point 11) which is located at the outside of the east side secondary shield wall above the operating floor, at elevation 353'. This reading dropped rapidly to 97°F, well below the air supply flow temperatures, and thereafter began a long slow rise. Within 25 minutes this temperature rose to above the two supply air temperatures. It was within 10°F of the other elevation 353' reading (point 12) 2.5 hours later. Because of this behavior, it is believed that the sensing element was damaged during the burn, and exhibited erroneous readings thereafter.

At 1356:07 the reactor building spray pumps were stopped. A short time later the rapid reactor building pressure decrease leveled off at approximately 2.2 psig and commenced a steady and continuous (2.5 hour) pressure decrease.

At 1530 the reactor building pressure became subatmospheric.

Between 1636 and 1700 the operators maneuvered the ERV block valve twice, causing reactor building pressure to rise to a peak of approximately +0.7 psig and then to decrease, dropping below atmospheric at 1730. At 1705 reactor building air cooling fan AH-E-11B tripped. Following this, the reactor building pressure and all temperatures stabilized for the remainder of the accident. The reactor building pressure remained at -0.6 psig and with an average building temperature of 114°F.

TABLE RB-1

REACTOR BUILDING TEMPERATURE RECORDER

AH-YMTR-5017 - Panel 25

PT	Element	Location
5	AH-TE-5015	Supply Air (column R19, outlet plenum, east side)
6	AH-TE-5027	Supply Air (column R1, outlet plenum, west side)
7	AH-TE-5013	Ambient-Impingement Area (feedwater/steam line penetration room)
8	AH-TE-5010	Ambient-Sump Area (pump room)
9	AH-TE-5011	Ambient-Letdown Cooler Area (cooler room)
10	AH-TE-5012	Ambient-Drain Tank Area (wall mounted, middle south wall)
11	AH-TE-5020	Ambient-Elev. 353-1 (outside s.s. wall, east side, elev. 353')
12	AH-TE-5021	Ambient-Elev. 353-2 (outside s.s. wall, west side, elev. 353')
13	AH-TE-5023	Ambient-Elev. 330-1 (column R5, elev. 326')
14	AH-TE-5022	Ambient-Elev. 330-2 (column R16A, elev. 330')
15	AH-TE-5014	Ambient-Elev. 310-1 (column R4, elev. 310')
16	AH-TE-5088	Ambient-Elev. 310-2 (outside stairwell wall, between columns R17a & R18a, elev. 310')

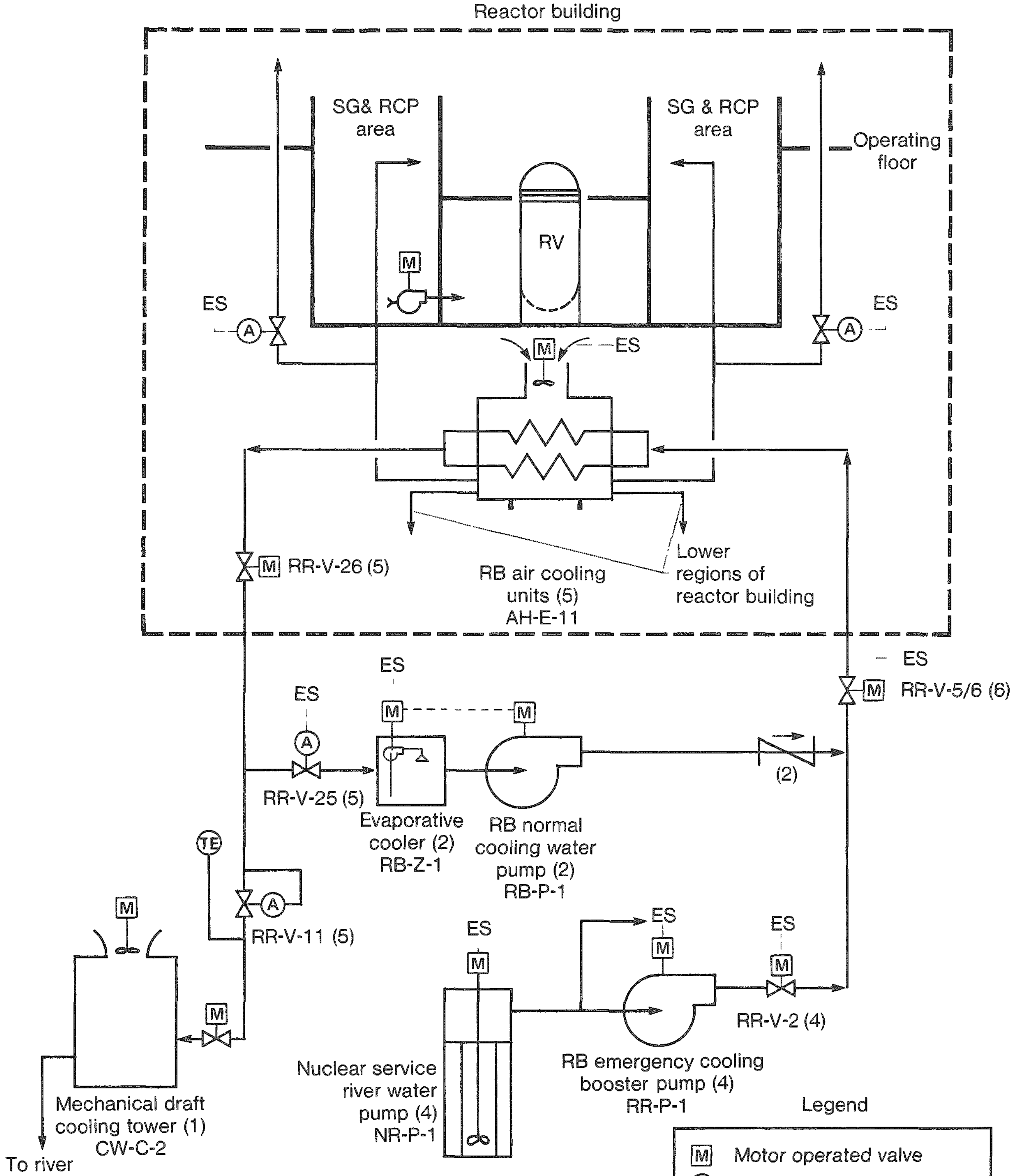

Reactor building
SG& RCP area
SG & RCP area
Operating floor
RV
M
ES
A
ES
RB air cooling units (5) AH-E-11
Lower regions of reactor building
RR-V-26 (5)
RR-V-5/6 (6)
RR-V-25 (5)
Evaporative cooler (2) RB-Z-1
RB normal cooling water pump (2) RB-P-1
(2)
TE
RR-V-11 (5)
Mechanical draft cooling tower (1) CW-C-2
To river
Nuclear service river water pump (4) NR-P-1
RB emergency cooling booster pump (4) RR-P-1
RR-V-2 (4)
Legend
M Motor operated valve
A Air operated valve
ES Operated by ESF signal

Figure RB-1

APPENDIX RCPCS

REACTOR COOLANT PRESSURE CONTROL SYSTEM

1. INTRODUCTION

The pressure in the reactor coolant system is controlled by the presurizer. The pressurizer (Figure RCPCS-1) is a vertical cylindrical vessel which is connected to the reactor coolant system through the surge line. The pressurizer establishes and maintains the reactor coolant pressure within prescribed limits and provides a surge chamber and water reserve to accommodate changes in reactor coolant density during operation. Through the use of the electric heaters and a reactor coolant spray system, steam and water in the pressurizer are maintained at saturation temperature conditions corresponding to the desired reactor coolant system pressure. During periods of reactor coolant system pressure decrease, some of the water in the pressurizer flashes to steam to maintain pressure. The electric heaters are also actuated to restore the pressure to the normal or desired pressure.

During periods of reactor coolant system pressure increase, any energized heaters are turned off first. Reactor coolant is then sprayed from a line connected to one of the reactor inlet lines into the pressurizer to condense steam, thus reducing pressure.

Coolant level in the pressurizer is controlled automatically by a circuit which positions the makeup control valve in the makeup and purification system. The input to this circuit is a temperature compensated signal selected manually from one of three redundant pressurizer level indicating instruments, and one of two pressurizer temperature elements.

2. COMPONENTS USED IN PRESSURE CONTROL

Figure RCPCS-2 shows reactor coolant system pressure operating and control parameters. Reactor coolant system pressure is controlled by actuation of the following components:

2.1 PRESSURIZER HEATERS

The pressurizer heaters are grouped in banks which are energized below a preset pressure. The lower and middle banks utilize proportional control through SCR controllers, and during steady state operation they operate at partial capacity to replace heat lost, thus maintaining the reactor coolant system pressure at the setpoint value. The remaining pressurizer heaters are controlled in an on-off mode which energizes each bank progressively upon decreasing reactor coolant system pressure. Each bank of heaters can be controlled either manually or automatically from the control room.

If the pressurizer coolant level decreases to below an indicated 80 inches, all pressurizer heater banks are automatically tripped to prevent heater damage.

When the heater banks are switched from one control mode to another, the heaters are automatically tripped, and then reenergized as appropriate in the selected control mode.

2.2 PRESSURIZER SPRAY VALVE

This valve is designed to modulate open as reactor coolant system pressure increases above the setpoint value.

2.3 ELECTROMATIC RELIEF VALVE

This valve is designed to reduce reactor coolant system pressure, if it exceeds 225 psig, by blowing steam from the pressurizer steam space to the reactor coolant drain tank.

2.4 PRESSURIZER CODE SAFETY VALVES

These two valves are designed to prevent overpressurization of the reactor coolant system by relieving fluid to the reactor coolant drain tank. These valves should lift only if quenching with reactor coolant spray and relief through the electromatic relief valve fail to stop an increase in reactor coolant system pressure.

2.5 PRESSURIZER VENT VALVE

This valve is designed to allow the operator to vent noncondensible gases from the pressurizer steam space to the reactor coolant drain tank. This valve is controlled manually from the control room by the operator, and it also can be throttled, as necessary, during venting.

3. COMPONENT OPERATION

3.1 PRESSURIZER HEATERS

3.1.1 Proportionally Controlled Pressurizer Heaters (Banks 1, 2, 3)

Controller in the Automatic Mode

Control in this mode is provided for pressurizer heater banks 1, 2, and 3, which are comprised of six heater groups with a total maximum heat input of 756 Kw. These heater banks respond sequentially through one controller to deviations in reactor coolant system pressure. In addition, a voltage varying device (5 CR controllers) allows a controller to program these heater banks from partially to fully energized. (See Figure RCPCS-2 for detailed settings).

Controller in the Manual Mode

The proportionally controlled pressurizer heaters can be manually operated in a variable-voltage sequential manner as described above or as a unit (all three banks) in a full on or full off manner.

Control Room Indications

The controller for the proportional heaters, when in the automatic mode, indicates the percentage of heater bank output being requested due to pressure deviation. A status light is provided to indicate the mode of operation selected. Reactor coolant system pressure indication is provided near the controls; therefore, pressure response to heat addition can be monitored. The computer is used to monitor the status of the heater power supplies.

3.1.2 On-Off Controlled Pressurizer Heater Banks (Banks 4 & 5)

Control in the Automatic Mode

With the control switch for heater banks 4 and 5 in the AUTO position, the heater banks are energized at preset values of the decreasing reactor coolant system pressures. These banks will remain fully energized until the reactor coolant system pressure has increased above the reset procedures. (For detailed settings, see Figure RCPCS 2.)

Control in the Manual Mode

When the control switch for heater banks 4 and 5 is not in the AUTO position, the heaters are either fully energized or fully deenergized depending on the position of the manual control switch.

Control Room Indications

Each heater bank has a labeled control switch with status light indication of the selected mode of operation.

Reactor coolant system pressure indication is provided near the pressurizer heater control station. This allows the operator to monitor pressure responses to heat addition.

The computer is used to monitor the status of the heater power supplies.

3.2 PRESSURIZER SPRAY VALVE (RC-V1)

3.2.1 Operation

The pressurizer spray valve can be automatically or manually controlled, depending upon the position of the control switch. In AUTO, the spray valve will cycle between 40% open and closed in response to deviations between reactor coolant system pressure and the pressure setpoint of the spray valve controller. When the control switch is in MANUAL the operator can adjust the spray valve position as necessary between full close and full open. (For detailed settings, see Figure RCPCS-2).

3.2.2 Control Room Indications

The pressurizer spray valve has a labelled control switch on the main control panel with valve position status light indications of CLOSED, 40% OPEN and OPEN. Reactor coolant system pressure indication is provided near the pressurizer spray valve control switch so that pressure responses to spraying action can be monitored.

3.3 ELECTROMATIC RELIEF VALVE

3.3.1 Operation

The electromatic relief valve is controlled by an on-off signal provided by an electronic pressure switch. A two position key lock switch, ON and AUTO, determines valve actuation. In the AUTO position, the valve opens and closes at the setpoint shown in Figure RCPCS-2. In the ON position, the valve will open regardless of reactor coolant system pressure.

3.3.2 Control Room Indications

The electromatic relief valve has a labelled keylock switch in the control room to indicate the mode of operation of the valve. A light on the control room console is on when the valve solenoid coil is energized, but this light does not necessarily indicate valve status. In the accident, the electromatic relief valve stuck open when the solenoid deenergized, but the indicating light went out, which would normally mean the valve was closed.

3.4 PRESSURIZER CODE SAFETY VALVES

3.4.1 Operation

The pressure code safety valves are spring loaded; they open when the pressure is sufficient to overcome the spring loading. No external signal or energy is necessary to cause them to open. At the setpoint listed in Figure RCPCS-2, the safety valves will open to prevent reactor coolant system overpressurization.

3.4.2 Control Room Indication

No direct indication of pressurizer code safety valve position is available in the control room. Valve position may be inferred from the rate of change of reactor coolant system pressure, reactor coolant drain tank pressure, temperature and level, and safety valve discharge temperatures. Reactor coolant system pressure is displayed on a front control console. Indications of reactor coolant drain tank pressure, level, and temperature, with alarm of these parameters when off-normal, is provided on a back panel in the control room. The computer is used to monitor thermocouples which are clamped to the tailpipes of the electromatic relief valve and the safety valves. An alarm is indicated if the thermocouple temperatures exceed 200°F.

3.5 CHANGE IN ELECTROMATIC RELIEF VALVE OPEN SETPOINT

In the aftermath of the TMI-2 accident, certain changes were made in all Babcock and Wilcox nuclear power plants to reduce the number of electromatic relief valve actuations and the resulting number of valve failures. The setpoint for opening the relief valves has been raised, and the reactor protection system high reactor coolant system pressure trip setpoint has been lowered. An additional reactor trip point has been included based on secondary system upsets such as turbine trip or feedwater pump trip. The result of these changes is that the electromatic relief valve no longer acts as a buffer in dampening mild reactor coolant system pressure transients and preventing a reactor trip. Rather, the reactor trips to avoid actuation of the relief valve during or in anticipation of a pressure transient.

The number of undercooling events has been reduced by these changes, but the number of overcooling events has been increased because of the rapid response of the Babcock and Wilcox plants to feedwater flow imbalances and the initiation of emergency feedwater. Overcooling can lead to initiation of high pressure injection and difficulty in differentiating between a small break loss of coolant accident and an excessive feedwater transient.

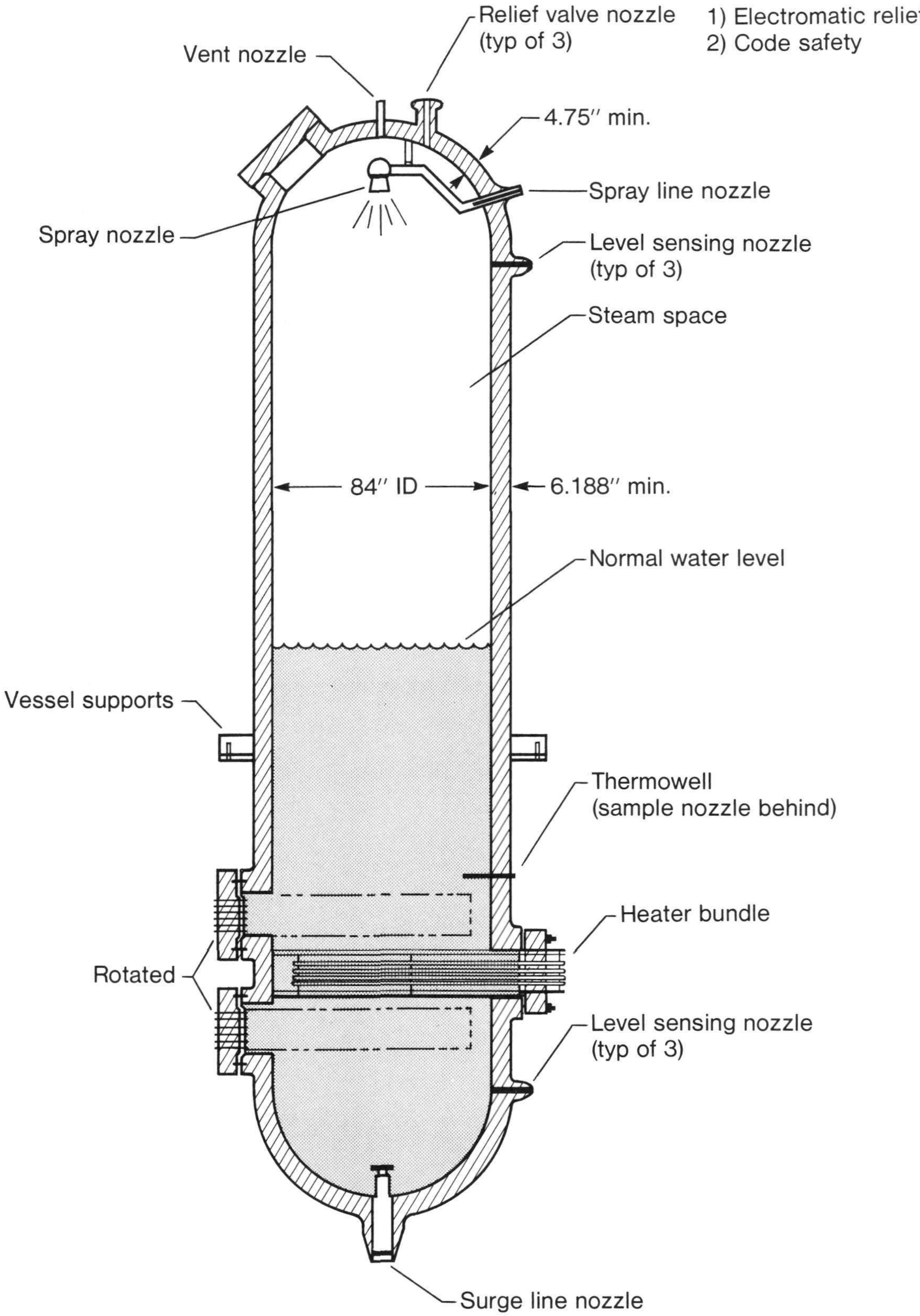
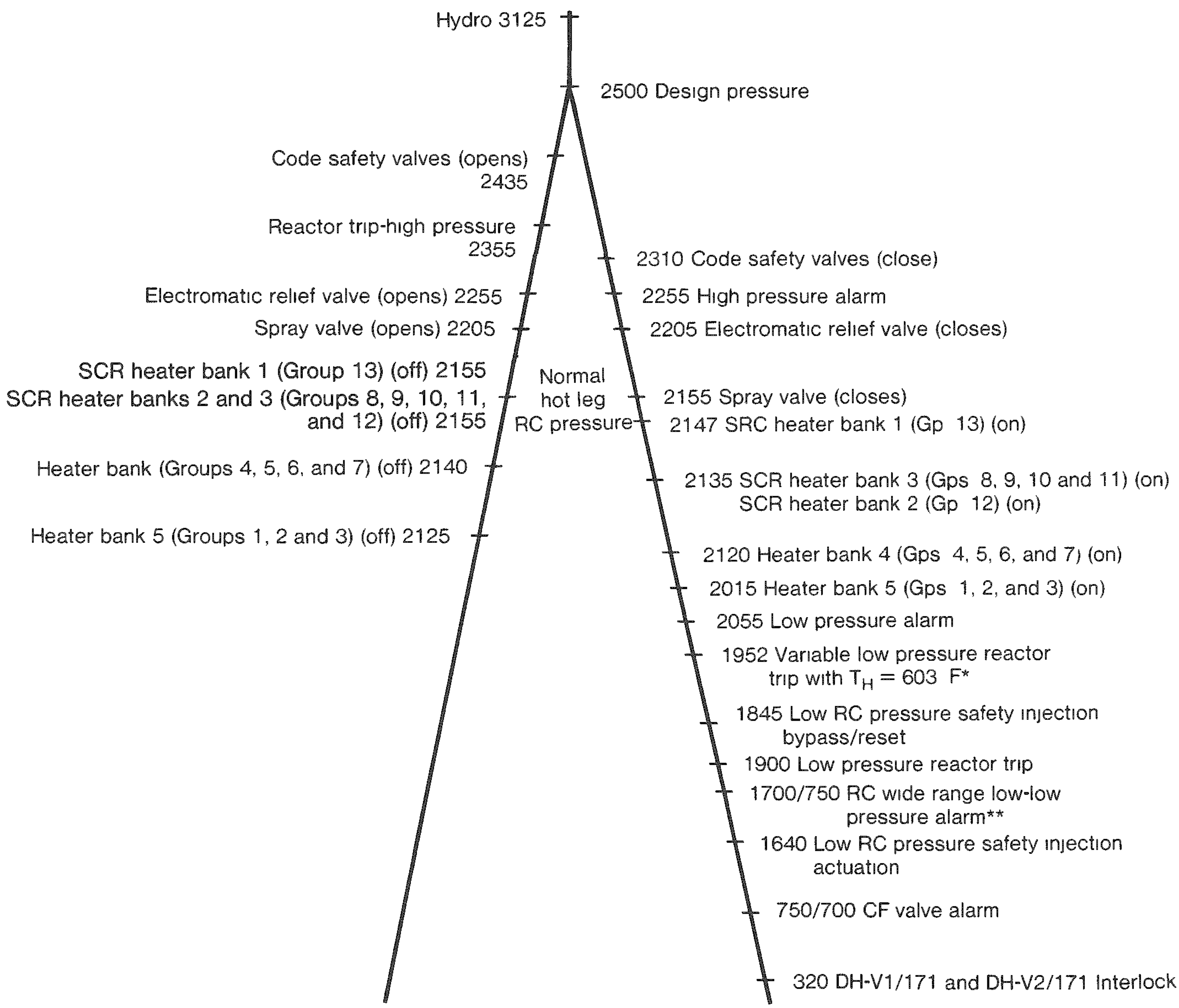


Figure RCPCS 1. Pressurizer Layout



Note Values in psig at 36 RCS
Hot leg piping pressure tap
* or (13 00 T_H – 5887) < RC pressure
**If related safety injection bistables are not bypassed

Figure RCPCS 2. Reactor coolant system pressure operating and control parameters.

APPENDIX RECRIT

ANALYSIS FOR POSSIBILITY OF RECRITICALITY

The Three Mile Island Unit-2 nuclear generating station is equipped with a variety of reactivity control features, designed for the purposes of keeping the plant within safe operating limits, under normal and abnormal service conditions. The reactivity control system design is tied to the station design basis, which includes a set of postulated transients or accident conditions. Since the Three Mile Island accident is believed to have exceeded the station design basis, questions have been raised as to the capability of reactivity control systems in maintaining the plant in a subcritical condition during the course of the event. Thus, the issue of recriticality has been addressed in the post-accident inquiry.

In this appendix the recriticality question is explored in terms of two rather broadly interpreted accident phases. The first phase extends from reactor trip through the initial core uncovery, but prior to significant core degradation or disarray. Nominally, this is the time period from 0400 to 0630. The second phase covers the balance of the accident period (i.e., after 0630). In this period, substantial reactor core disarray is believed to have occurred.

The subsequent discussion will conclude that there is little likelihood of recriticality or conditions approaching recriticality during the first phase of the accident. This conclusion is contrary to primary indications, construed by reactor operators, that the reactor may not have been adequately shut down (subcritical). For the second accident phase it is concluded that recriticality or near criticality was not likely to have occurred. However, the uncertainties in regards to both the dynamics and extent of core degradation makes this conclusion less definitive.

First Phase (0400-0630)

In a normal reactor trip, control rods are inserted to the bottom of the core, and the power level begins to decay in accordance with the 80-second period, consistent with the longest delayed neutron group half life. The reactor power falls below the power and intermediate ex-core instrument ranges, entering the source range. A typical source range power decay is shown in Figure 1. The power decay continues in accordance with the 80-second period until intercepted by the base count rate, defined by the source neutron production and subcritical multiplications. At Three Mile Island Unit-2 two neutron sources are important in determining the normal count rate curve: (1) installed Am-Be-Cm start-up sources, located at diametrically opposite locations at the core midplanes; (2) photo-neutrons (γ -n) generated by interaction of high-energy fission product gammas (primarily Kr^{88} and La^{140}) with deuterium (D_2O). During the early accident period, the photo-neutron source is the most important; the installed sources fix the ultimate core level count rate after photo-neutron sources die away.

The actual power decay time history at Three Mile Island Unit-2 was quite different from the nominal shutdown curve, as illustrated in Figure 2a. Instead of breaking from the 80-second period and continuing a downward trend, at a slow rate of decay, the source range recording began turning upwards at about the 30-minute mark. This upward trend continued until the reactor operator secured the reactor coolant pumps (at 100 minutes after trip), whereupon the count rate abruptly dropped to the base count rate level. Almost immediately thereafter, the count rate commenced a steep rise, reaching a peak that is nearly three decades above the normal. The intermediate range instrument recording (not shown) follows the source range recording where the two instrument ranges overlap.

In the time interval the source and intermediate range instruments were near their peak values (approx. 0630), some of the in-core self-powered neutron detectors began to behave erratically. High currents on some detectors were suggestive of substantial neutron fluxes in localized core regions.

The reactor operators initiated a manual (precautionary) scram at 0420 and checked rod bottom indicators to assure control rods were properly inserted. This may have been in response to abnormal ex-core neutron detector readings. As count rates continued to rise, the operators requested boron analysis at 0605 and 0630. The successive samples gave boron concentrations of 700 ppm and 400 ppm. These concentrations were low relative to the normal boration requirements at the existing stage in the fuel cycle, and tended to reinforce notions that the reactor may not have been adequately shut down as power boron concentrations a few hours earlier were 1030 ppm. Emergency boration was commenced by the operators prior to 0640.

The ex-core detector readings, in-core self-powered detector data, and boron analyses all point to a reactivity problem when these data are interpreted at face value. Nevertheless, careful analysis of instrument behavior, given a general understanding of what was going on in the core at the time, provides an alternative explanation.

In the minutes after the reactor trip, the primary system water inventory began to decrease as fluid was lost through the stuck-open electromatic relief valve. At saturation pressure, steam voids began to accumulate in the system. As two-phase mixture was pumped through the downcomer and core, three effects were manifest: (1) less water in the core decreased the intrinsic neutron source reading; (2) decreased fluid density in the downcomer permitted more neutrons to leak out to the ex-core detectors; (3) increased leakage from the core reduced neutron multiplication.

In order to reconcile the three somewhat competing effects neutron transport analyses have been performed to explain the source range detector behavior. In the first set of calculations one-dimensional (ANISN) transport analyses were used to determine detector count rates for homogeneous voiding of the core and downcomer regions. This model is appropriate for understanding the source range recording (Figure 2a) during the period of time the reactor coolant pumps were running (up to 0140 hours after reactor trip.) The results from these calculations are discussed immediately below. This discussion is followed by a presentation of two-dimensional neutron transport analyses, appropriate for the period immediately after reactor coolant pumps were secured (at 0140 hours).

The results of ANISN calculations for homogeneous voiding of the reactor core and downcomer are summarized in Table 1. A series of calculations were performed at varying void fractions. The homogeneous assumption and one-dimensional transport analyses are assumed to be valid on the basis of pump operation, acting to mix and distribute steam voids throughout the core and downcomer regions. Core average temperature was assumed to be 500° and soluable boron concentration at 1030 ppm for these calculations. A nominal core geometry was used.

Comparing the peak detector count rate in Figure 2a at 0140 hours, it may be observed that the average void fraction in the core/downcomer region was somewhere between 40-50% just prior to securing the reactor coolant pumps. This value is generally consistent with independent estimates of void fraction, based upon two-phase pump performance.

The one-dimensional analysis results confirm that the dominant influence on detector response is voiding the reactor vessel downcomer. This contributes to an increase in detector efficiency which more than out-weighs the effect in loss of source and water moderator. The net result was increasing counts seen by ex-core detectors, even while the reactor was becoming more subcritical. Consequently, it is fair to conclude that while homogeneous voiding prevailed (i.e., when reactor coolant pumps were running) the reactor was actually less reactive than immediately after shutdown. The upturn in the source range recording was the product of increased detector efficiency, due to the accumulation of steam voids in the downcomer.

The picture is more complicated after the pumps were stopped and phase separation occurred (after 0140 hours). As forced coolant flow ceased, falling liquid temporarily filled the downcomer. This resulted in an abrupt drop in the detector count rate (c.f. Figure 2a). As the core commenced to boil down, the downcomer water level dropped and more of the core came into view of the neutron detectors (Reference Figure CI-6 Appendix CI). As water was boiled out of the core the γ-n source began to diminish. In addition, increased neutron leakage from the core caused a reduction in neutron multiplication.

Evaluation of these competing effects under the non-homogeneous configuration necessitated multi-dimensional neutron transport analyses.

The multi-dimensional transport problem was analyzed using a DOT code R-θ /R-Z calculation under a 42-group Hansen and Roach cross section format. Core average temperature, soluable boron concentration, and geometry were the same as in the one-dimensional analysis. Results are shown in Figure 3. The curve and values for K_{eff} in the figure are based upon an axial void fraction profile which has been revised. Nevertheless the general trends are believed to be representative.

The transport analysis suggests that the "unshuttering" effect accompanying the drop in the downcomer water level dominates until the downcomer water level drops to about 6 feet. This is consistent with the one-dimensional results for a homgeneously voided downcomer (and core). When the core water level drops below a certain point the loss in γ-n source tends to assert itself. This causes the curve to bend over (reference Figure 3).

The change in reactivity during core boil-down is relatively modest until the water level almost reaches the bottom. K_{eff} drops from .937 to about .88 and holds fairly steady down to about two feet. This analysis is based upon an assumed boron concentration of 1030 ppm. Concentration by core boil-off may have somewhat reduced these K_{eff} values.

The shape of the curve in Figure 3 is fully consistent with the source range curve in Figure 2a, after 0142 minutes. The drop in downcomer water level leads to an increased detector efficiency, which produces increased count rates. The reactor remains subcritical, and is less reactive than when it was filled with coolant.

The two-dimensional neutron transport calculations permit conclusions to be drawn which are similar in nature to the one-dimensional results: voiding of the core and downcomer regions will produce source range detector responses that are entirely consistent with the recorded plant data. Recriticality was unlikely, given fairly reasonable assumptions about conditions that prevailed and K_{eff} values obtained.

The neutron transport analyses were used to characterize ex-core neutron detector behavior. However, these analyses do not explain the high currents observed on in-core self-powered neutron detectors.

The analysis of in-core self-powered neutron detector behavior during core boil-down and heat-up also suggests that detector currents were not a product of core recriticality. As explained in Appendix CI, the rhodium-Inconel detectors are subsceptible to a thermionic effect at abnormally high temperatures. Recent oven tests indicate that the detectors develop a small positive current (< 50 na) up to about 1000°F, whereupon the current abruptly changes polarity, reaching large negative values at high temperatures (> 2000°F). Positive and negative currents were observed at Three Mile Island; however, the small positive currents obtained from oven test is less than recorded currents at Three Mile Island*.

Although the large positive currents that were observed at Three Mile Island have not been fully confirmed by oven tests, it is reasonable to conjecture that temperature, as opposed to neutron flux, is the dominant factor influencing their behavior.

Satisfactory explanation of low boron concentrations, determined from samples at 0605 and 0630, has been a continuing problem. In some post-accident analyses these low concentrations have been ascribed to "flashing" in the letdown line or other inadvertent means of deriving "unrepresentative" boron concentrations. Sample analyses are believed to have been correct, since independent analyses by different persons yielded essentially consistent results, using the 0630 sample.

It now appears that low boron concentrations are the product of boron dilution in the A loop side, caused by distillation of borated water in the core and the accompanying condensation of boron free steam in the A loop steam generator, (boron volatility is low).

Prior to securing the reactor coolant pumps, plant operators commenced feeding the A loop once-through steam generator (OTSG) secondary side to re-establish level in the operating range.

* Oven tests were performed without the presence of gamma radiation, and it is believed that the radiation may accentuate positive currents at the elevated temperatures. Consideration is being given to experimental study of this behavior.

Feedwater spraying onto the OTSG tubes provided an efficient condensing medium for steam generated in the core; it is believed that a majority of liquid lost from the core during the initial boildown was transported into the A loop, rather than passing out the open relief valve. The net effect was a gradual reduction in boron concentration in the A loop on account of the dilution and increased boron concentration in the core. The imbalance in boron concentrations persisted at least until the reactor vessel had been refilled above level of the cold leg penetration.

Since chemistry samples are drawn from the low point in the A loop, it is not unreasonable to expect the low boron concentrations measured by the operators.

A final argument relative to the boron concentration problem has to do with the effect on reactivity, given that such dilution of boron in the core actually occurred. According to the station safety analysis report, boron worth is figured at approximately 0.01% Δ K/K per ppm for an undamaged core. A reduction in boron from 1030 to 400 ppm should have increased reactivity by about 6%. However, rod worth inserted at reactor shutdown is in the neighborhood of 7%; transient xenon can be estimated at this time period at about 2%. On balance, then, the reactor would have been 3% subcritical after the supposed dilution (nominal core geometry assumed).

This assessment is approximate, and assumes an intact core geometry at a 500°F temperature. Other analyses(1) postulate different fuel damage conditions which give higher reactivity values. In some extreme cases (e.g., complete control rod and burnable poison rod destruction or removal) recriticality is possible.

Second Phase (after 0630)

After core disarray the recriticality question is difficult to answer conclusively, owing to uncertainties in fuel geometry. It has been superficially argued that recriticality is unlikely simply on the basis that any core degradation will represent a departure from a near-optimal geometry, designed for criticality in the first place. Conversely, it is possible to show (Nuclear Safety Guide TID - 7016) that 2.6% enriched uranium, optimally mixed with water moderator/reflector can produce a critical volume of under 70 liters (150 g/ℓ of UO_2); this is consistent with the station safety analysis report that a minimum of two clean moderated fuel assemblies are together sufficient to achieve criticality. Both extreme positions are likely to fall on either side of the range of conditions which actually occurred at TMI.

The case for or against recriticality must ultimately depend upon plant data analysis. Here, it is possible to show that recriticality is not likely to have occurred; however, it is not entirely clear whether or not core degradation may have substantially reduced the margin of shutdown.

Nominally at least, an uncontrolled criticality would be accompanied by a sudden change in neutron count rates and (possible) evidence of energy release necessary to rearrange the fuel configuration into a subcritical configuration. Within the limits of resolution, the downward trend in the count rate should differ from the upwards trace, on account of the delayed neutron fraction.

Reviewing the source range instrument recording (Figure 2b) three candidate events are identified, occurring at 0747, 1350 and 1830. Among these the event at 0747 is the most interesting. That a significant energy release took place is evident by the overlay of other plant parameters, shown in Figure 4. A review of the sequence of events indicates that whatever happened at 0747 originated from within the core region and not from operator or equipment action outside. The event is likely to have occurred after core refill, since the high pressure injection system had been in operation for some 18 minutes prior to 0747.

The major difficulty in attributing the 0747 event to recriticality is the small variation in the source range signal; count rates changed by only about a factor of two. A simple thermal hydraulic analysis of the 0747 event suggests an energy release on the order of 2.3×10^6 BTU's. Assuming (conservatively) that fission produced this amount of energy over a 1 sec. time interval, power generation in excess of 80% full power would have been achieved. A spike in the source range, followed by decay in accordance with the 80-second period should have occurred; it did not. Moreover, pulses in the intermediate and power ranges should have been observed. None such were observed on the intermediate range. Power range data were recorded by the reactimeter at 3-second intervals; no statistically significant variations in power range detector current can be discerned. It is concluded that the 0747 event while yet unexplained, is unlikely to have been caused by recriticality.

The events at 1350 and 1830 are dismissed from consideration on the basis of: the small magnitude change in source range count rates and the lack of any significant energy release coincident with the event. Although evidence points against recriticality for these instances, it is worth pointing out that they were accompanied by small power range perturbations on the reactimeter. These perturbations are presently interpreted as being due to shielding variations caused by changing core water inventory, permitting fluctuations in gamma energy reaching the uncompensated detectors.

It may be concluded, simply on the basis of the available plant data, that recriticality was improbable. This is an important conclusion. However, it does not address the possiblility that there may have been a significant reduction in the margin that the reactor was shut down. This could have been the result of change in core configuration caused by damaging events in the accident sequence. The following discussion explores the question of whether or not a substantive change in the margin of reactor shutdown might have transpired.

Although recriticality is considered unlikely over the time interval of concern, there are legitimate questions which relate to the margin of shutdown. Comparing the source range recording against the base count rate, Figure 2b, it may be observed that the source range value is high. The high source range count rates persisted for some time and were confirmed with scale measurements by M. Shultz (TMI Industry Advisory Group) and R. Ball (B & W) on 4/19 and 4/25. Both readings were in the neighborhood of 25 cps.

A study of the source count rate decay curve was performed by H. Richings (USNRC).[2] To interpret the observed count rate, it is necessary to subtract off the fixed source neutron contribution which derives appreciably from the installed neutron sources. Since the count rate at TMI ultimately decayed to a constant 5 cps, this value can be used as the base count rate level.

Richings compared the actual decay curve with an expression built around a 12.8 day half life. The 12.8 day half life corresponds to the decay of Ba^{140}, which is the controlling factor in the La^{140}-D_2O photo-neutron production.

Richings' comparison over the time period 4/13 - 4/30 is shown in Figure 5. The close resemblance between the curve and count rate

* The comparison between the TMI time history and the nominal decay curve is based on reactor trip at full power of an Oconee nuclear unit, fitted to the TMI recording. A normal trip of TMI-2 at full power is not available.

data strongly suggests that photo-neutron production from the La^{140} decay governed the long-term decay process.

The study of source range count rate decay does not account for the high count rate levels that were observed. The high count rate may be due to one or a combination of three possibilities: greater source strength; greater neutron multiplication; increased detector efficiency. The variation in source strength can be ruled out on account of the fixed relationship between core power history and Ba^{140} production, which is invariant to subsequent core degradation.

The neutron multiplication factor was originally pursued by M. Shultz.(3) Essentially, the analysis compares the nominal count rate to observed counts for the equivalent source term, and nominal K_{eff}. That is:

$$K_{eff2} = 1 - \left(\frac{CR_1}{CR_2} \right) (1 - K_{eff1})$$

After 22 days (time of Shultz's analysis) the photo-neutron source count rate (CR_1) was certainly less than 5 cps. Nominal K_{eff} is estimated at 0.71. Therefore for an observed count rate of 25 cps, the actual K_{eff} must be greater than 0.94. This is indicative of a major change in shutdown margin. However, it is not suggestive of imminent recriticality

Evidence contrary to the reactivity theory was also developed by Shultz. In the period between 4/13 and 4/17 the primary system was deborated from 3400 ppm to 3000 ppm. This deboration should have introduced reactivity net worth in the neighborhood of 4.0% to 5.3% $\Delta K/K$. This is enough to have caused a significant variation in the count rate (enough in fact to achieve criticality if $K_{eff} \geq 0.95$). The fact that no variation in the source range count rate was observed suggests that the reactor was actually far subcritical.

The remaining possibility is that the source range detector efficiency was somehow changed. This line of reasoning postulates a significant release in Ba^{140} from the fuel into the coolant. The Ba^{140} is presumed soluble, decaying to soluble La^{140}. Some of the La^{140} finds its way into the downcomer annulus, producing photo-neutrons that are readily detected by source range instrumentation. The detector efficiency is increased in the sense that photo-neutrons have been physically moved (from the core) closer to the detector (e.g., the downcomer). On the debit side, however, is their incapability for neutron multiplication outside the core region.

H. Richings (NRC) has performed a scoping study of the downcomer $\gamma - n$ postulation, based upon a primary sample La^{140} activity (as of 4/11/79) of 150 mc/ml[2]. He concludes that detector efficiency for neutrons produced in the appropriate downcomer region must be on the order of 1.42×10^{-2}. This is considered rather high for the situation at hand.

Richings' work has been independently checked and a supplemental analysis has been performed to estimate photo-neutron production directly in the primary shield. It is not possible, using simplified analyses, to justify the high source range count rate. Although both analyses are based on primary sample Ba^{140} concentrations, there is no evident reason to think these concentrations are not representative of downcomer Ba^{140} content.

The evidence at hand suggests that source neutrons emitted directly from the downcomer may be the cause of high source range count rates, rather than caused by a variation in shutdown margin. The analysis is not conclusive, and refined calculations may be warranted. One consideration which should be borne in mind, however, is the fact that source range count rates ultimately dropped to the neighborhood of 5 cps. This is consistent with the base count rate which would be sustained by the two installed (AM-Be-Cm) neutron sources. The low count rate value that was ultimately reached means the reactor was sufficiently subcritical to start with, or somehow became that way by gradual insertion of negative reactivity. This would have had to be achieved at a 12.8 day half life, coincident with Ba^{140} decay -- an unlikely possibility.

TABLE I

CALCULATED K_{eff}, DETECTOR EFFICIENCY, SOURCE

AND COUNT RATE FOR HOMOGENEOUS VOIDINIG

STATE	K_{eff}	E*	SOURCE (NEUTRONS/SEC)	COUNT RATE
1030 ppm B Rods Crit	1.0	-	-	-
1030 ppm B Rods In	0.9368	1.2×10^{-10}	1.43×10^{11}	284
20% (Voids)	0.9097	4.9×10^{-8}	1.14×10^{11}	615
40% (Voids)	0.8582	2.3×10^{-9}	8.60×10^{10}	1430
60% (Voids)	0.7665	1.33×10^{-8}	5.74×10^{10}	3282
80% (Voids)	0.6146	1.31×10^{-7}	2.88×10^{10}	9791
100% (Voids)	0.4900	6.85×10^{-4}	2.05×10^{8}	2750

*Detector Efficiency is defined as the ratio of neutrons detected to the neutrons generated in the core.

REFERENCES

1. "TMI-2 Post Accident Criticality Analyses", GPU Services Technical Data Report 049, 31 August 1979.

2. H. Richings memo to K. Kriel (USNRC), "TMI-2 Source Range Detector Count Rate", 5/11/79

3. Industry Adivsory Group (IAG) memo, IA-23, "Examine The High Counting Rate of BF^3 Neutron Detector", M. Shultz, 4/20/79.

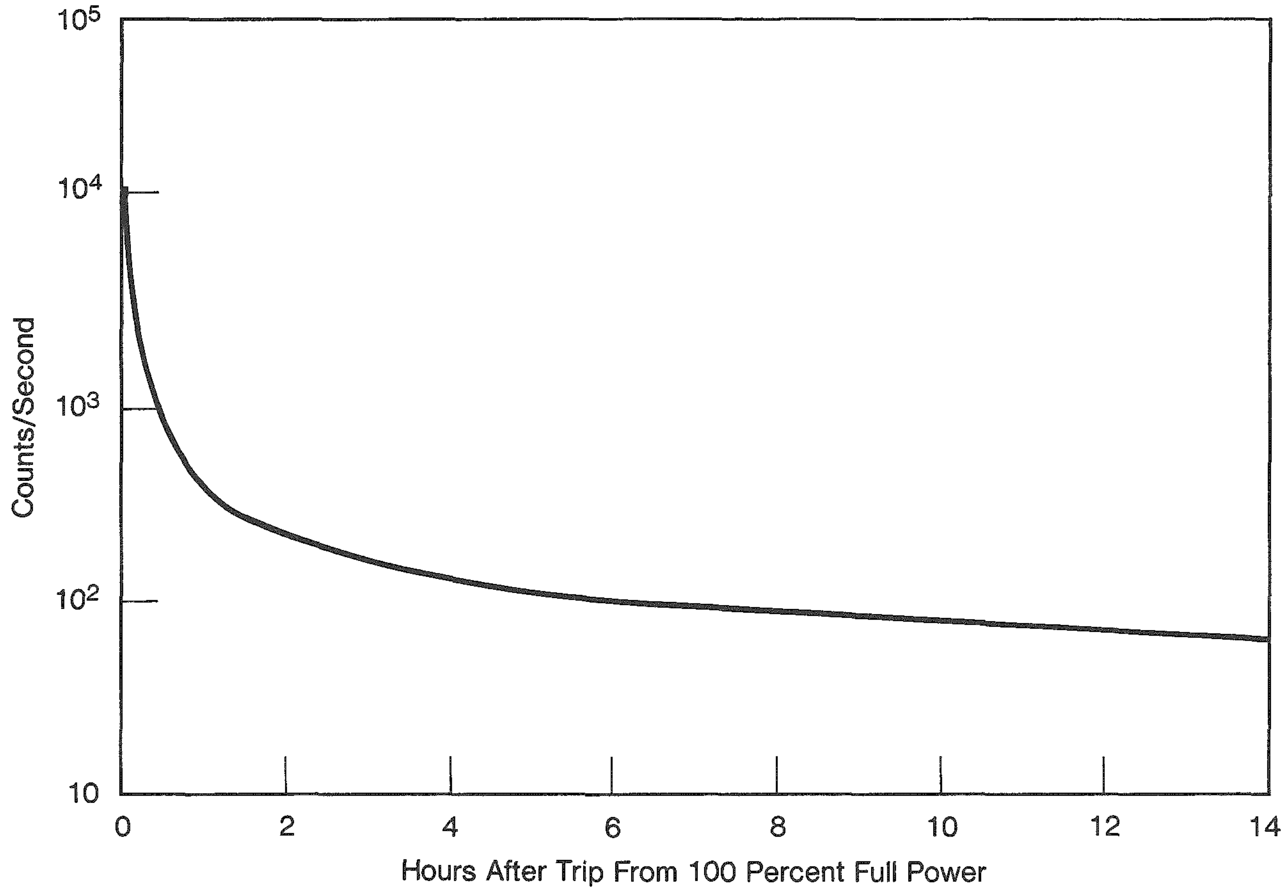


Figure RECRIT 1. Typical Source Range Power Decay Curve

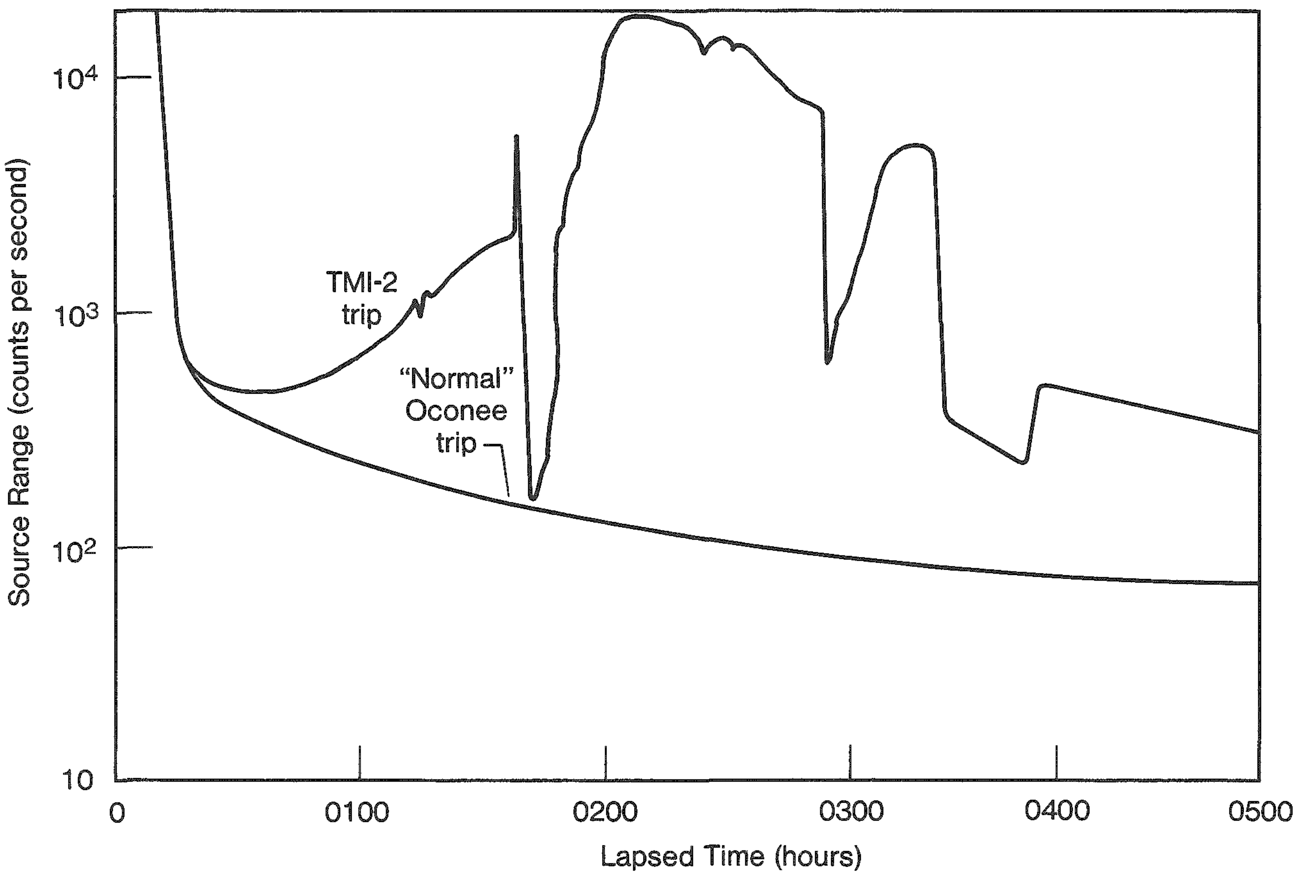


Figure RECRIT 2a. Short-term source range trace for TMI-2 (3/28/79).

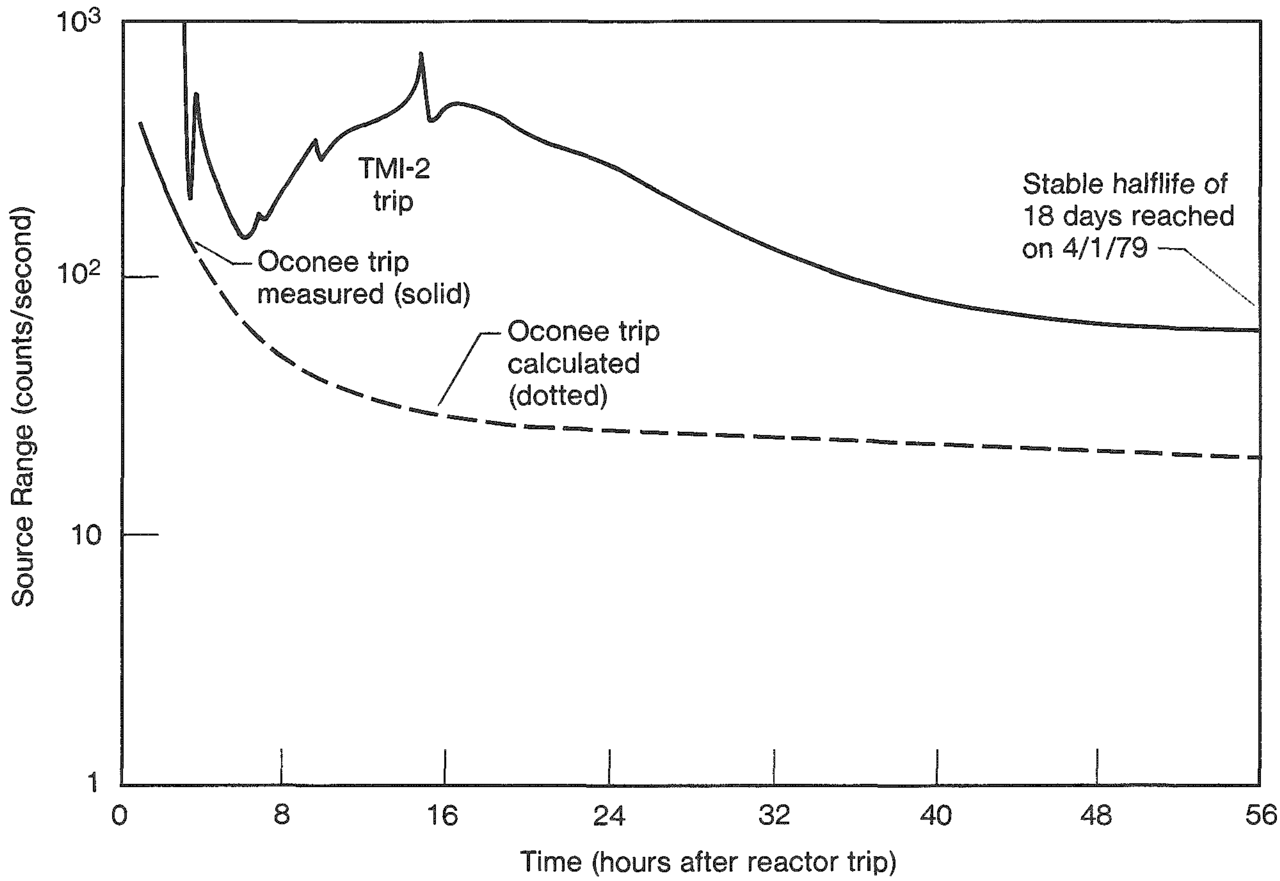


Figure RECRIT 2b. Long-term source range trace for TMI-2 (3/28/79–3/30/79).

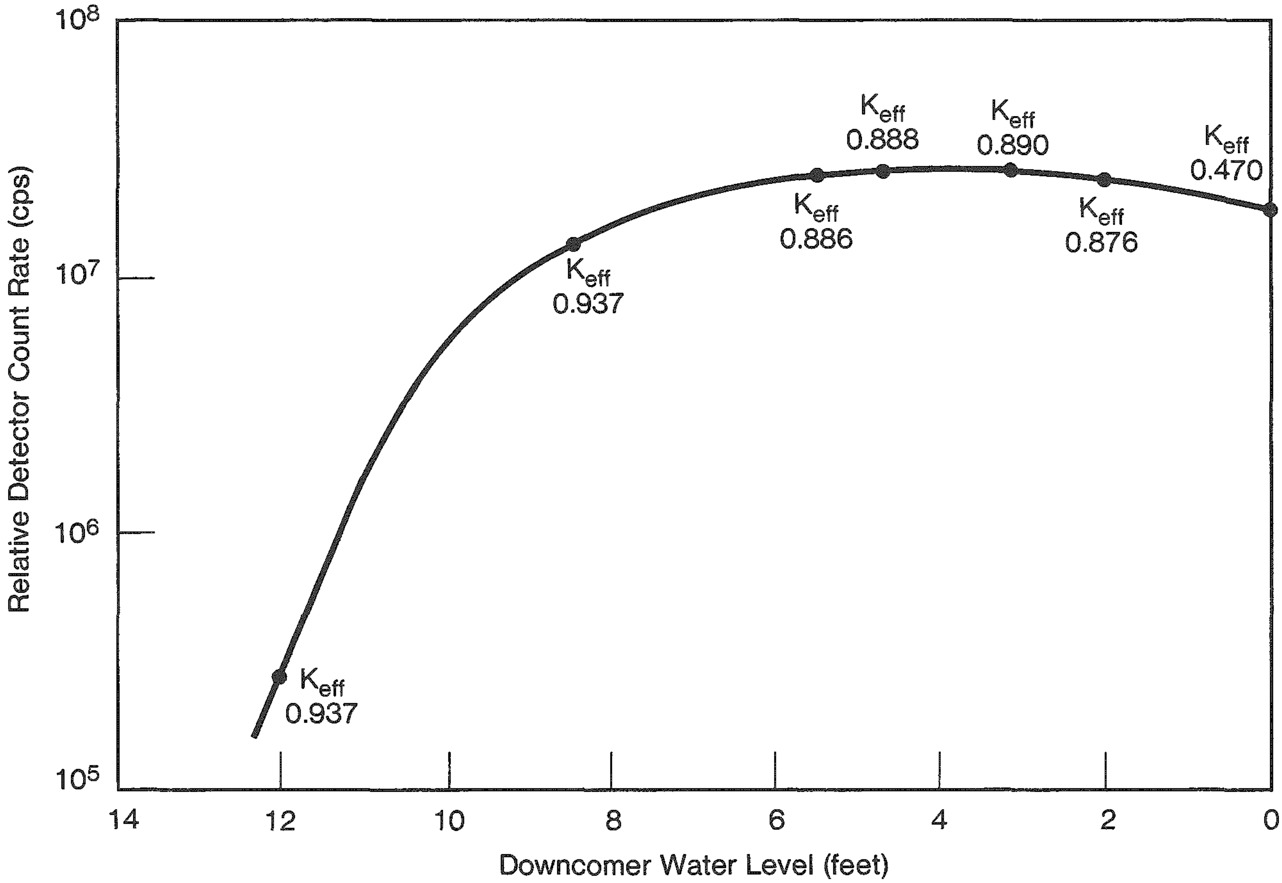


Figure RECRIT 3. Downcomer Water Level Versus Detector Count Rate

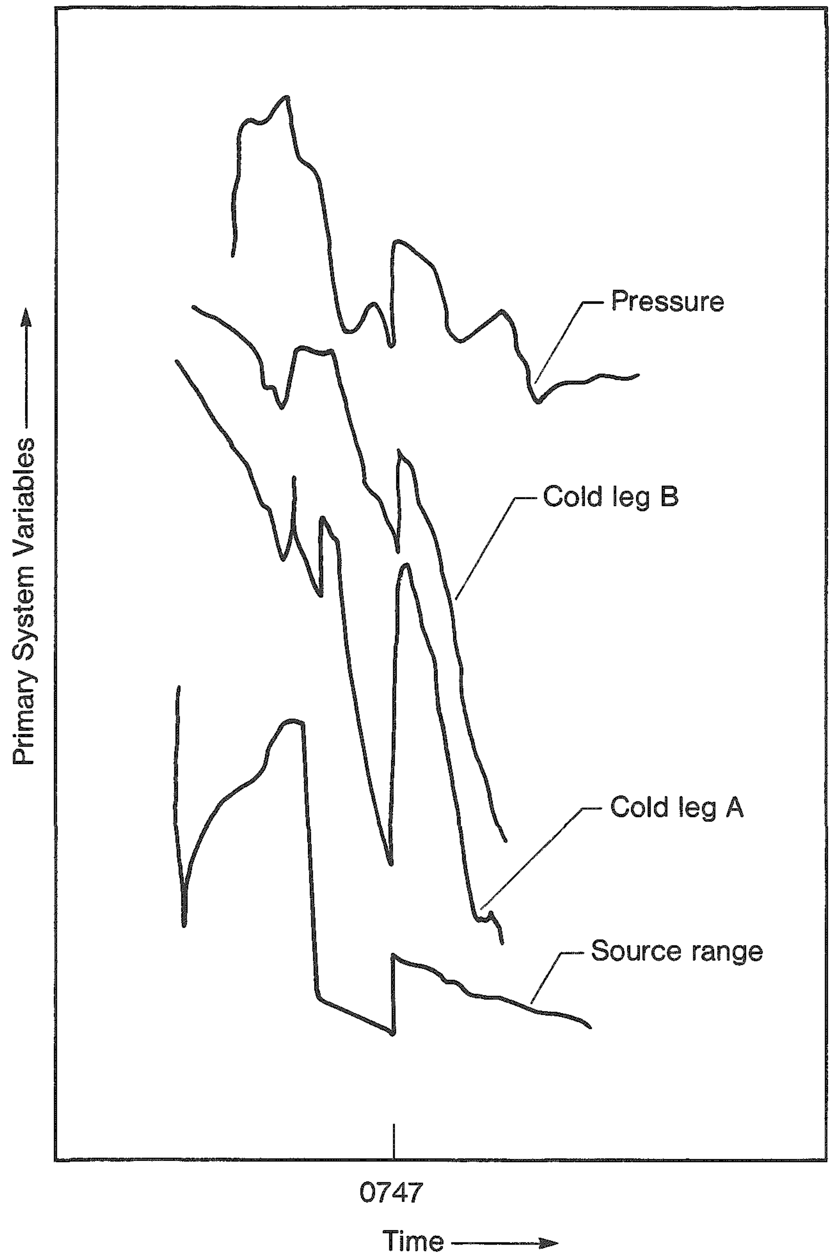


Figure RECRIT 4. System Response to 0747 Event

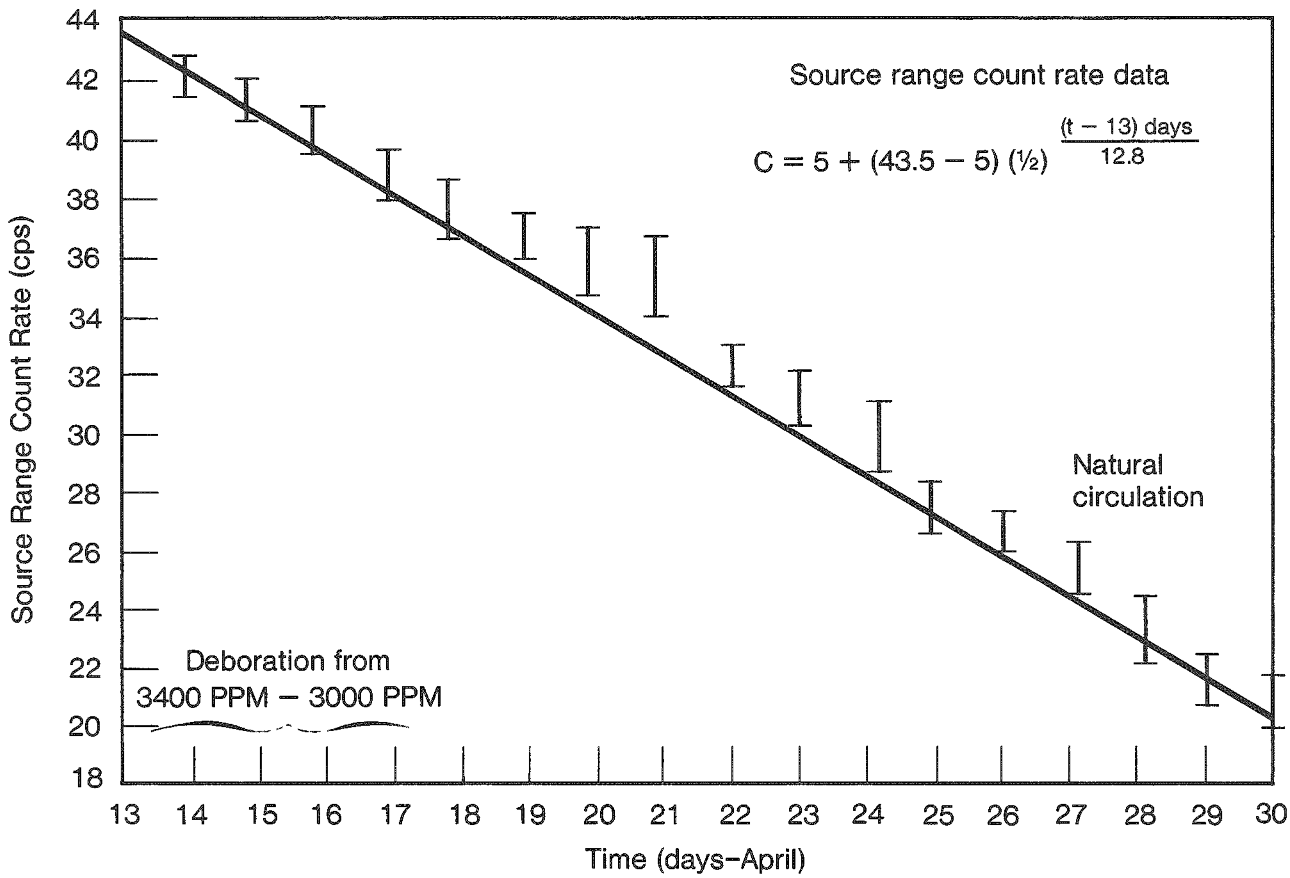


Figure RECRIT 5. Comparison Against Source Range Count Rate Data

Reference: H. Richings (USNRC/core perf. br/DDS)

APPENDIX RM

AREA RADIATION MONITORS

The area monitors are located throughout the reactor facility. The ones of interest are numbered HP-R-210, HP-R-212 (or 213) and HP-R-214 (A and B). HP-R-210 is located at the south end of the fuel handling bridge, see Figures 1b, 2a, and 2b. It is a GM tube type instrument covering a range of 10^{-1} mR/hr to 10^{4} mR/hr. HP-R-212 (or 213) is the same type of instrument as HP-R-210 and is located at the reactor building equipment hatch, see Figures 1a, 1b, 2b, and 2c. (Note: there is some uncertainty about the identity of HP-R-212. The markings on the chart are somewhat ambiguous because of poor inking. Some people have identified the channels for HP-R-212 and are calling these channels HP-R-213. Some of the responses observed during the first few hours tend to support one identification while other observations tend to support another. We can only conclude that either identification is possible.) HP-R-214, the dome monitor is an ionization chamber type instrument covering a range of 10^{-1} mR/hr to 10^{7} mR/hr when used in an unshielded configuration. However, at Three Mile Island-2 (TMI-2), it was enclosed in a shielded environmentally sealed configuration. The attenuation factor is in the published specifications for this type of monitor. For location of this monitor at TMI-2, see Figures 1b, and 2c.

Hardware Survivability

Specifications for the in-containment components show an upper limit of 140°F for temperature, 30 psig for pressure and 100% relative humidity for HP-R-210 and HP-R-212. For HP-R-214, the upper limits are 250°F, 50.5 psig and 100% humidity. There is no direct measure of the environmental conditions at the component locations but it is reasonable to assume that the instrument specifications may have been exeeded briefly during some part of the transient, especially the temperatures of HP-R-210. However, prior to the first up-scale swing, at about 140-146 minutes elapsed time, measurements of containment temperatures indicate only a border-line possibility of exceeding temperature specifications and, at the worst, should have been only decalibrated to a small degree (less than 10x). Additionally, up to this time, two of the instruments had been reading a background level which exhibited the usual statistical level variations typical of a healthy, functioning channel. Therefore, it is reasonable to assume that at this early time period of interest, the monitors were supplying reasonably accurate information on the gamma radiation in the containment over the designed energy range 100 keV to 1.5 MeV.

Chart Time Calibration

The timing of fission product release is strongly dependent on proper synchronization of chart time with real time. An operator made a note on the chart which he judged to be at 7:00 a.m. The radiation monitors responded to several events during the time period 4:00 a.m. to 7:30 a.m., the initial phases of the ccident, which tend to corroborate the operator annotations. All three of the channels were initially (prior to 4:00 a.m.) recording normal upscale background levels (levels that had persisted for several days before the accident). One read 0.4 mR/hr, another read 0.6 mR/hr and the third read 100 mR/hr. Readings on HP-R-213 and HP-R-214 dropped abruptly at ~4:00 a.m. to new levels (see Fig. 3). The original higher background levels are thought to be from activation products such as ^{16}N or direct shine effects. In any event the decreases seem to coincide well with the scram at 4:00 a.m. Furthermore, when one looks forward in time in the charts at the rate of 8 inches per hour, the operator's annotation is in error by only 6 minutes, i.e., the notation of 7:00 should have been 6:54 instead. Further verification comes later in the event when opening and closing the relief block valve (ahead of the ERV) at 7:13 and 7:17, respectively, coincides very closely to the annotated chart time. Based upon the above correlations it is believed that the chart time can be synchronized with clock time to an accuracy of ± 1 or 2 minutes. In summary, we are confident that the chart speed was 8 inches per hour and the times 4:00 a.m., 6:54 a.m. and 7:13 a.m. are well established.

Indication of Release

The nuclide most likely to be present in the containment following loss of fuel rod integrity is ^{133}Xe, which emits an 80 keV gamma ray. The response of the monitors having GM tubes will be down approximately 20 to 30% at 80 keV due to the presence of the a 0.020-in. lead shield surrounding the detector whose purpose is to flatten the response in the 100 keV to 1.5 MeV range. Some simplified calculations have shown that the ^{133}Xe released from only a few rods (<10) will drive the containment area monitors upscale significantly. From this it is concluded that although the absolute calibrations may be in error due to minor environmental excesses, the area monitors would have responded rapidly to a few fuel pin failures if they release 1% of their ^{133}Xe content to the containment. As has been mentioned elsewhere, some of the channels began their upswing form a background level of a few tenths mR/hr and some began from off-scale. Even so, all began their upswing as approximately the same time, 6:25 a.m. (145 minutes elapsed time).

NOTE: Two curves are given for the dome monitor (HP-R-214). Curve A was proposed on the assumption that the span of HP-R-214 covers the full 10 inches of HP-UR-1901 chart paper. A calibration performed after the incident at TMI-2 suggested that the span of the instrument only covers 8 inches, i.e., full range of 10^7 mR/hr occurred at 3.8 decades on HP-R-1901 chart. Since such a shift in calibration could have occurred immediately after the referenced incident, there is no way of determining which calibration applied at the time of the incident. These curves may be considered as upper and lower bounds.

Looking later at the response of HP-R-210 and 214 to operation of the relief block valve at 7:13 a.m., it is noted that the response follows by about 3 minutes. The response to relief block valve closure at 7:17 is also delayed by 2 to 3 minutes. This indicated rapid response provides an estimate of the dispersion time from the core region throughout the containment building. The probable response time from the drain tank to the monitor ranges from 30 seconds to 1-1/2 minutes. Probable holdup time from the core to the dump tank, via the open ERV, is estimated to be less than this same increment. Therefore, total flight time from a rod failing until its fission products (from the gap region) can reach the monitors should be less than 5 minutes but greater than 2 minutes.

Also note HP-R-212 started up scale about one minute later due to its position near the drain tank, considered the source of the radiation; curves for this response are represented by Figures 4.

From the above analysis it is concluded that no significant number of fuel pin ruptures occurred before approximately 6:20 a.m., or 140 minutes elapsed time. We believe this conclusion is supported by the fact that the ERV block valve was not closed until 6:22 a.m., and any significant number of fuel pin failures before this time would have been observed on the area monitors. It therefore appears that initial failure was independent of, but nearly coincidental with, ERV block valve closure.

<u>Significance of Magnitude of the Signals on the Area Monitors</u>

It is tempting to try to quantify the pin failures as a function of time by comparing the readings of the containment area monitors in mR/hr to the predicted increase in dose rate in the containment of 100 mR/hr per pin failure. Although such analyses can provide broad scoping information there are too many uncertainties to permit drawing detailed conclusions. Some uncertainties are the unknown state of calibration of the instruments at the start of the accident, effects of the severe environmental conditions during the accident, etc. Also, contributions from the noble gas group can dominate the response in the short term. Other factors can change the absolute response in the dome monitor such as the effect of the thick lead shield on the softer spectrum gammas. Also, direct shine from the core and other major components cannot be properly factored into the response. For qualitative information, the response of the three area monitors considered are given in Figure 5.

SPECIAL NOTE: Another area monitor's response is shown as a dotted line on Figures 4 and 5. We have not been able to identify this instrument because of chart inking problems. However, its low, early response is consistent with what might be anticipated from the nuclear building personnel hatch monitor if ventilating air mixing in that area is poor.

APPENDIX RM

LIST OF RADIATION MONITORS AND LOCATIONS

PROCESS MONITORS

Monitoring Point	Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns & Roe Drawing Number	Location on Drawing	Plant Elevation
Primary Coolant Letdown HI	HP-UR-3264	7	1	MU-R-720	2066	AE/A63.5	305'
Primary Coolant Letdown LO	HP-UR-3264	7	2	MU-R-720	2066	AE/A63.5	305'
Intermediate Coolant Letdown Cooler B	HP-UR-3264	7	3	IC-R-1091	2060	R11	282' 6"
Intermediate Coolant Letdown Cooler A	HP-UR-3264	7	4	IC-R-1092	2060	R11	282' 6"
Intermediate Coolant Letdown Cooler Outlet	HP-UR-3264	7	5	IC-R-1093	2066	AB/A62.8	305'
Plant Effluent Unit II	HI-UR-3264	7	6	WDL-R-1311			
Decay Heat Closed A Loop	HP-UR-3264	7	7	DC-R-3399	2065	AK/A67	280' 6"
Decay Heat Closed B Loop	HP-UR-3264	7	8	DC-R-3400	2065	AK/A67	280' 6"
Nuc. Serv. Closed Cooling	HI-UR-3264	7	9	DC-R-3401	2066	AE/A61	305'
Spent Fuel Cooling	HP-UR-3264	7	10	SF-R-3402	2066	AN/A64.7	305'

APPENDIX RM

LIST OF RADIATION MONITORS AND LOCATIONS

AREA MONITORS

Area Monitored	Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns & Roe Drawing Number	Location on Drawing	Plant Elevation
Control Room	HP-UR-1901	1	1	HP-R-201	2381	CB/C48	341'
Cable Room	HP-UR-1901	1	2	HP-R-202	2380	CC/C48	305'
Emerg. Cooling Booster Pump	HP-UR-1901	1	3	HP-R-204	2065	AB/A61	280'
R.C. Evap. Control Panel Area	HP-UR-1901	1	4	HP-R-205	2065	AG/A63	280'
Makeup Tank Area	HP-UR-1901	1	5	HP-R-206	2066	AG/A64	305'
Intermediate Cooling Pump Area	HP-UR-1901	1	6	HP-R-207	2066	AA/A63	305'
Fuel Handling Bridge North	HP-UR-1901	1	7	HP-R-209	2064		355'
Fuel Handling Bridge South	HP-UR-1901	1	8	HP-R-210	2064	See	355'
R.B. Personnel Access Hatch	HP-UR-1901	1	9	HP-R-211	2064	Appendix RM	310'
R.B. Equipment Hatch	HP-UR-1901	1	10	HP-R-212	2064	Figures 1	314'
Incore Instrument Panel Area	HP-UR-1901	1	11	HP-R-213	2062	and 2	371'
Reactor Building Dome	HP-UR-1901	1	12	HP-R-214	2064		372'
Fuel Handling Bridge	HP-UR-1902	2	1	HP-R-215	2068	AM/A66.5	347'
Waste Disposal Storage Area	HP-UR-1902	2	2	HP-R-218	2066	AR/A67	305'
Aux Bldg Sump Tank Filer Room	HP-UR-1902	2	3	HP-R-231	2066	AQ/A62	280'
Aux Bldg Access Corridor	HP-UR-1902	2	4	HP-R-232	2066	AR/A61	305'
Aux Bldg Access Corridor	HP-UR-1902	2	5	HP-R-233	2066	AN/A63	305'
Aux Bldg Access Corridor	HP-UR-1902	2	6	HP-R-234		CE/C50A	280'
RB Purge Unit Area	HP-UR-1902	2	7	HP-R-3236	2067	AE/A63	328'
Aux Bldg Exh. Unit Area	HP-UR-1902	2	8	HP-R-3238	2067	AJ/A63	328'
Fuel Handling Exh. Unit Area	HP-UR-1902	2	9	HP-R-3240	2067	AL/A63	328'

APPENDIX RM

LIST OF RADIATION MONITORS AND LOCATIONS

AIRBORNE MONITORS

Monitoring Point	Stripchart	Recorder Number	Channel Number	Recorder Designator	Burns & Roe Drawing Number	Location on Drawing	Plant Elevation	Monitored Variable
Station Vent	HP-UR-1907	4	1	HP-R-219	2067	AF/A65	328'	Particulate
Station Vent	HP-UR-1907	4	2	HP-R-219	2067	AF/A65	328'	Iodine
Station Vent	HP-UR-1907	4	3	HP-R-219	2067	AF/A65	328'	Gas
Control Room Intake	HP-UR-1907	4	4	HP-R-220	2381	CD/C48	351' 6"	Particulate
Control Room Intake	HP-UR-1907	4	5	HP-R-220	2381	CD/C48	351' 6"	Iodine
Control Room Intake	HP-UR-1907	4	6	HP-R-220	2381	CD/C48	351' 6"	Gas
Fuel Handling Bldg. Exh. Upstream of Filter	HP-UR-1907	4	7	HP-R-221A	2067	AT/A63	328'	Particulate
Fuel Handling Bldg. Exh. Upstream of Filter	HP-UR-1907	4	8	HP-R-221A	2067	AT/A63	328'	Iodine
Fuel Handling Bldg. Exh. Upstream of Filter	HP-UR-1907	4	9	HP-R-221A	2067	AT/A63	328'	Gas
Fuel Handling Bldg Exh. Downstream of Filter	HP-UR-1907	4	10	HP-R-221B	2067	AT/A63	328'	Particulate
Fuel Handling Bldg Exh. Downstream of Filter	HP-UR-1907	4	11	HP-R-221B	2067	AT/A63	328'	Iodine
Fuel Handling Bldg Exh. Downstream of Filter	HP-UR-1907	4	12	HP-R-221B	2067	AT/A63	328'	Gas
Hydrogen Purge	HP-UR-1907	4	13	HP-R-229	2067	AF/A65	328'	Particulate
Hydrogen Purge	HP-UR-1907	4	14	HP-R-229	2067	AF/A65	328'	Iodine
Hydrogen Purge	HP-UR-1907	4	15	HP-R-229	2067	AF/A65	328'	Gas
RB Purge Air Exhaust Duct A	HP-UR-2900	5	1	HP-R-225	2067	AB/A64	328'	Particulate
RB Purge Air Exhaust Duct A	HP-UR-2900	5	2	HP-R-225	2067	AB/A64	328'	Iodine
RB Purge Air Exhaust Duct A	HP-UR-2900	5	3	HP-R-225	2067	AB/A64	328'	Gas
RB Purge Air Exhaust Duct B	HP-UR-2900	5	4	HP-R-226	2067	AB/A64.5	328'	Particulate
RB Purge Air Exhaust Duct B	HP-UR-2900	5	5	HP-R-226	2067	AB/A64.5	328'	Iodine
RB Purge Air Exhaust Duct B	HP-UR-2900	5	6	HP-R-226	2067	AB/A64.5	328'	Gas
Aux Bldg Purge Air Exh. Upstream of Filter	HP-UR-2900	5	7	HP-R-222	2067	AT/A63	328'	Particulate
Aux Bldg Purge Air Exh. Upstream of Filter	HP-UR-2900	5	8	HP-R-222	2067	AT/A63	328'	Iodine
Aux Bldg Purge Air Exh. Upstream of Filter	HP-UR-2900	5	9	HP-R-222	2067	AT/A63	328'	Gas
Aux Bldg Purge Air Exh. Downstream of Filter	HP-UR-2900	5	10	HP-R-228	2067	AT/A63	328'	Particulate
Aux Bldg Purge Air Exh. Downstream of Filter	HP-UR-2900	5	11	HP-R-228	2067	AT/A63	328'	Iodine
Aux Bldg Purge Air Exh. Downstream of Filter	HP-UR-2900	5	12	HP-R-228	2067	AT/A63	328'	Particulate
Reactor Building Air Sample	HP-UR-3236	6	1	HP-R-227	2066	AB/A63	328'	Particulate
Reactor Building Air Sample	HP-UR-3236	6	2	HP-R-227	2066	AB/A63	328'	Iodine
Reactor Building Air Sample	HP-UR-3236	6	3	HP-R-227	2066	AB/A63	328'	Gas
Waste Gas Discharge Duct	HP-UR-3236	6	4	WGD-R-1480	2067	AB/A62.5	328'	Gas
WDG-T-1A Waste Gas Decay Tank Discharge	HP-UR-3236	6	5	WGD-R-1485	2066	AG/A62.5	305'	Gas
WDG-T-1B Waste Gas Decay Tank Discharge	HP-UR-3236	6	6	WGD-R-1486	2066	AG/A62.5	305'	Gas
Condenser Vacuum Pump Discharge	HP-UR-3236	6	7	VA-R-748				Gas

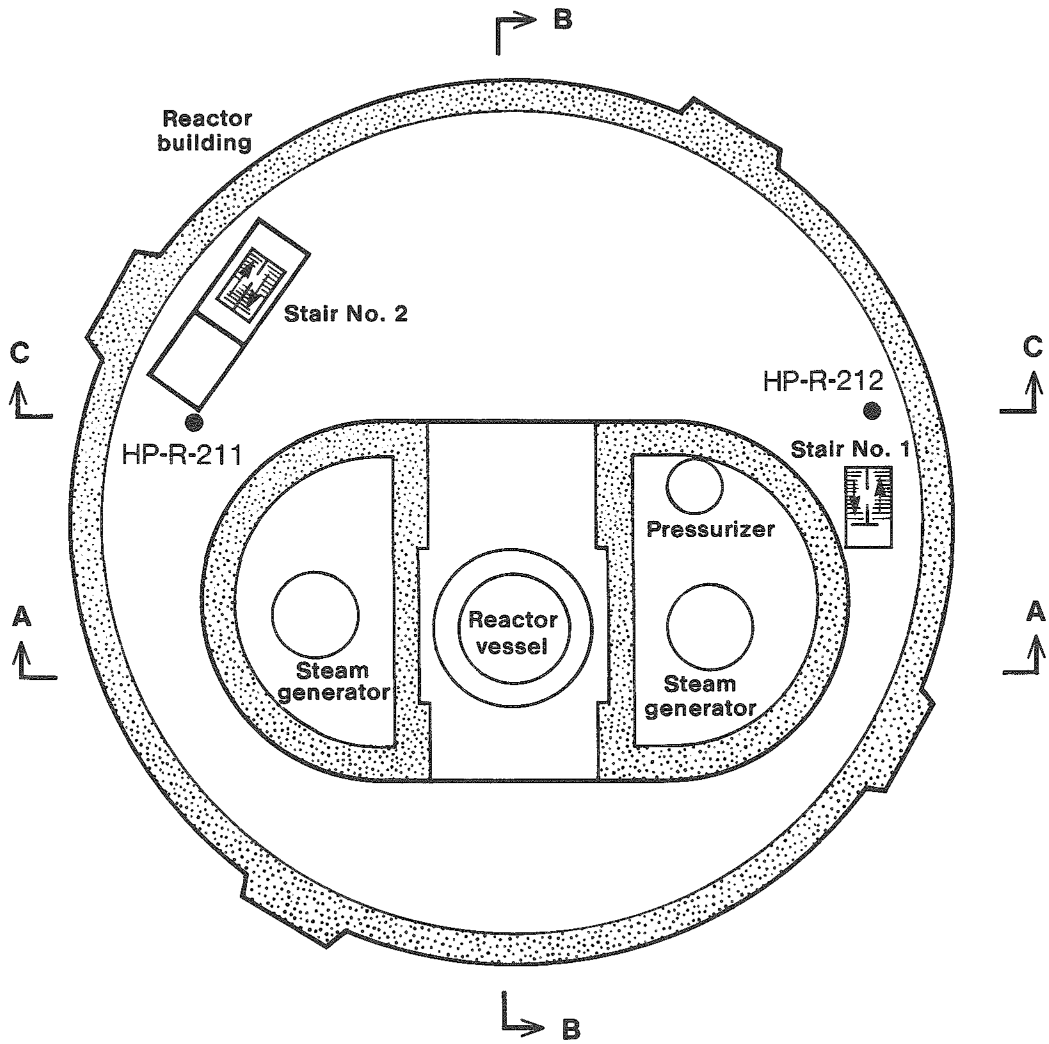


Figure RM-1a. Three Mile Island – Unit 2 reactor building locations of radiation monitors at level 305 feet.

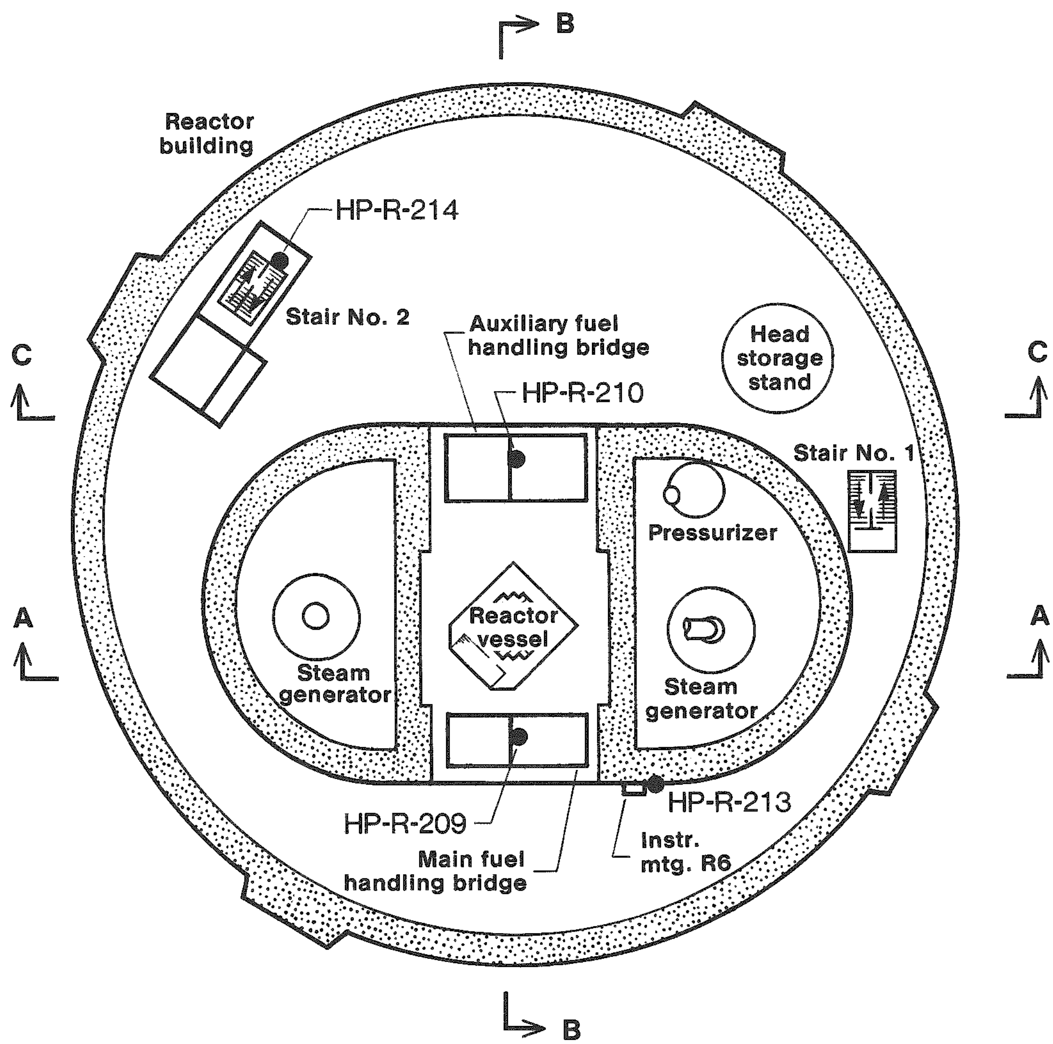


Figure RM-1b. Three Mile Island – Unit 2 reactor building locations of radiation monitors at level 347.5 feet.

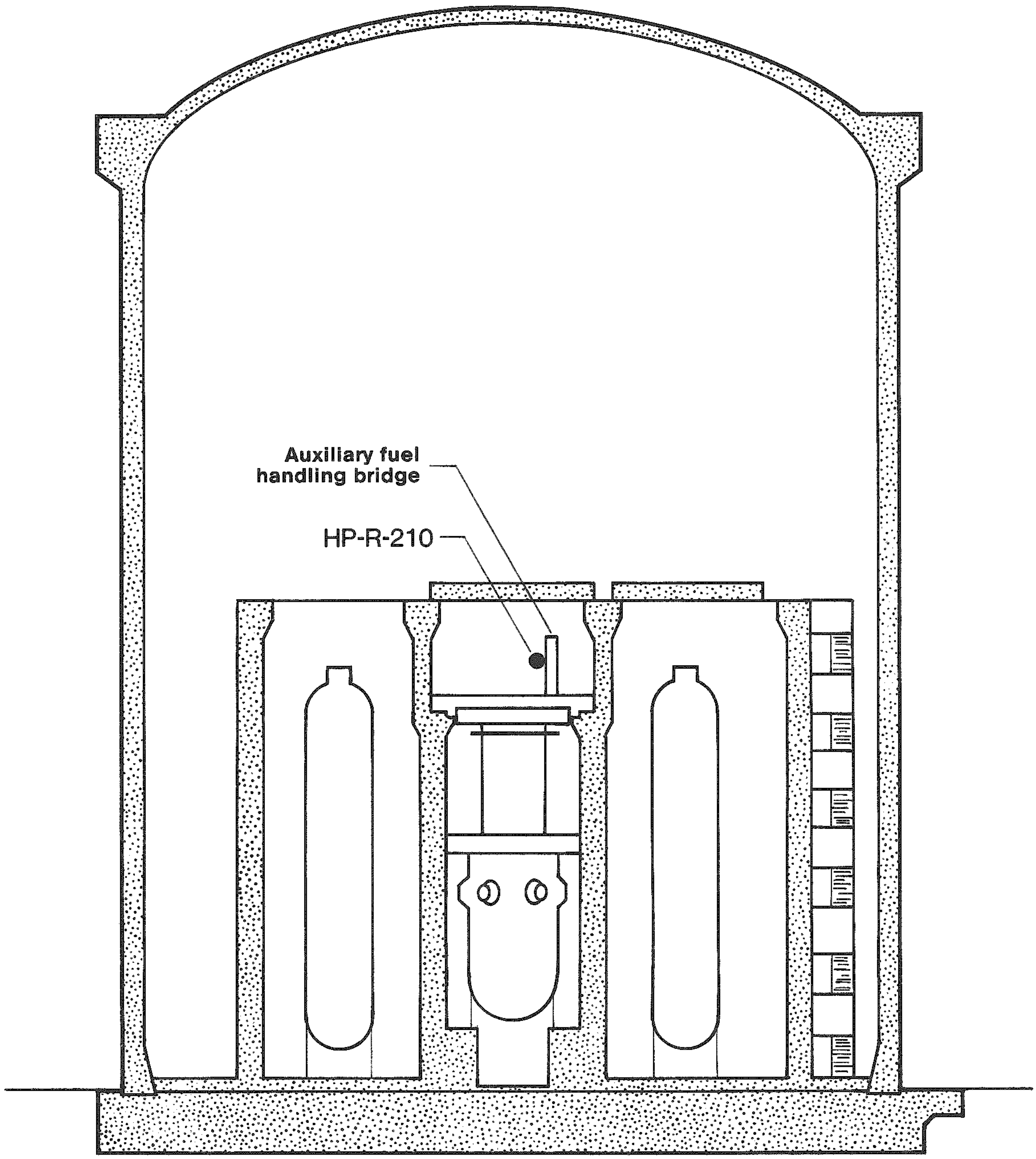


Figure RM-2a. Three Mile Island – Unit 2 building locations of radiation monitors along Section A-A.

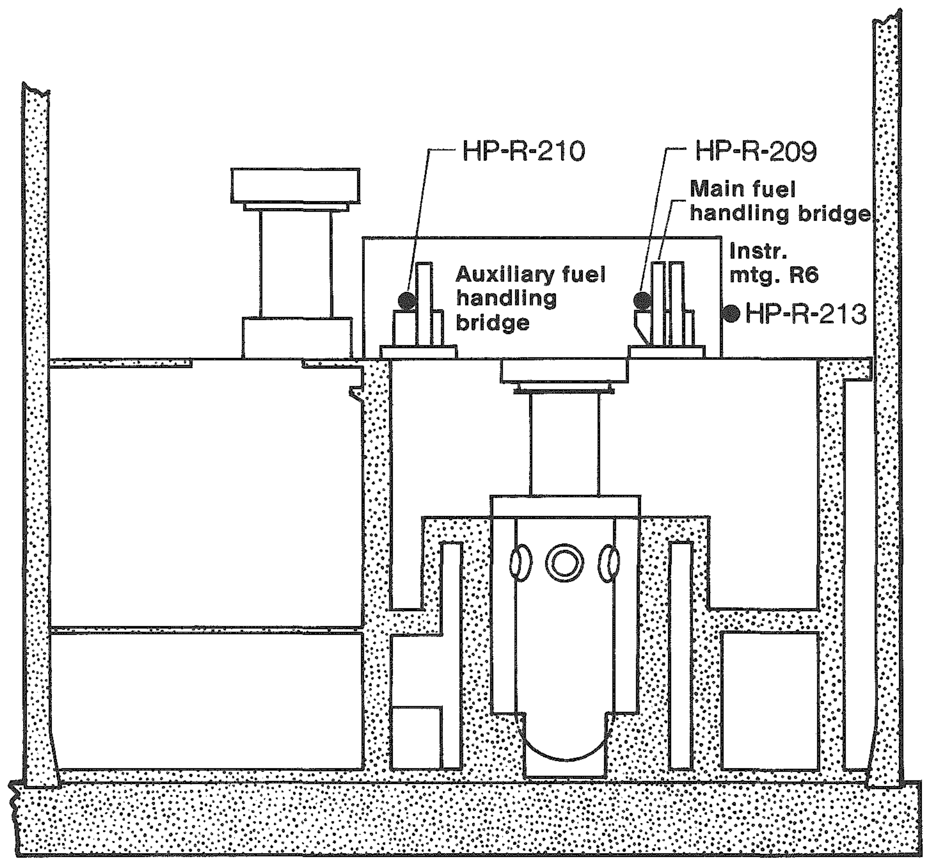


Figure RM-2b. Three Mile Island – Unit 2 reactor building locations of radiation monitors along Section B-B.

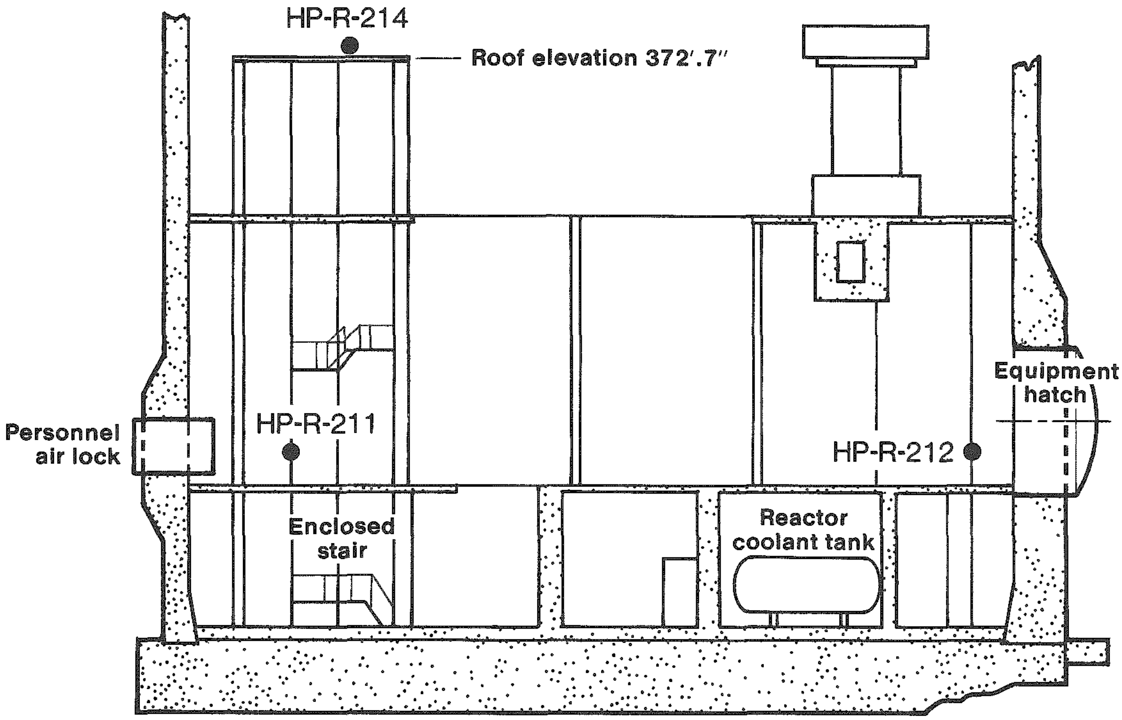


Figure RM-2c. Three Mile Island – Unit 2 reactor building locations of radiation monitors along Section C-C.

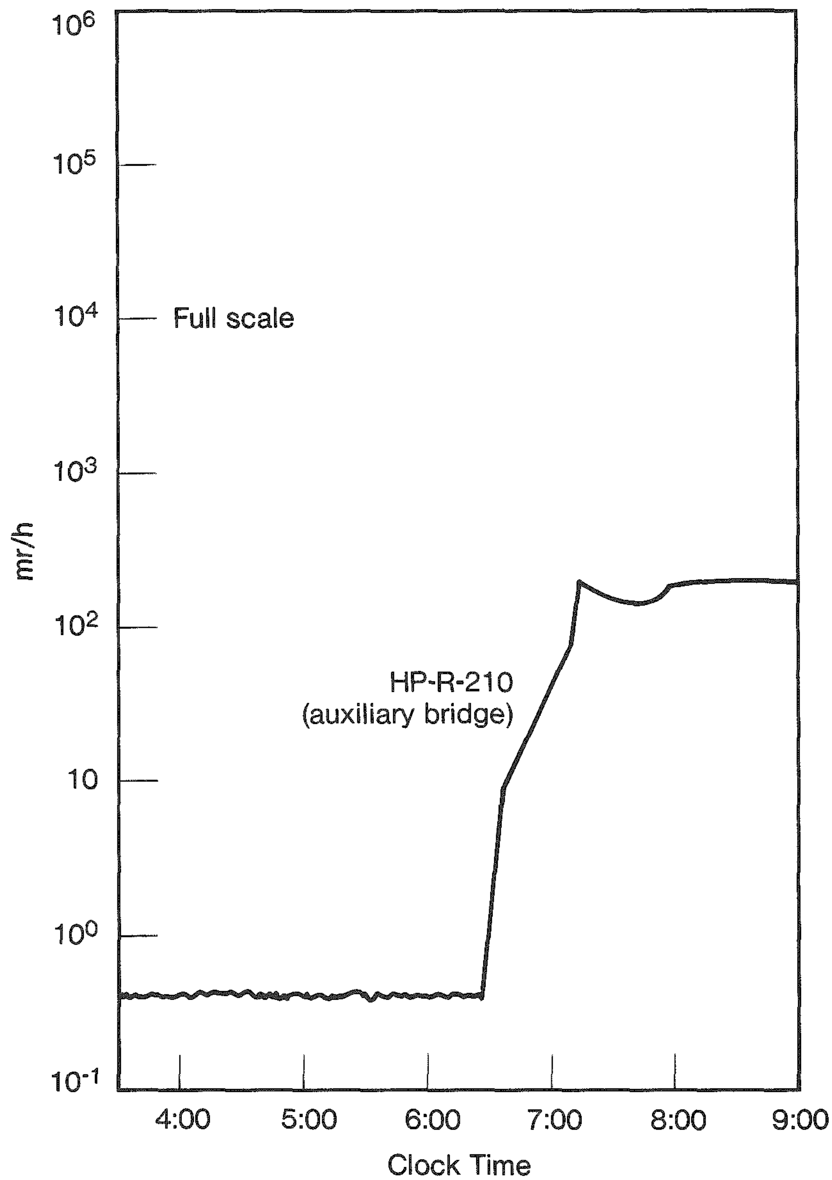
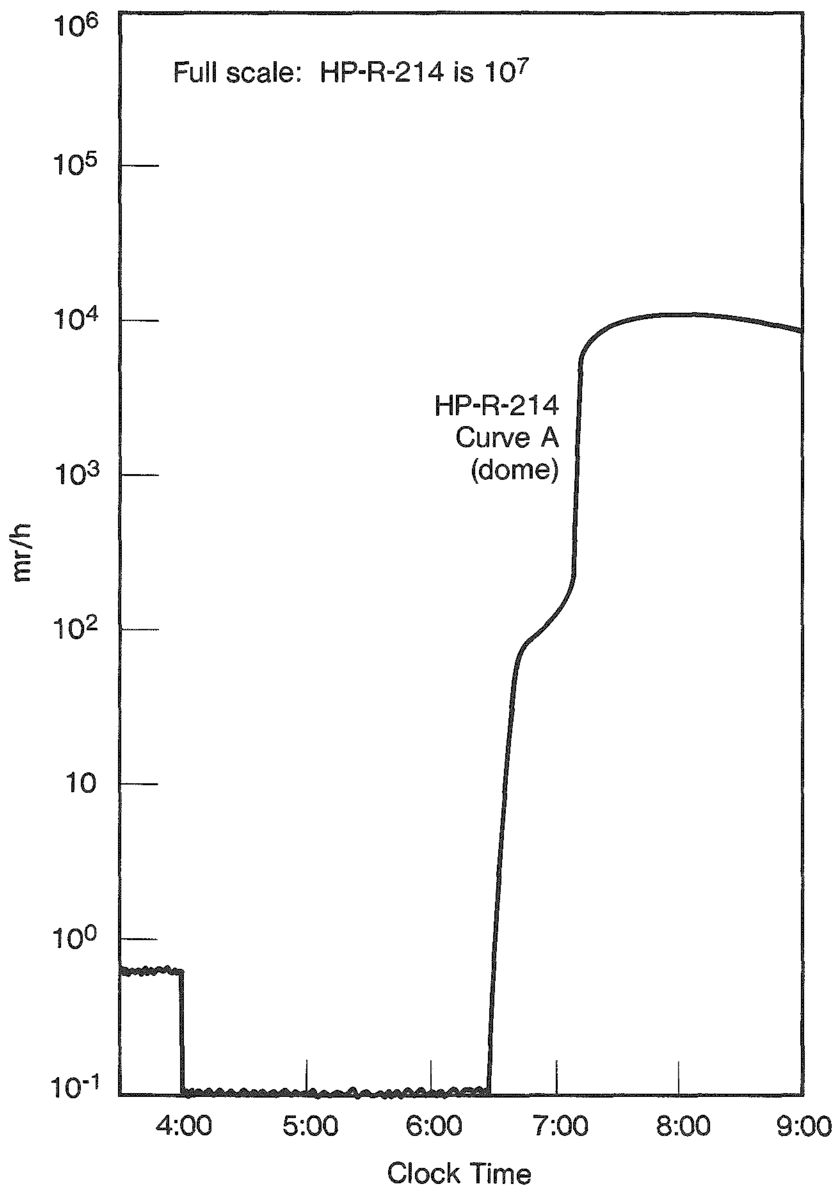


Figure RM-3a. Three Mile Island – Unit 2 area personnel monitors response.



Full scale: HP-R-214 is 10^7
HP-R-214
Curve A
(dome)
mr/h
10^6
10^5
10^4
10^3
10^2
10
10^0
10^{-1}
4:00
5:00
6:00
7:00
8:00
9:00
Clock Time

Figure RM-3b.

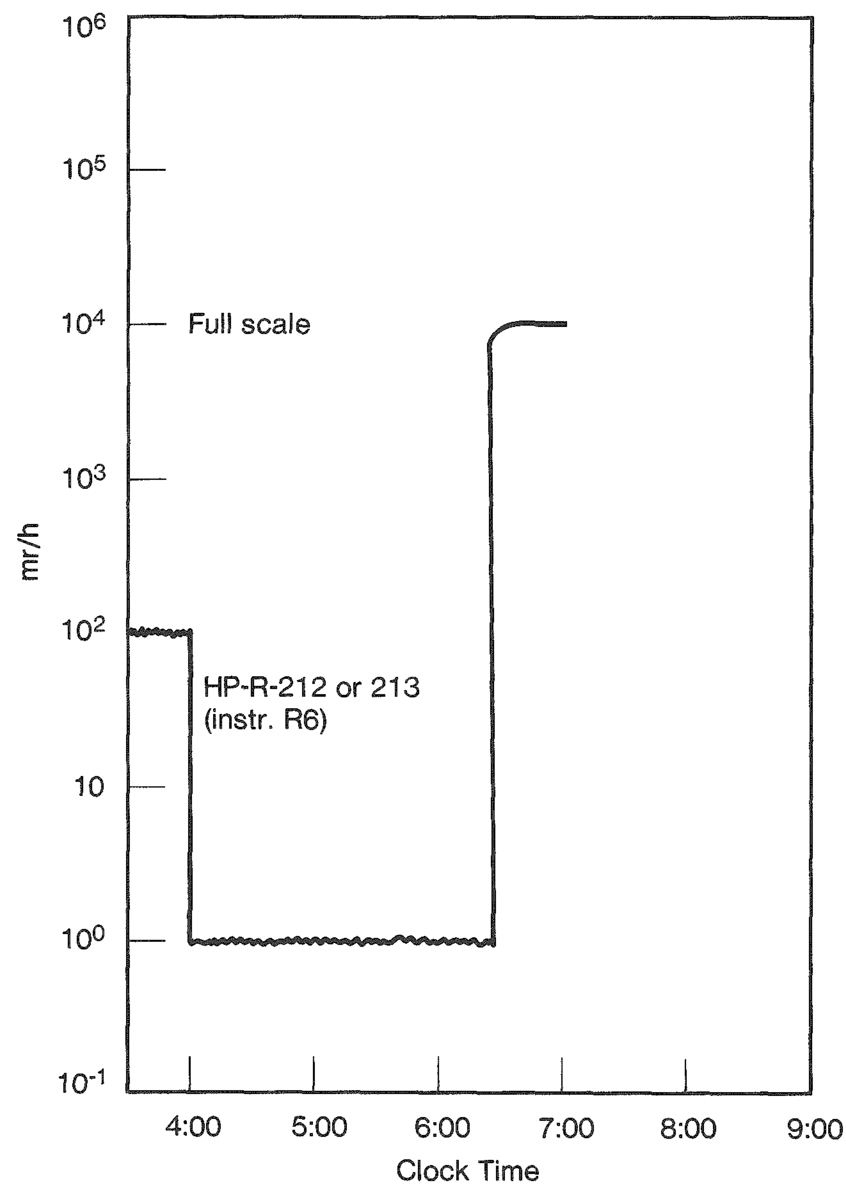

10^6
10^5
10^4
Full scale
10^3
mr/h
10^2
HP-R-212 or 213
(instr. R6)
10
10^0
10^{-1}
4:00
5:00
6:00
7:00
8:00
9:00
Clock Time

Figure RM-3c.

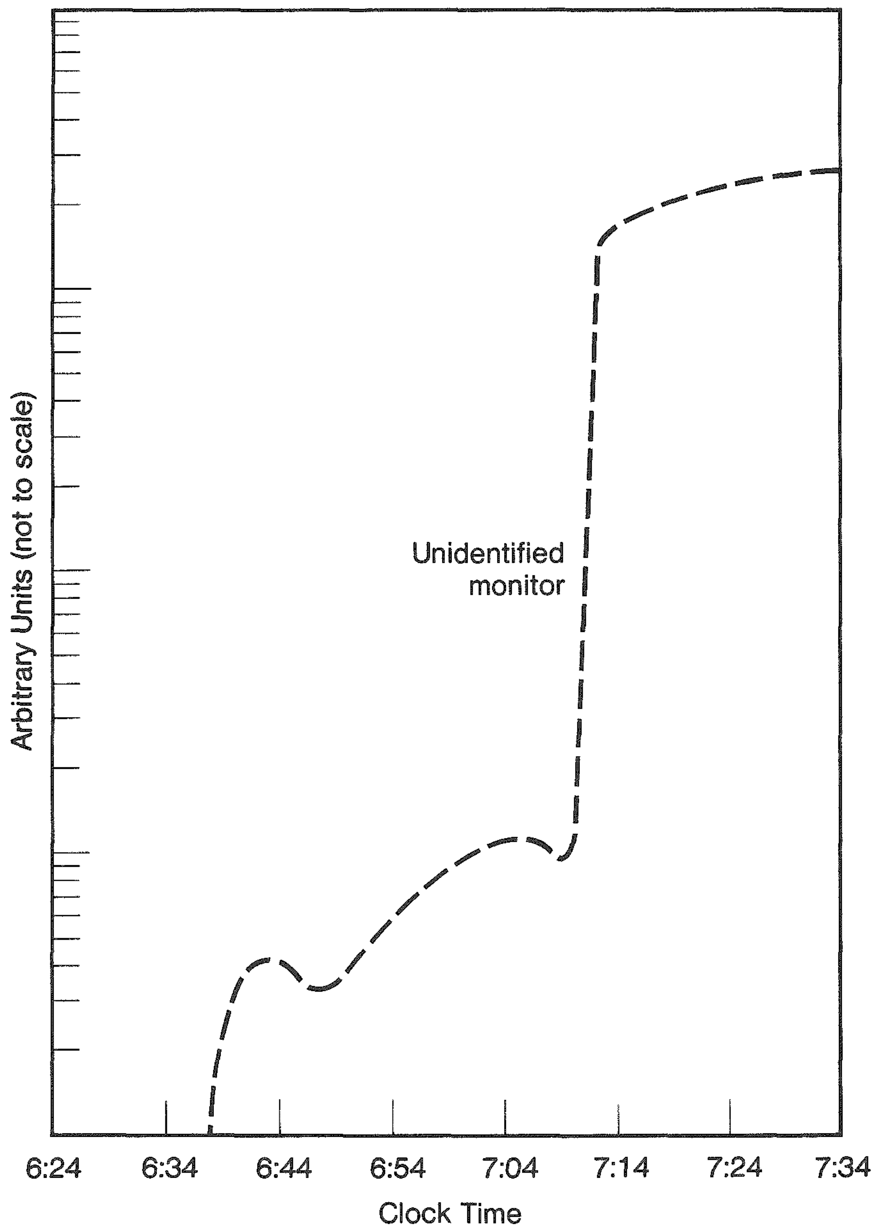


Figure RM-4a. Three Mile Island – Unit 2 beginning upscale swing on containment area monitors.

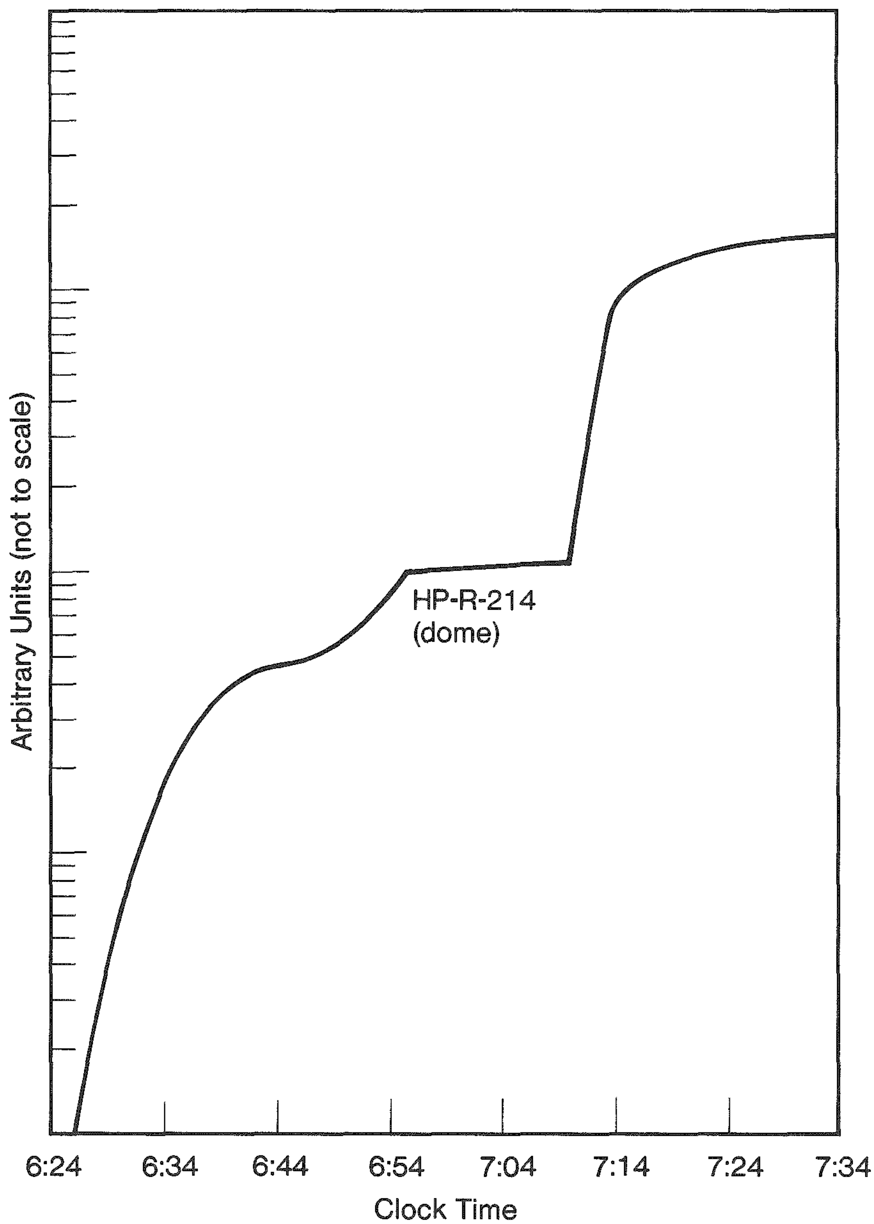

Arbitrary Units (not to scale)
HP-R-214
(dome)
6:24
6:34
6:44
6:54
7:04
7:14
7:24
7:34
Clock Time

Figure RM-4b.

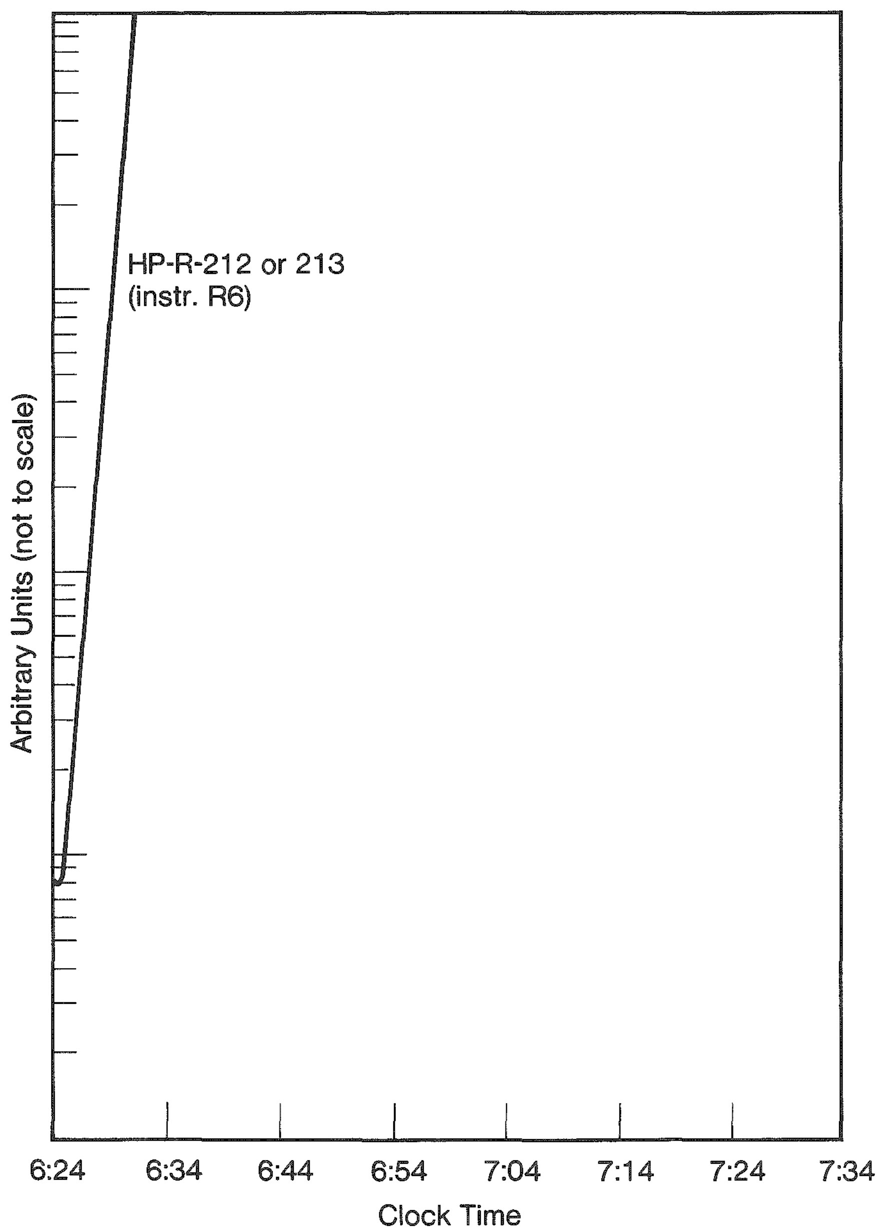
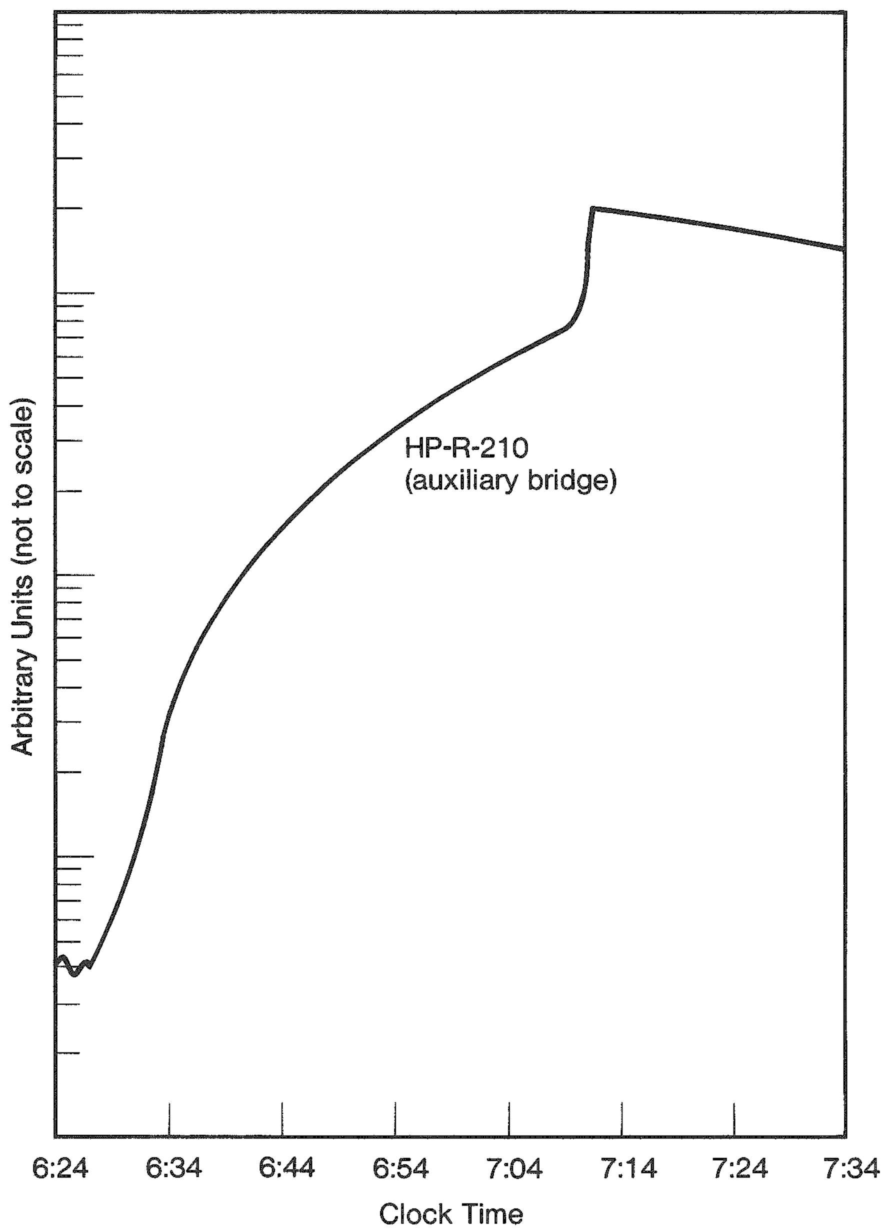

HP-R-212 or 213
(instr. R6)
Arbitrary Units (not to scale)
6:24
6:34
6:44
6:54
7:04
7:14
7:24
7:34
Clock Time

Figure RM-4c.



HP-R-210
(auxiliary bridge)
Arbitrary Units (not to scale)
6:24
6:34
6:44
6:54
7:04
7:14
7:24
7:34
Clock Time

Figure RM-4d.

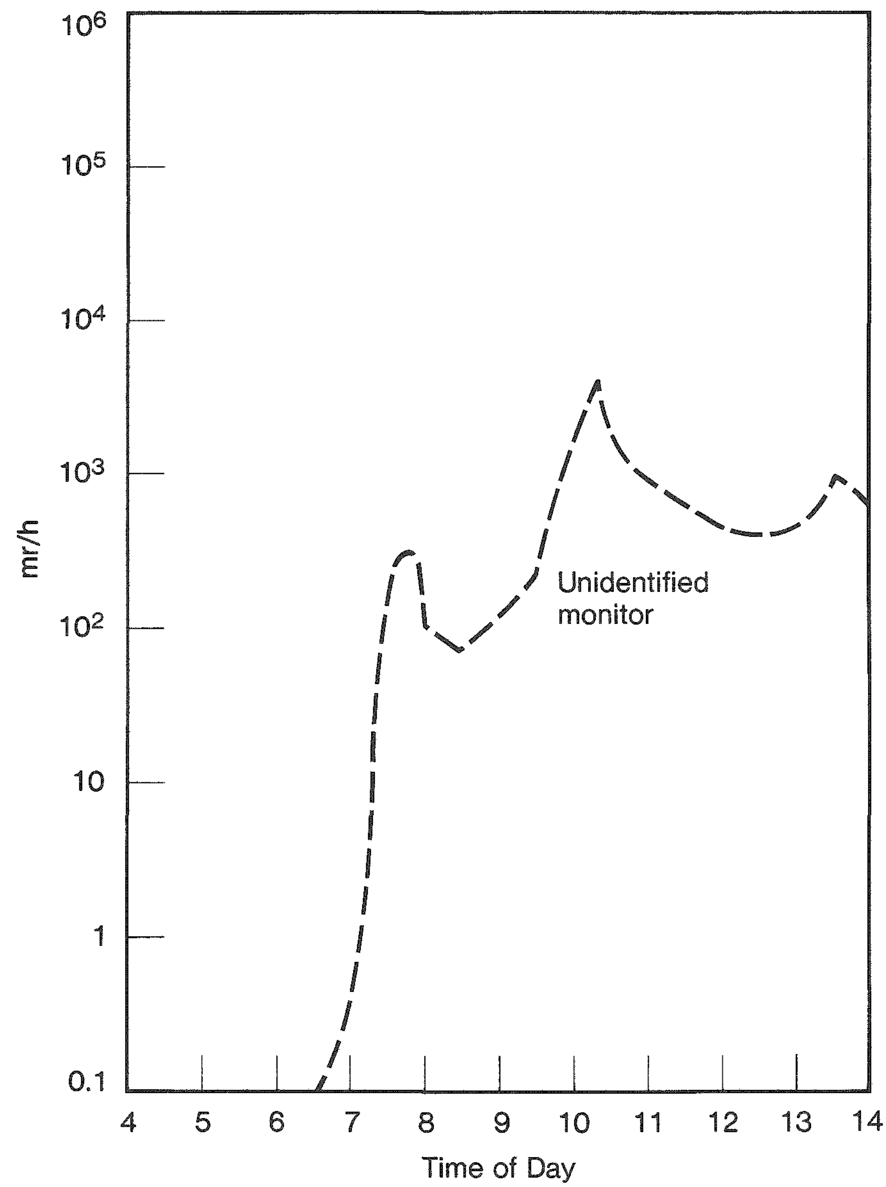


Figure RM-5a. Three Mile Island – Unit 2 response to selected containment area monitors.

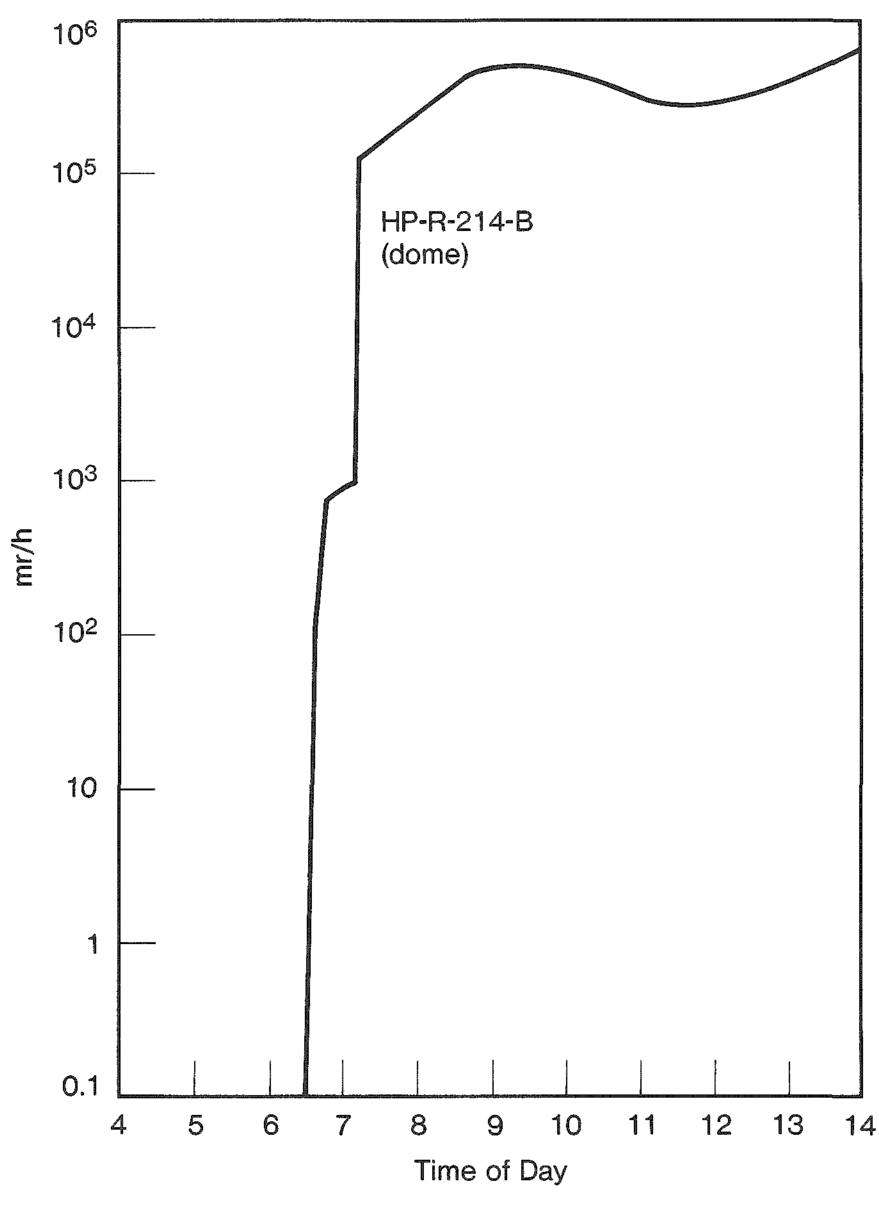
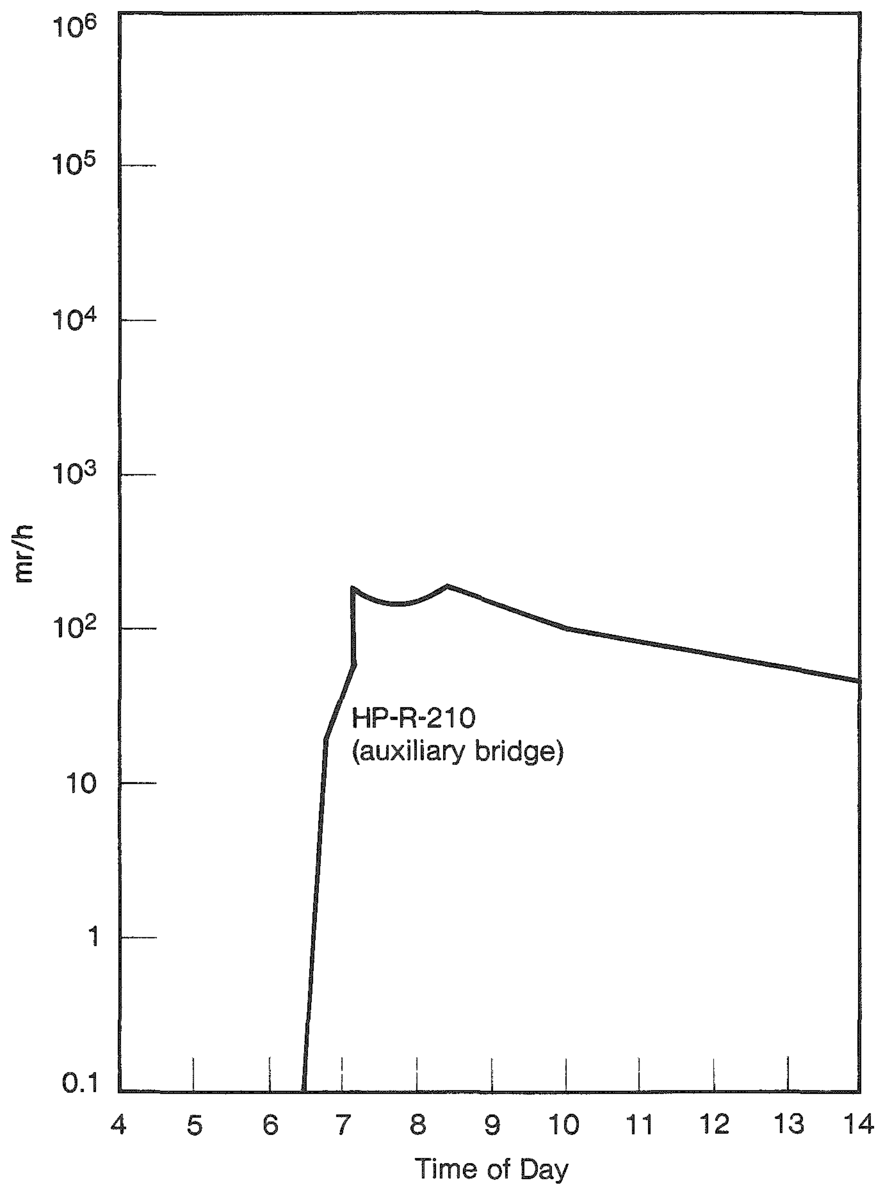

10^6
10^5
10^4
10^3
10^2
10
1
0.1
mr/h
HP-R-214-B
(dome)
4
5
6
7
8
9
10
11
12
13
14
Time of Day

Figure RM-5b.



mr/h
10⁶
10⁵
10⁴
10³
10²
10
1
0.1
HP-R-210
(auxiliary bridge)
4
5
6
7
8
9
10
11
12
13
14
Time of Day

Figure RM-5c.

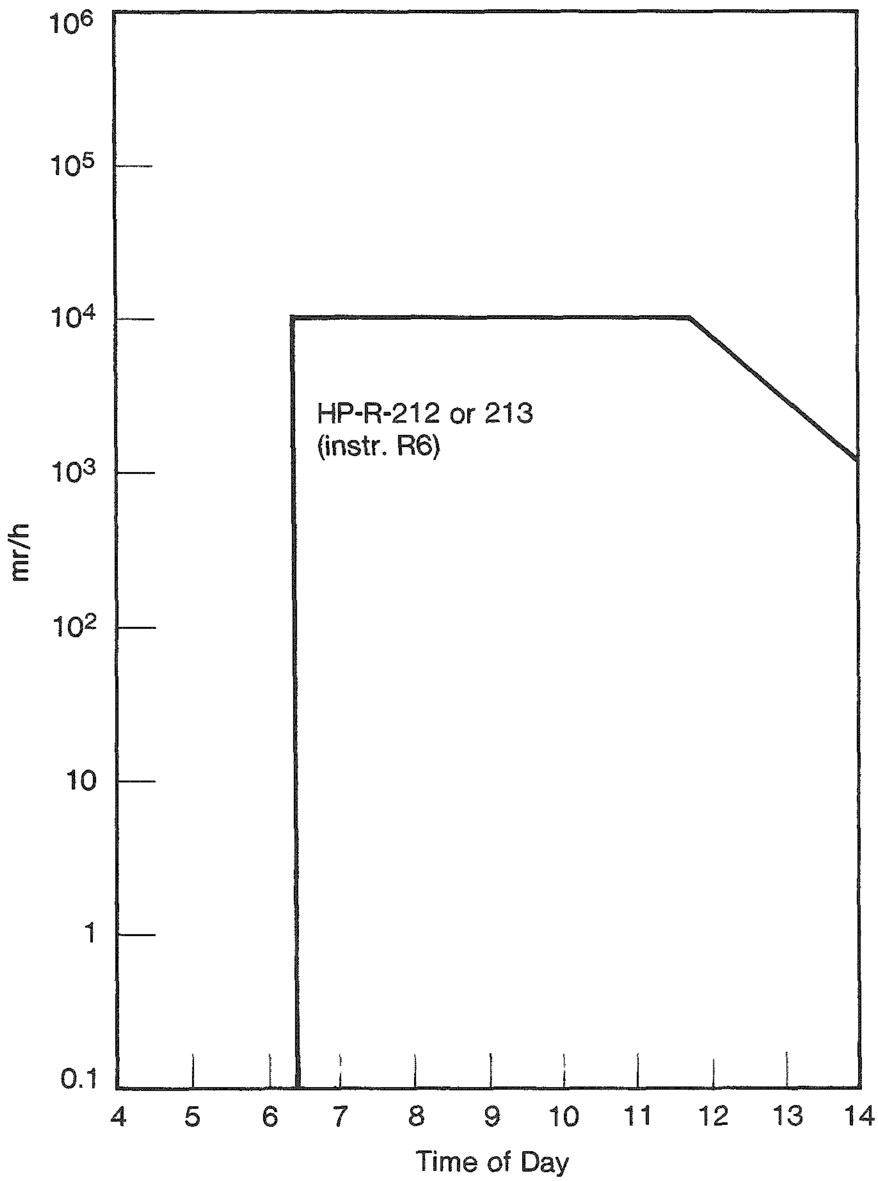
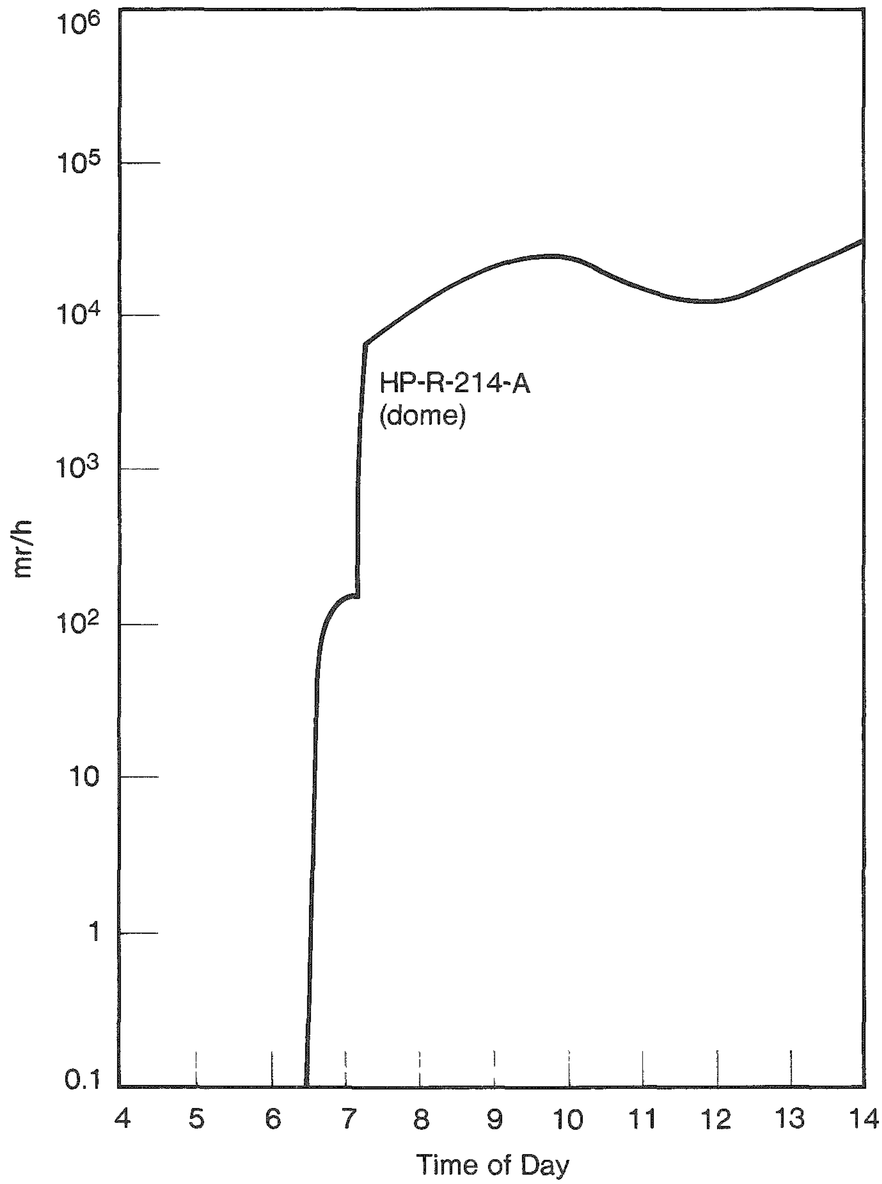

10^6
10^5
10^4
10^3
10^2
10
1
0.1
mr/h
HP-R-212 or 213
(instr. R6)
4
5
6
7
8
9
10
11
12
13
14
Time of Day

Figure RM-5d.



10^6
10^5
10^4
10^3
10^2
10
1
0.1
mr/h
HP-R-214-A
(dome)
4
5
6
7
8
9
10
11
12
13
14
Time of Day

Figure RM-5e.

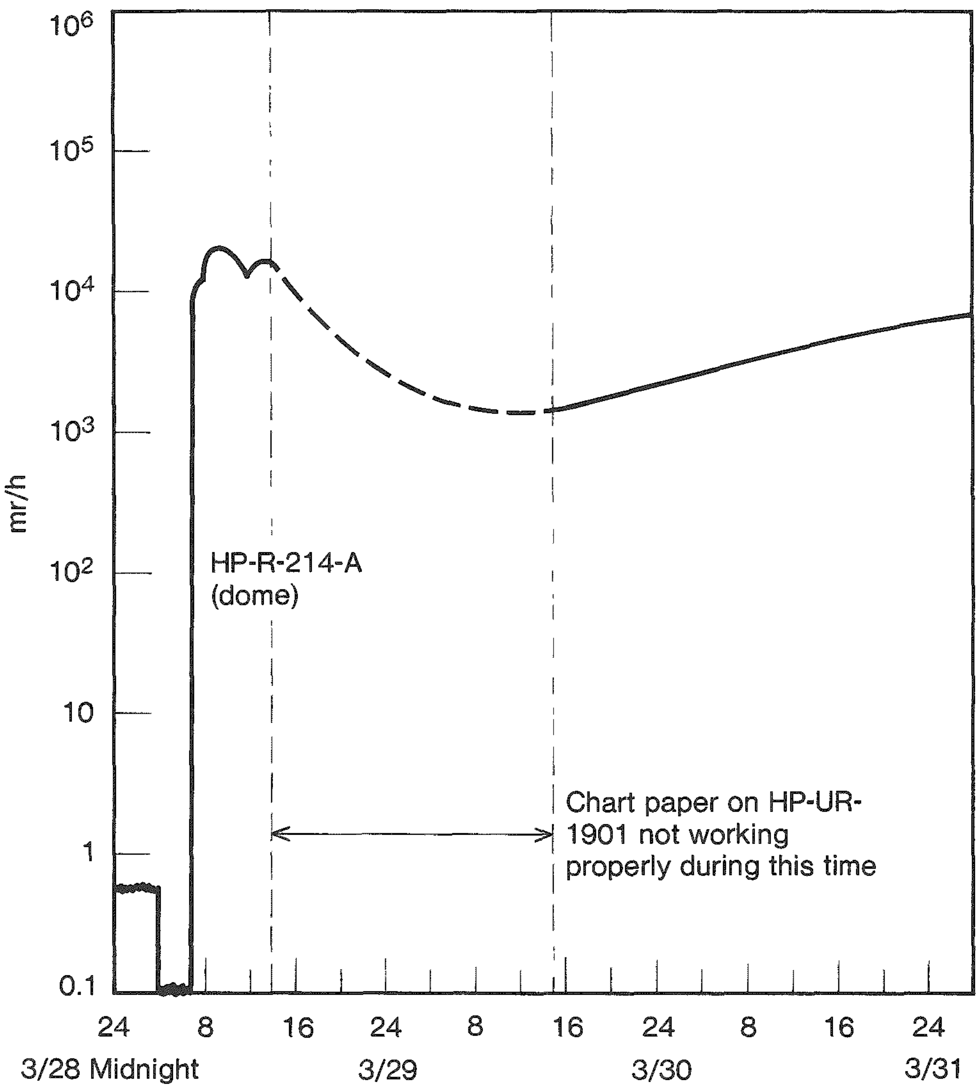
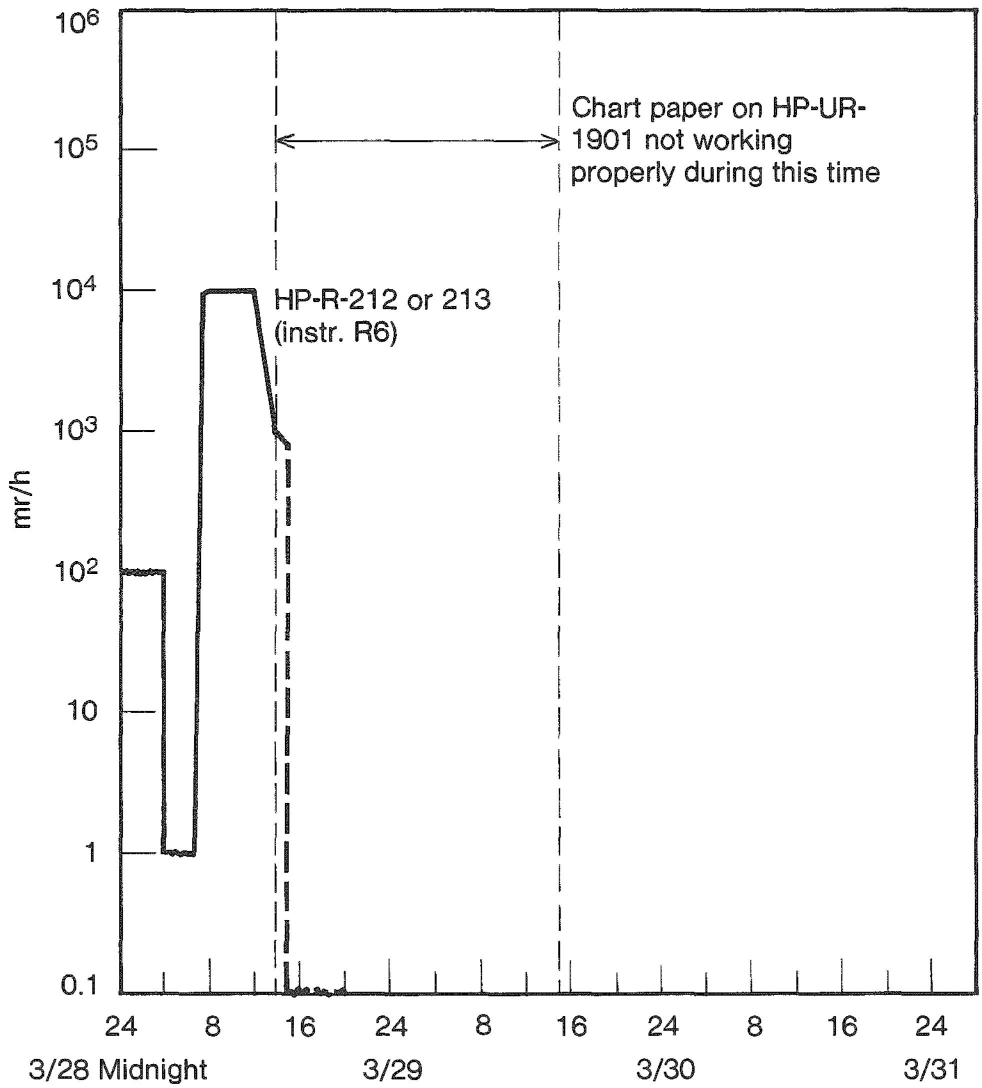
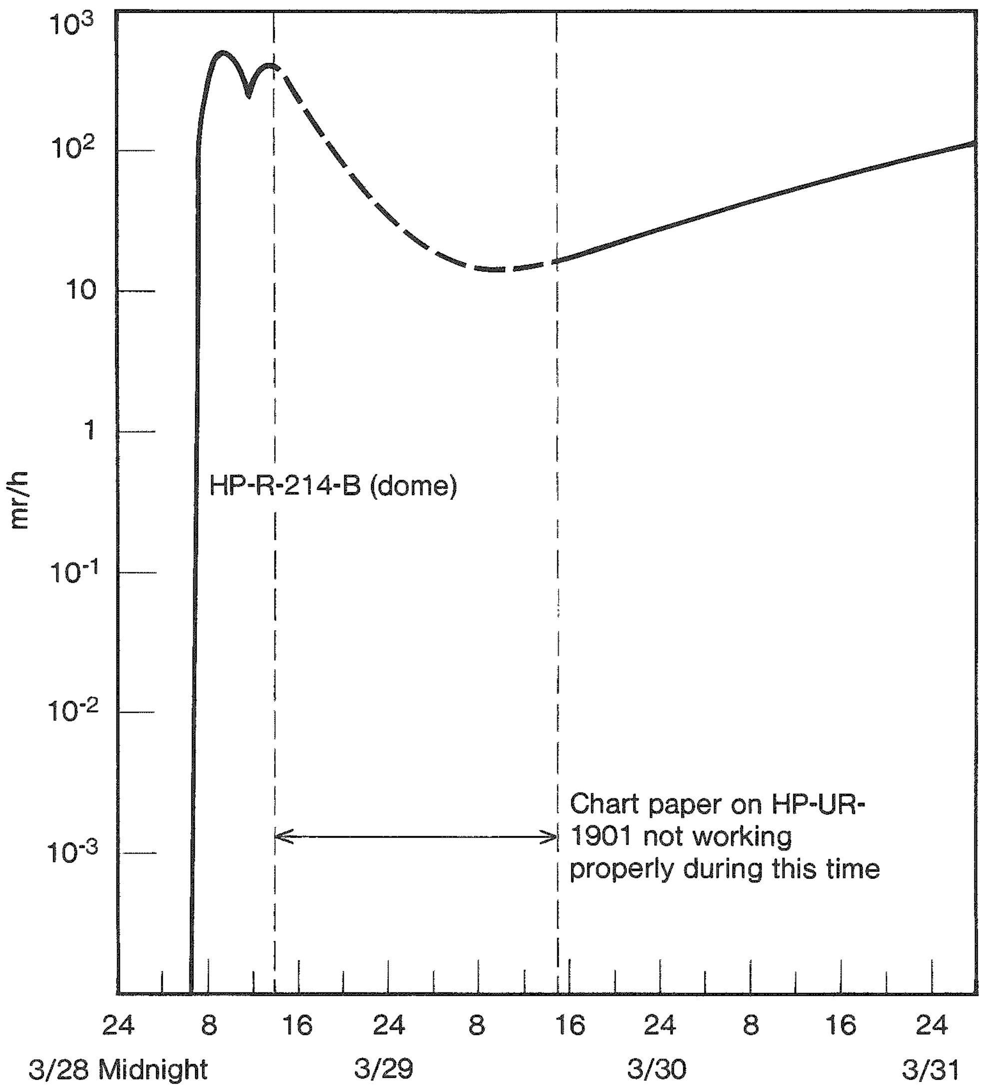


Figure RM-6a. Three Mile Island – Unit 2 response of selected containment area monitors.



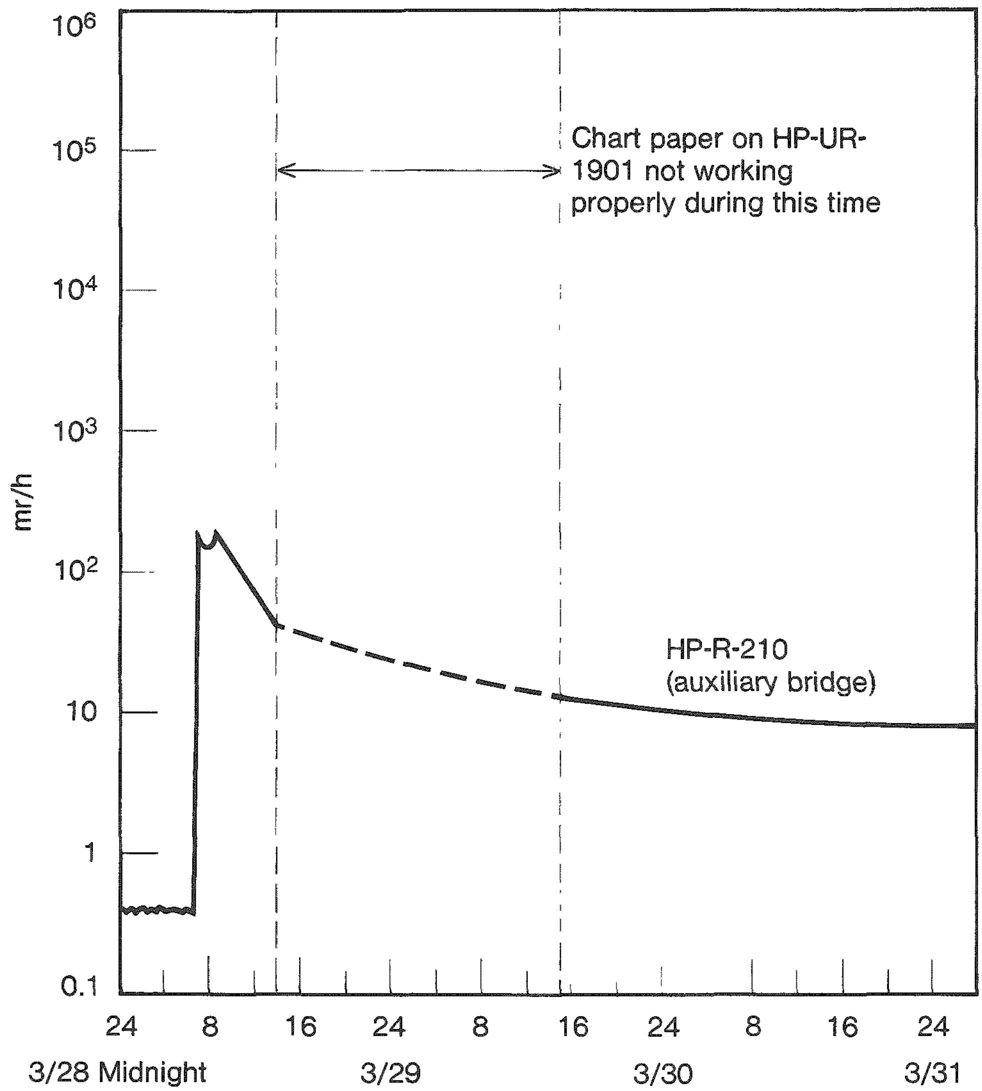
10^6
10^5
10^4
10^3
10^2
10
1
0.1
mr/h
Chart paper on HP-UR-1901 not working properly during this time
HP-R-212 or 213 (instr. R6)
24
8
16
24
8
16
24
8
16
24
3/28 Midnight
3/29
3/30
3/31

Figure RM-6b.



10^3
10^2
10
1
10^{-1}
10^{-2}
10^{-3}
mr/h
HP-R-214-B (dome)
Chart paper on HP-UR-1901 not working properly during this time
24 8 16 24 8 16 24 8 16 24
3/28 Midnight
3/29
3/30
3/31

Figure RM-6c.



10^6
10^5
10^4
10^3
10^2
10
1
0.1
mr/h
Chart paper on HP-UR-1901 not working properly during this time
HP-R-210 (auxiliary bridge)
24
8
16
24
8
16
24
8
16
24
3/28 Midnight
3/29
3/30
3/31

Figure RM-6d.

APPENDIX ROUTES

POTENTIAL ROUTES OF RADIOACTIVITY TRANSPORT

Introduction

The fission products embody a group of elements from about mass 80 to about mass 150. While the elements exhibit widely differing chemical properties, most are radioactive and exhibit either β or γ radiation. In order to assure that the fission products will not escape control , "a defense-in-depth" has been the guiding principle through which containment of radionuclides has been effected in reactor design. Containment levels include the cladding, the reactor vessel, and the reactor building. Each of these is a high integrity system serving to back up the others if one is breached during reactor operation.

While not considered directly as a contributing level of defense against fission product dispersal, the fuel itself can and does act in this capacity. Because of the chemical behavior of the fuel-fission product system, the fuel retains the bulk of the fission product inventory*. The only event expected to perturb this condition would be one resulting in gross fuel dispersal.

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*A naturally occurring event that supports the postulate that the fuel will retain the majority of the fission products produced is the French discovery at Oklo in the Gabon Republic in Africa. Studies of the Oklo natural reactor indicate that about two-thirds of the total fission events were caused by the ^{235}U initially present and the other one-third by ^{239}Pu converted from the ^{238}U by neutron capture and decay. Further the Oklo ore deposit has exhibited remarkable stability. The ^{235}U and ^{238}U fuels have been confined within the grains of the ore which has also retained the plutonium. Also at least one-half of the thirty or so fission product elements have remained immobilized in the ore.

Fission products can be placed in one of three major categories. These categories are:

1) Volatiles, e.g., rare gases, cesium, iodine, cesium iodide;

2) Oxide formers, e.g., zirconium, cerium, neodymium; and

3) Alloy formers e.g., ruthenium, palladium, molydenum.

In the discussion to follow primary attention is given to the release of volatile fission products. Table I gives release fractions from analysis of TMI-2 reactor coolant samples and indicates about 10 percent of the core inventory of cesium was in the coolant. The purpose of the present appendix is to identify the most probable routes of uncontrolled and controlled releases of radioactive material from the reactor to the reactor building, the auxiliary building, the fuel handling building and to the environment. Work aimed at estimating the quantities of fission products released will be reported later.

It is believed that minimal fission product transport took place prior to 0622 on March 28. Present scenarios suggest aggressive metal-water reaction about this time resulting in extensive cladding oxidation and cladding breach bringing about fission product release shortly thereafter. While reactor building isolation occured at 0756 due to high pressure (~4 psig) in the reactor building, several routes were reestablished to enable "normal" reactor functions, e.g., makeup and purification systems.

Path 1--Liquid and/or Gas Releases from the Letdown System

This path is considered the major contributor to radioactive release both initially and for the duration of the accident (see Figure Routes 1). The majority of the letdown, purification, and makeup systems are located in the auxiliary building. These systems normally handle large quantities of reactor coolant which during the course of the accident became highly contaminated as the result of fuel-cladding breach. The amount of coolant released to the auxiliary building and the effect of this release on airborne radioactivity levels in the plant are not accurately known. However, coolant transported into the auxiliary and fuel handling buildings was cooled and depressurized, thus releasing entrained and dissolved gases. Further, evaporation of water leaking onto the floors probably contributed to airborne and particulate activity inside the auxiliary and fuel handling buildings. Several pathways for liquid/gas releases into the auxiliary and fuel handling buildings from the makeup and purification system are considered possible. There are several relief valves in these systems; Table II lists their location, setpoint pressure and points of discharge.

Of the valves listed in Table II, MU-R-3*, MU-R-5A and MU-R-5B are the most likely pathways for release of reactor coolant to the floor drain and from there to the auxiliary building sump. There were difficulties in maintaining letdown flow on March 28 and the block orifice MU-1-FE was bypassed in an attempt to increase letdown flow. Letdown flow restrictions could have caused a pressure increase resulting in the lifting of these relief valves. Operator reports showed letdown flow oscillation indicative of relief valve(s) opening and closing. The lifting of relief valve MU-R-1 would only have led to liquid release to the reactor coolant bleed holdup tanks, increasing the potential for these tanks to discharge liquid into the waste gas system vent header and out through vent header liquid drains. A liquid release through the reactor coolant bleed hold up tank vent line probably did not occur before March 30.

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*Operator interviews indicate that the piping from MU-R-3 to the reactor bleed holdup tanks may not have been installed.

Relief valves MU-R-2 and MU-R-4 were determined to represent no potential for release. MU-R-2 discharges to the makeup tank and MU-R-4 had been replaced with a blind flange.

The makeup tank is designed to operate with a gas space occupying about one-third its normal tank volume. It was in this space that gases evolved from the cooled and depressurized reactor coolant. One indication that this tank was a source of gaseous leakage was that area radiation monitor readings were high when the liquid level in the makeup tank was high. This correlation was considered reliable since the makeup tank is not normally vented and a change in level causes a corresponding change in pressure thus increasing gas leakage from the tank if leaks are present. It is suspected that leakage did occur through the make up tank vent valve.

Path 2--<u>Liquid from the Reactor Building Sump to Auxiliary Building Sump</u>

Shortly after the turbine and reactor trips the reactor coolant system pressure reached the electromatic relief valve (ERV) opening setpoint of 2255 psig. When the ERV did not close properly at lower system pressure (2205 psig), reactor coolant continued to be discharged to the reactor coolant drain tank (RCDT). This created an overpressure in the RCDT which first lifted the RCDT relief valve and then burst the RCDT rupture disc when 192 psig was reached. Steam and water were released into the reactor building sump. The sump pumps activated and pumped some 8120 gallons (run time of each sump pump times 140 gpm, the acceptance test for each pump) of low activity water to the auxiliary building sump tank. This tank is estimated to have had only 700 gallons freeboard prior to this transfer. There is a low probability that the discharge valve lineup was to the miscellaneous waste holdup tank but strongest evidence indicates lineup to the auxiliary building sump tank (see Figure Routes 2). The first sump pump started at 0408 on March 28 and the second at 0411 (see Table III for Sequence of Events); both pumps remained on until approximately 0428 when they were manually turned off. The amount of radioactivity transferred from the reactor building sump in this event was relatively small since transfer was completed prior to the known time of fuel cladding breach.

With transport of 8120 gallons of water to the auxiliary building sump tank the auxiliary building sump also must have overflowed since it was indicating about 60% of capacity prior to the accident (sump capacity equals 6371 gallons). Flooding of the auxiliary building sump created a situation that would later facilitate the transport of extremely contaminated coolant water into various parts of the auxiliary and fuel handling buildings through flooded floor drains. Water went to the 280' 6" floor level of the auxiliary building, and via floor drains to the auxiliary building sump at the 258' 6" level. After the reactor building sump pumps were shut off, a siphon may have been established between the reactor building sump and the auxiliary building sump tank. Although the top of the auxiliary sump tank is approximately 6 feet above the top of the reactor building sump, between 0630 and 0756 on March 28 there was a 2 to 4 pound per square inch positive pressure in the reactor building developing a driving head of 5 to 10 feet of water. Also, contaminated water accumulated on the reactor building floor effectively raising the water depth above the top of the reactor building sump. However, because of sensitivity to reactor building pressure it is likely that the siphon contributed only minimally to water transport.

In summary the open ERV allowed reactor coolant to flow to the RCDT and through the failed rupture diaphragm into the reactor building sump. The reactor building sump pumps were turned off at ~ 0438 after pumping as much as 8120 gallons of relatively low activity water to the auxiliary building sump tank, which must have overflowed into the auxiliary building sump which also overflowed.

Path 3--<u>Liquid and/or Gas from the Reactor Coolant Drain Tank to the Waste Gas Disposal System</u>

Of the several pathways that lead from the RCDT to the auxiliary buildings it is believed that only the RCDT vent to the radwaste gas system vent header made any substantial contribution to transport of radioactive material to the auxiliary buildings.* It has been determined that a small release of radioactive gases occurred from the waste gas vent header system and from the vent stack prior to 0756. The formed pathway would be identified later as the principal pathway for release of radioactive gases.

The RCDT is connected to the vent header via two paths:

1) the former connects the RCDT to the reactor coolant bleed hold up tanks (RCBHT) by a two inch line and through valve WDL-V1095 (see Figure Routes 3), and

2) the latter is a 1 inch vent line from the RCDT to the waste gas vent header; this line is normally isolated from the vent header by valves (WDL-V-126 and WDL-V-127 (see figure Routes 4).

In the former case valve WDL-V-1095 automatically closes at 10 psig in the RCDT to protect the reactor coolant bleed hold up tanks but reopens at 6 psig to reestablish communication with the reactor coolant bleed hold up tanks at lower pressure. The RCDT pressure exceeded 10 psig some 2 minutes after the reactor trip

*Conflicting data exist for the liquid pathway from the reactor coolant drain tank. The flow recorder for this pathway indicated (with some ambiguity) that a substantial volume of water may have been transported to the auxiliary building via this route prior to the initial reactor building isolation. However, the operator interviews state that this manually initiated pathway was not exercised. At any rate, this was only a pathway to the reactor bleed holdup tanks and vent gas header.

which presumably closed valve WDL-V-1095. The valve presumably reopened after the rupture diaphragm failed (~ 0435) thus reducing the RCDT pressure to less than 6 psig and would have provided an open line between the RCDT and the RCBHT's that remained open until reactor building isolation at 0756. (Valves WDL-V105 and WDL-V1092 also function as containment isolation valves, closing on receipt of an engineered safety features signal and remaining closed unless opened by operator action.)

The RCDT is normally vented through the latter path mentioned above via the two valves (WDL-V126 and WDL-V127) to the reactor building vent header and then to the radwaste gas system vent header throuth the containment isolation valves (WDG-V2 and WDG-V199). All four valves are normally open and are believed to have been open at the time of the accident; in fact valves WDL-V126/127 were open until being closed on June 5. Following bursting of the RCDT rupture diaphragm, water vented from the pressurizer would have flashed to steam in the RCDT (temp. 225°F). The steam would then flow to the radwaste gas system vent header.

Buildup of pressure in the RCDT would have pressurized the waste gas vent header and created a driving pressure as high as 192 psid just before the rupture disc failed. The pressurization of the RCDT to the rupture disc failure point coincides with a step increase in gasous activity as measured at 0415 by the ventilation system monitors. Flow by either of the two pathways described would have eventually been released by any leaks in the vent header system. It has since been confirmed that some leaks were present but it has not been established that they were large enough to account for the released gaseous activity.

Transporting reactor coolant (water/steam) into the radwaste gas system vent header would result in broader distribution of radioactive material. Contaminated water transferred into the radwaste gas system would have ended up in the auxiliary building sump or in the auxiliary building sump tank. Radioactive gases released to the vent header would have been subject to any leakage paths from the radwaste gas system. Probably most important though is the presence of excessive water and/or steam and pressure in the radwaste gas system which may have degraded the performance of the system.

Early on March 29, the buildup of noncondensible gases in the makeup tank caused a reduction in letdown flow due to back-pressure build up in the makeup tank. Manual venting of the makeup tank to reduce pressure was initiated at 0710 on March 29 and continued until sometime after March 30. The venting process initially consisted of short periods of time when MU-V-13 was opened to vent makeup tank pressure into the waste gas decay tanks via the leading vent header. However, the pressure build-up became too rapid to control with short venting bursts. Between 0600 and 0710 of March 30 the pressure in the makeup tank forced relief valve MU-R-1 to open, releasing all the gas and liquid from the make-up tank to the reactor coolant bleed holdup tanks. Sometime after the opening of MU-R-1 the pressure of the reactor coolant bleed holdup tanks was observed to exceed the relief valve setpoint of 20 psig. Because of the operational problems with the tanks in this condition the makeup tank was vented to reestablish normal pressure and liquid level. Because of the high pressure in the reactor coolant bleed tanks, an uncontrolled release may already have been in progress via the relief valves on the reactor coolant bleed tanks.

Subsequent tests of the radwaste gas disposal system have revealed a large hole (~ 3/4 inch diameter) in the B waste compressor. The impeller of the gas compressor circulates water in the head assembly to compress and discharge gases. The down-stream end of the waste gas compressor system is connected to the reactor coolant bleed tanks, the makeup tanks, the miscellaneous waste holdup tanks and the auxiliary building sump. Upstream the system is connected to the waste gas header.

It is not known when the hole developed. The waste gas compressor is routed to the RCDT via valves WDG-V2 and WDG-V199. As these valves were shut at 0756, the hole would have had to be present prior to that time in order to have represented a significant transfer path. It is possible that the hole did not develop until after the accident.

The record shows that the waste gas compressors were operating abnormally for periods of time following the accident; during one period the compressors operated at 18 amperes instead of their normal 36 amperes. The compressors suddenly returned to normal sometime between 3/30 and 4/2. Whether this abnormality was caused by water in the compressor or by pumping against an 80 psig backpressure is not clear. With the predominantly high pressures existing in the system, the corrosive break through on the waste gas compressor head assembly may have served to relieve system pressure and returned it to normal operation.

RadioIodine Release Following Accident

One of the more interesting scientific aspects of the Three Mile Island accident was the lack of any significant release of radioactive iodine.* The released iodine was in most cases not detectable even by sophisticated modern techniques. The equally non-detectable environmental effects must be estimated using mathematical models and worst-case analysis. All experts that have analyzed the data have come to similar conclusions. The extremely intense program of radiation measurements, extensive analysis of milk, water and vegetation coupled with direct measurements in people living nearby the plant make the possibility of any significant undetected release of radioactive

*An estimated 10 to 20 curies of radioactive iodine was released from the site relative to 2 to 10 million curies of radioactive gases.

iodine vanishingly small. The radiation dose caused by iodine released is calculated to be a maximum of a few millirem to the thyroid of a child living very close to the plant. This should be compared to the tens of millirem received from "natural sources" and a thousands of millirem received routinely for medical diagnoses.

Table IV lists the various estimates of whole body and thyroid doses from the TMI accident. These doses in all cases are small relative to background (natural) levels.* For comparison purposes, data are included for the Windscale accident in 1957. Although a thousand times more radioiodine was released in this acident, no adverse health effects attributed to this accident have been reported in more than 20 years.

*The largest Iodine concentrations in milk samples were about 1/10 of those found in mid-Pennsylvania after fallout from a Chinese weapons test in October 1976.

TABLE I

Core Inventory Release Fractions to Primary Coolant and Auxiliary Building for Various Classes*

Class	Release Fraction
Noble Gases	0.55
Halogens	.1
Mo, Y	.01
Cs, Rb	.1
Solubles	.01
Insolubles	.001

*Release fractions were derived from early analytical data on the Reactor Coolant System contents (A. Miller, EPRI) and calculations of core inventory of major radioisotopes at time zero. More recently, analysis of the water standing in the reactor building indicates the possibility that about 44% of the core inventory of 137 Cs has been released from the core altogether. There are also indications that some of the other volatile fission products may have exhibited significantly higher release fractions than those given in this table.

TABLE II

Letdown, Makeup, and Purification System's Relief Valves

Valve Number			
MU-R-1	Downstream of Makeup Tank - Makeup Tank Relief Valve	80 psig	Discharges to RCBHT
MU-R-2	Upstream of seal return cooler	150 psig	Discharges to make-up tank
MU-R-3	Downstream of Block Orifice Letdown Relief Valve	130 psig	Discharges to RCBHT*
MU-R-4+	Inlet to core flood tanks	700 psig	Discharges to floor drain
MU-R-5A/B	Makeup and purification demineralizer relief valves	150 psig	Discharges to floor drain

*Personnel interviews indicate that the discharge line may not be installed, i.e. valve discharges to the room (floor drain).

+Investigations have indicated that this relief valve has been removed and replaced with a blind flange and pressure gauge.

TABLE III

Sequence of Events Relevant to Operation of the Reactor Building Sump Pumps Following the Accident

Time	Comments
04:03:47*	RCDT relief valve opens - 120 psig
04:08:06+	RB sump pump A starts - WDL-P-2A
04:10:56+	RB sump pump B starts - WDL-P-2B
04:11:05+	RB sump high level alarm - 4,650 feet
04:15:25*	RCDT rupture disc bursts - 192 psig
04:38:47+	RB sump pump A stops
04:38:48+	RB sump pump B stops

*Time based on reactimeter data.

+Computer alarm printer.

TABLE IV

WHOLE BODY AND THYROID DOSES FROM THE TMI-2 ACCIDENT

Thyroid Radiation Dose

Reference Number	Calculated Maximum Dose (mrem) to Thyroid for 131						U.S. Average Dose Due to Internal Dionuclides (mrem)	Ci Released to Atmosphere (I-131)	I-131, Cs-137 Whole Body Counting (mrem)	
(1) TMI	Inhalation			Cow Milk						
	Infant		Adult	Infant		Adult				
	2.3			5						
(2), (6) TMI	New Born	One Year	Adult	New Born	One Year	Adult	Thyroid			
	2	6.5	5.4	6.9	4.7	0.6	27/year			
(3) TMI								10		
(4) TMI								14		
(5) TMI	Adult 3.1			Infant 0.97			Infant 7.8 over 28 days			
(7) TMI									ZERO	
(6) TMI									ZERO for 760 persons within 3 miles TMI	
(8) TMI								8		
(9), (10) Windscale, 1957								20×10^3	Adult 1800	Child 12,000

Whole Body Radiation Dose (Off-Site)

Reference	TMI Maximum Dose to Individual	Background at TMI: Individual	Background at TMI: Population Dose	Calculated TMI Population Dose (person-rem)	Calculated Average Individual Dose (mrem)	Radioactivity Released (Ci)
(1)	83 mrem	9 mrem/28 days (116 mrem/year)	19,740 person mrem/28 days (250,984 person mrem/year)	3200	1.5	
(2)	70 mrem			2800	1.2	2.5×10^6
(3)				2000	0.9	2.9×10^6
(4)	43.3 mrem			3500	1.6	10×10^6
(11)		5-11 mr/28 days (60-145 mr/year)				
(6)				400^{+3600}_{-360}		2.4×10^6
(6) Appendix B	70 mrem	0.7 mrem/4 days		2000	1.3	

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(4) Anonymous, "Assessment of Offsite Radiation Doses from the Three Mile Island Unit 2 Accident" Pickard, Lowe and Garrick Report TMI-116 (Rev. 0) 1979.

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(11) A. E. Fritzsche, "An aerial radiological survey of the Three Mile Island Station Nuclear Power Plant, (Goldsboro, Pennsylvania) Data of Survey: August 1976", EG&G Inc. reort number EGG-1183-1710 (March 1977).

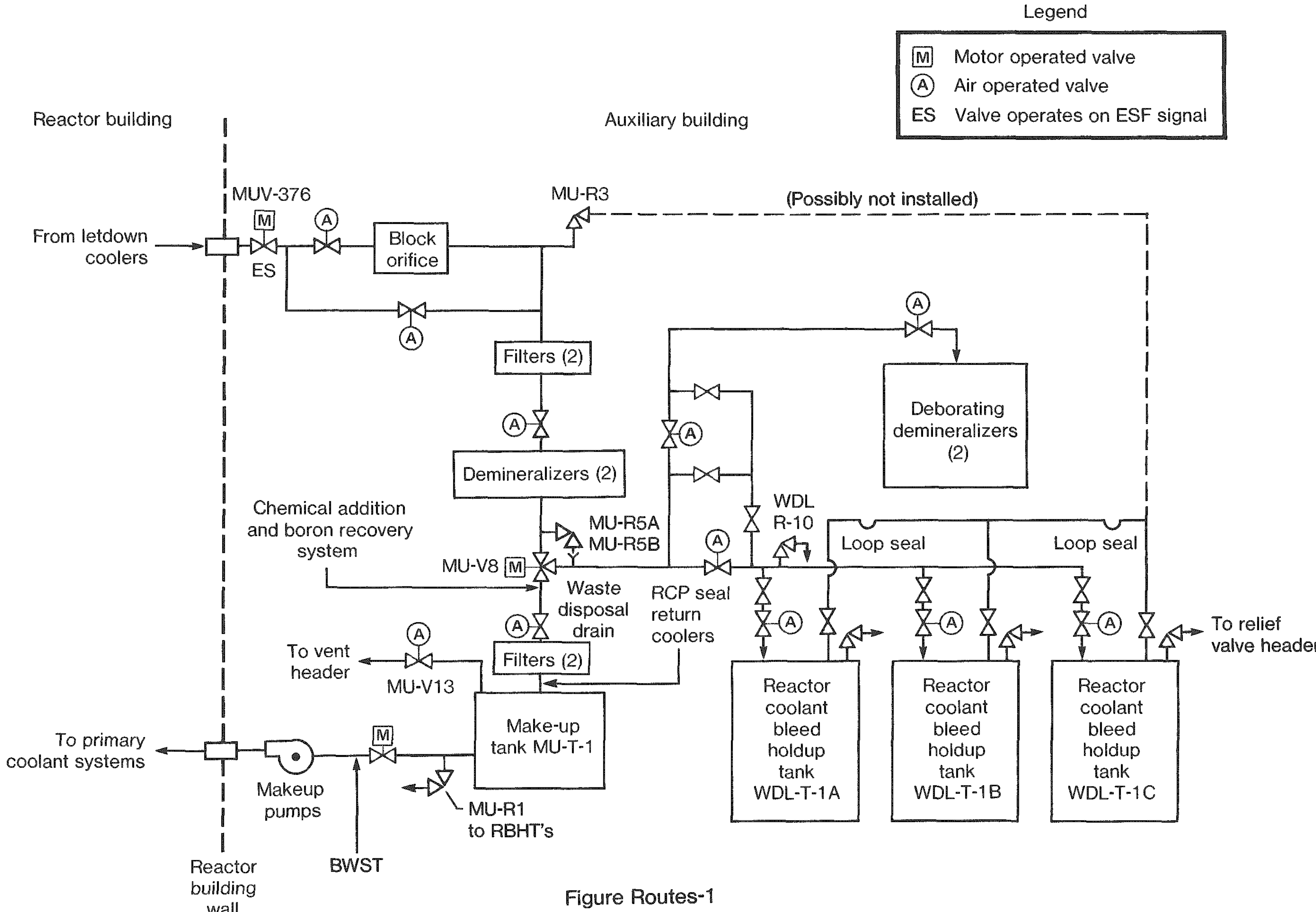


Figure Routes-1
Path-1

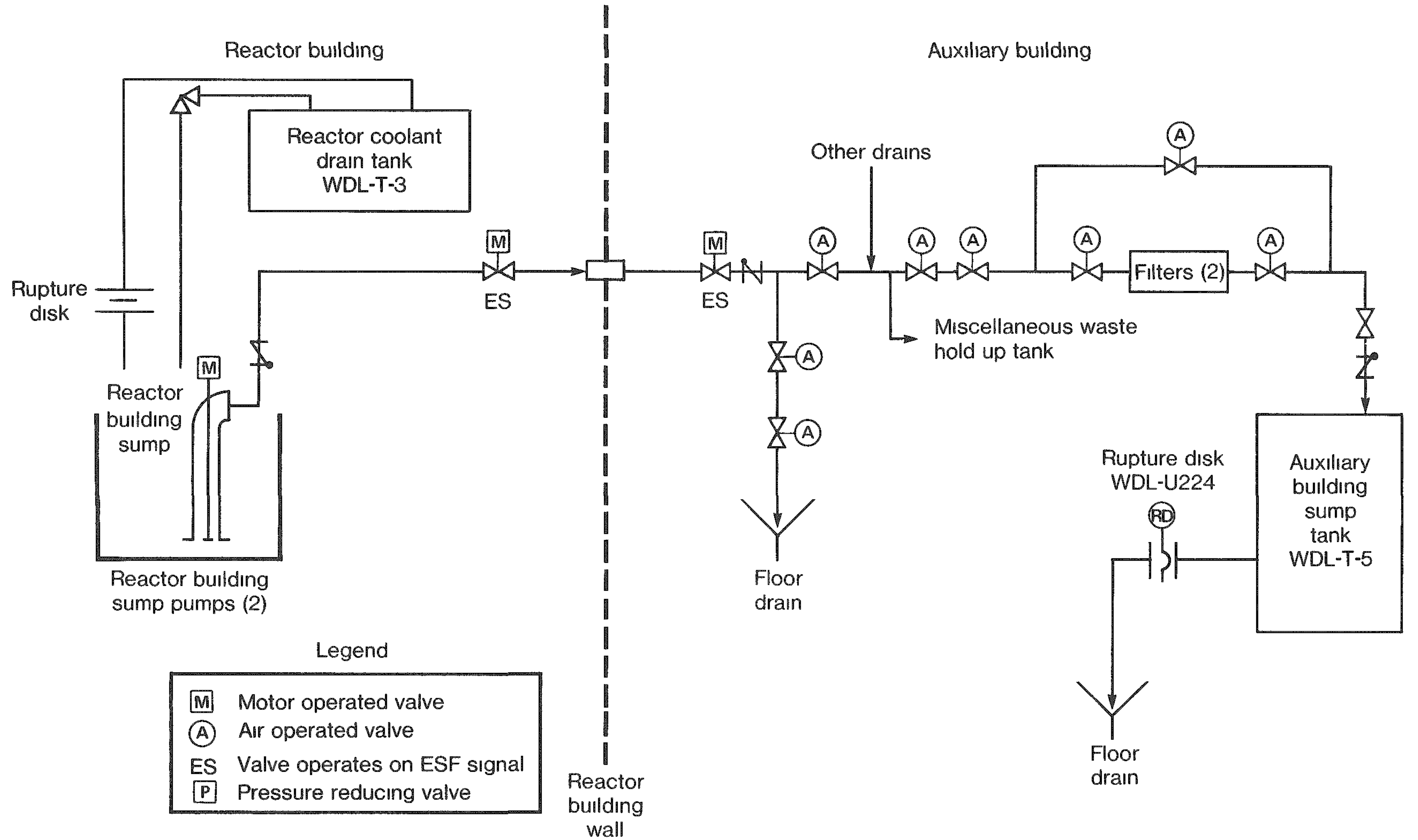
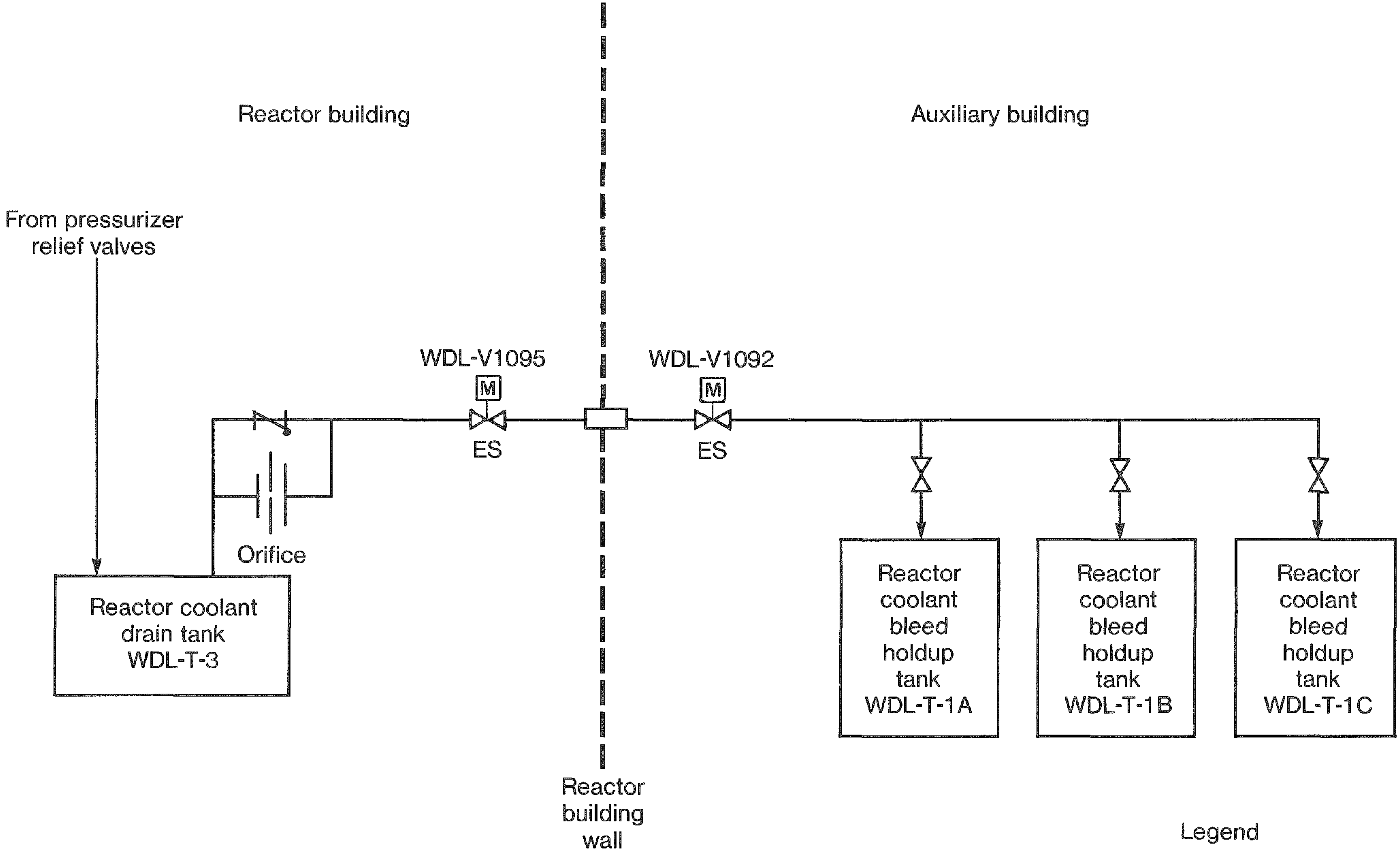


Figure Routes-2
Path-2



Legend

M	Motor operated valve
A	Air operated valve
ES	Valve operates on ESF signal

Figure Routes-3
Path-3

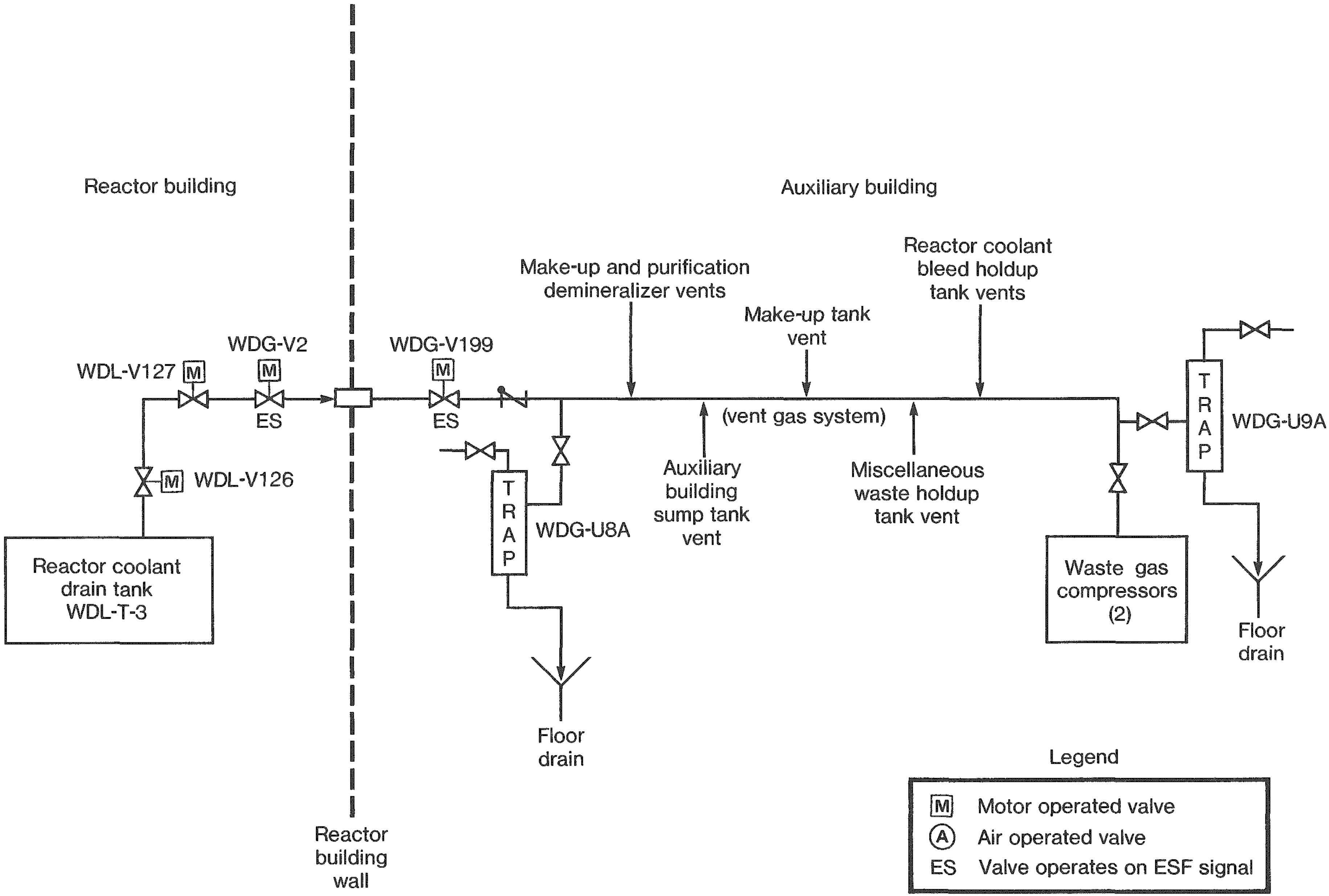


Figure Routes-4
Path-4

APPENDIX SP

REACTOR BUILDING SUMP SYSTEM

The reactor building floor drains, the fuel transfer canal drains, the decay heat removal piping drain system, and the quench tank relief valves discharge flow into the reactor building sump. The sump is a 280 ft^3 steel lined pit partitioned into a wet and dry section.

During a loss of coolant accident, the water in the wet section will overflow into the dry section. The decay heat removal pumps then take suction from the flooded, normally dry section and return this water via heat exchangers to the reactor vessel or to the suction of the high pressure injection pumps. The reactor building spray pumps can also take suction from this section of the sump.

Two reactor building sump pumps are provided to transfer the liquid collected in the wet section through a discharge filter to the miscellaneous waste holdup tank or the auxiliary building sump tank located in the auxiliary building, Figure SP-1. Each single stage centrifugal pump has a capacity of 200 gpm, with the pump motors rated at 7.5 hp. each.

Each pump is controlled by a selector switch (MAN-OFF-AUTO) located on radwaste panel No. 301 in the auxiliary building. When AUTO is selected, one pump starts when water level increases above 38 in. as sensed by a level switch. If the level in the sump continues to rise, the second pump starts automatically at approximately 53 in. and this condition is alarmed on radwaste panel No. 301. The sump pumps automatically stop when the level decreases below 25 in. If the selector switch is set on MAN, the selected sump pump will start and run until the water level decreases and actuates the low level trip switch which causes the pump to stop. Selection of the OFF position shuts down the pumps, and prevents them from starting automatically. When the pumps and associated reactor building isolation valves (WDL-V126 and WDL-V271), receive an engineered safety features (ESF) isolation signal, the pumps trip and the valves close. Indication of sump level is provided on radwaste panel No. 301.

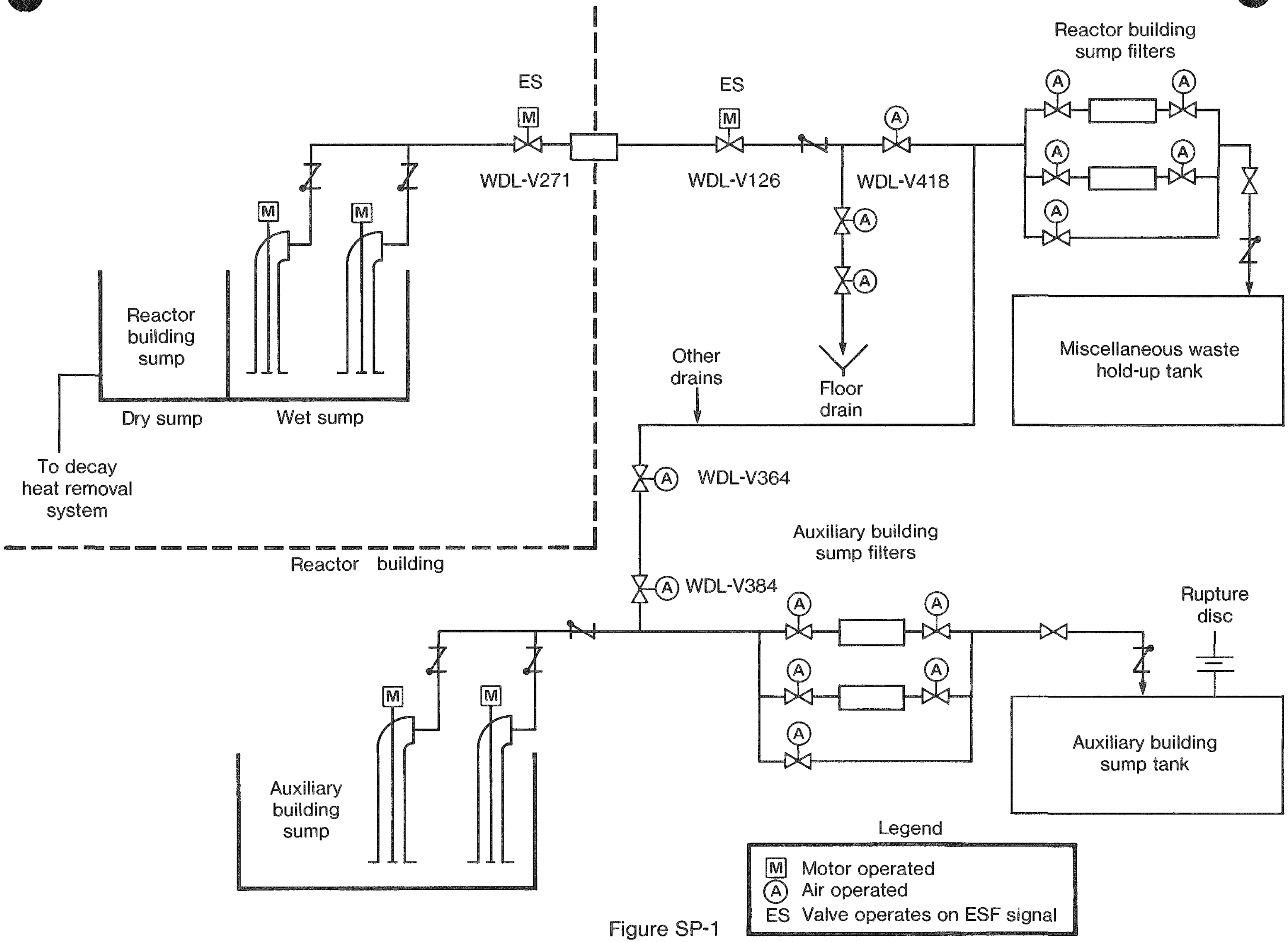

Reactor building sump filters
ES
ES
WDL-V271
WDL-V126
WDL-V418
Reactor building sump
Dry sump
Wet sump
To decay heat removal system
Other drains
Floor drain
Miscellaneous waste hold-up tank
WDL-V364
Reactor building
Auxiliary building sump filters
WDL-V384
Rupture disc
Auxiliary building sump tank
Auxiliary building sump
Legend
M Motor operated
A Air operated
ES Valve operates on ESF signal

Figure SP-1

APPENDIX STEAM DUMP

STEAM PRESSURE CONTROL SYSTEMS

1.0 INTRODUCTION

The main (secondary) steam system is designed to deliver steam from the steam generators to the high pressure turbine of the turbine generator set. It is also designed to deliver steam to the main and emergency steam generator feedwater pump turbines, to the turbine gland seal system, and to the moisture-separator reheaters.

Steam pressure control, when the turbine generator set is not in operation, is provided by safety valves, atmospheric dump valves, and turbine bypass valves. Figure STEAM DUMP-1 depicts the relative locations and flow paths for these components.

2.0 MAIN STEAM SAFETY VALVES

2.1 System Design

Pressure relief, required for protection against over-pressure in the system, is provided by the main steam safety valves. The design relief capacity of the valves is the steam flow equivalent to 120% of the design reactor power level of 2772 Mwt plus the 16 Mwt contribution from the reactor coolant pumps. This design capacity is sufficient to accept the transient resulting from a high power level trip and allows for measurement error.

2.2 System Operation

Pressure relief is initiated at the system design pressure of 1050 psig, and the first safety valve bank is set to relieve at this pressure. Additional safety valve banks are set to relieve at pressures up to 1102 psig, as allowed by the ASME Code. For each steam generator there are:

- Four 6 x 10 inch valves set at 1050 psig
- Two 6 x 10 inch valves set at 1065 psig
- Two 6 x 10 inch valves set at 1075 psig
- Two 6 x 10 inch valves set at 1102 psig

The safety valves discharge to the environment outside of the turbine building.

2.3 Control Room Indications

- Audio Indication - The control room is equipped with audio indication of main steam safety valve operation via a microphone which is located in the vicinity of the safety valves.

- Pressure - Steam generator pressure and turbine header pressure are available on the front console in the control room. By these indications, the operator can determine if the safety valve setpoint pressure has been exceeded or not.

3.0 TURBINE BYPASS VALVES

3.1 System Design

The turbine bypass valves which discharge steam to the condenser are located on separate branch lines off the main steam lines.

Each steam generator incorporates two turbine bypass valves which exhaust steam to the condenser. These valves have a combined steam dump capacity of 15 percent of the full load steam flow rate. The turbine bypass valves are located at the condenser on a branch line which is cross-connected between the two main steam lines of each steam generator.

3.2 System Operation

During startup, before the generator is synchronized, turbine control is on manual and the turbine bypass valves are in automatic. Steam pressure is controlled at this time by the bypass valves and their pressure control setpoint is established manually. Normal turbine header pressure is 885 psig.

After generator synchronization is accomplished and initial load is picked up by the turbine, the turbine control is usually placed in automatic, which places the turbine in a throttle pressure control mode. As turbine load increases, the bypass valves close. The Integrated Control System (ICS) automatically biases the setpoint of the bypass system to approximately 50 psig greater than normal turbine header pressure. The bypass valves then serve as overpressure relief valves operating at the higher setpoint.

A maximum overpressure relief control setpoint is established for the bypass valve control system. When individual steam generator pressure is higher than 1050 psig, the bypass valves will open.

An interlock prevents the turbine condenser bypass valves from opening on the loss of the condenser which is sensed by either low condenser vacuum or loss of condenser cooling water. Loss of the condenser automatically switches the turbine bypass control to the atmospheric dump valves. Note that in order to maintain condenser vacuum, the turbine gland sealing system which seals the interface between the turbine shaft and the shells must be in operation. This system requires steam for its operation which, during startup, is supplied from the auxiliary boilers at Three Mile Island, and which, during operation, can be supplied from the main steam system.

A steam pressure control interlock is provided by the ICS which increases the bypass valve pressure control setpoint following a reactor trip to 125 psi above the normal setpoint. The purpose of this interlock is to raise the steam pressure, which will reduce the heat transfer in the steam generator, and should prevent excessive cooling of the reactor coolant.

3.3 Control Room Indicators

- Pressure - Turbine header pressure is available to the operator from the selected turbine header pressure transmitter on the front console in the control room; each individual steam line turbine header pressure is recorded on a rear panel recorder in the control room. From these recorders the operator can determine if proper header pressure is being maintained.

- Valve position - Each turbine bypass valve has status light indications received from valve stem limit switches. These status lights are on the front console in the control room and indicate whether a valve is open or closed.

- Valve Controller Hand/Auto Control Station - When control station is in automatic, a meter indicates the demand position for the valves based on the deviation of the turbine header pressure from its setpoint. This control station is on the front console in the control room and controls demand for both the bypass valves and the atmospheric dump valves. Condenser status and valve position are used to determine which valves are being controlled by this controller.

4.0 ATMOSPHERIC DUMP VALVES

4.1 System Design

The atmospheric dump valves, which discharge steam to the environment are located on separate branch lines off the main steam lines.

Each steam generator incorporates one atmospheric dump valve, which is connected to one of the two steam lines from that steam generator. Each valve has a steam dump capacity of 3.2 percent of the full load steam flow rate.

The atmospheric dump valves provide for control of steam pressure when the normal heat sink (main condenser) is unavailable.

4.2 System Operation

An interlock that senses the loss of the normal heat sink (loss of condenser vacuum or loss of condenser cooling water) automatically switches main steam pressure control from the preferred bypass valves to the alternate atmospheric dump valves. The atmospheric dump valves can then be opened or closed by changing the steam pressure setpoint on the turbine header pressure control station, when the Hand/Auto control station for the valves is in automatic.

4.3 Control Room Indicators

- Pressure - Turbine header pressure is available to the operator on the front console in the control room, and this can be used to determine if proper steam pressure is being maintained.

- Valve Position - Each atmospheric dump valve has status light indications received from valve stem limit switches. These status lights are on the front console in the control room and indicate whether the valve is open or closed.

- Valve Controller Hand/Auto Control Station - This control station has a demand meter which indicates, when the control station is in automatic, the demand position for the valve based on the deviation of the turbine header pressure from setpoint. This control station is on the front console in the control room and controls the demand for both the bypass valves and the atmospheric dump valve. Condenser status and valve positions are being used to determine which valves are being controlled by the controller.

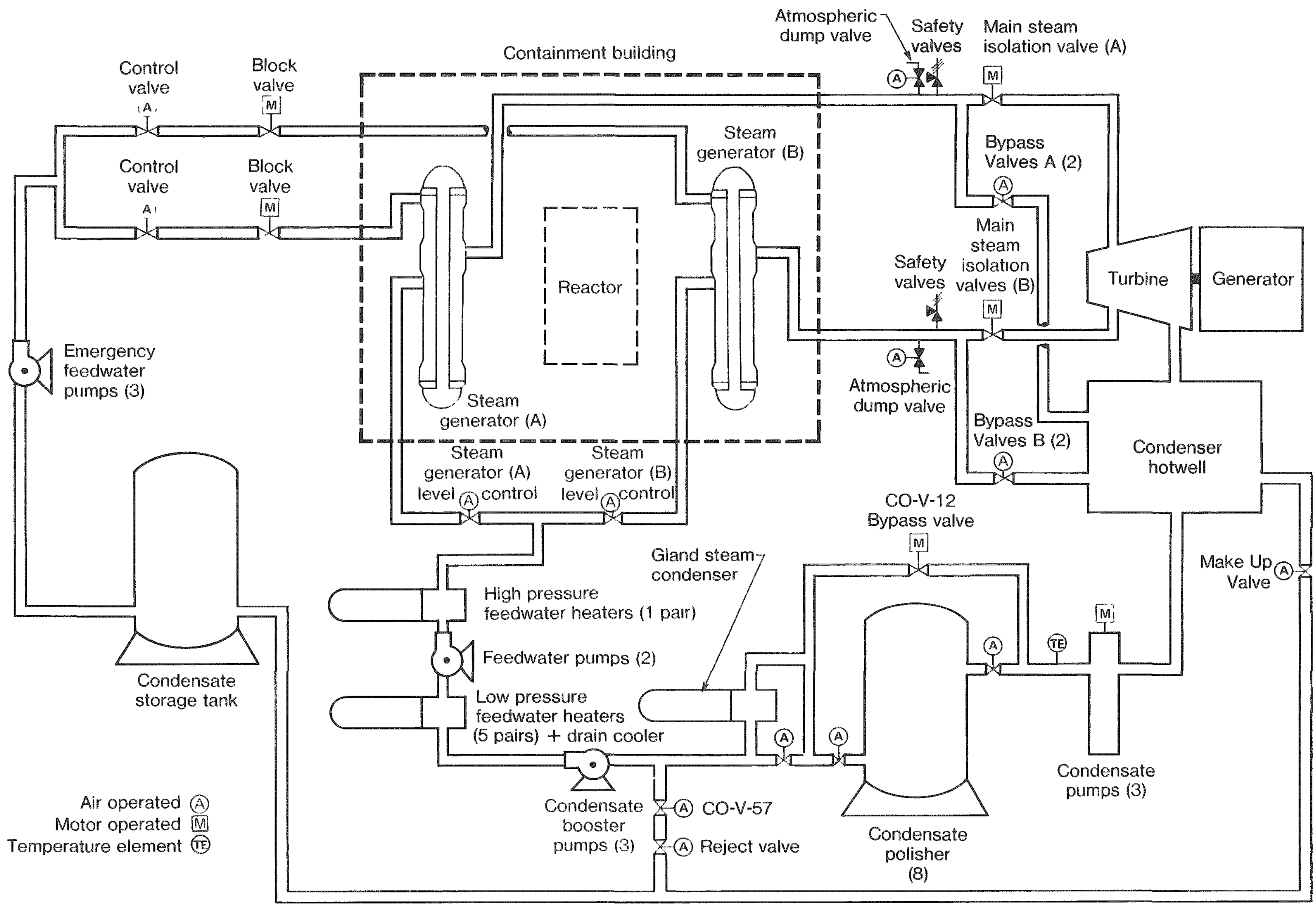


Figure STEAM DUMP-1

APPENDIX TH

SYSTEM THERMAL-HYDRAULIC BEHAVIOR
THREE MILE ISLAND UNIT 2
LOSS OF COOLANT ACCIDENT, MARCH 28, 1979

1.0 INTRODUCTION

1.1 Purpose and Scope

The purpose of this appendix is to describe the primary system thermal-hydraulic behavior of the Three Mile Island Unit 2 plant during the accident of March 28, 1979. The scope of the evaluation includes the primary coolant system and the steam generator secondary side out to the steam and feedwater nozzles. The core behavior is discussed in Appendix CI; for the purposes of this discussion the core is treated as a heat source to the primary system fluid. The interpretations discussed are in most cases suggested by a cross comparison of the transient behavior of several parameters. In some cases they are judgmental and not backed up by detailed analysis. Thus some may be proven invalid by additional analysis. The philosophy in documenting this evaluation has been to include such items to prevent the issues from being lost and to provide guidance to organizations which will be performing more detailed analyses of the accident.

1.2 Organization of Appendix TH

The time period of the accident is divided into six major phases, as defined below, covering the period from the initiation of the accident to the continuous operation of reactor coolant pump 1A about sixteen hours later. The selection of phases of the accident is somewhat arbitrary. The intent is to divide the accident time into intervals representing various operating modes that occured during the accident. The time for events occurring is given in (hours:minutes) referenced to the start of the accident.

Phase 1 - Initiation. This phase covers the period from the turbine trip to the shutdown of the loop B reactor coolant pumps. Forced circulation of primary coolant was maintained in all loops during this phase with continuous loss of primary system coolant. Time period: 00:00 to 1:13.

Phase 2 - Continued Depressurization. This phase covers the period from the trip of the loop B reactor coolant pumps to the closure of the relief block valve ahead of the electromatic relief valve. The primary system coolant continued to degrade, approaching the conditions which resulted in major damage to the core.
Time period: 1:13 to 2:22.

Phase 3 - Initial Repressurization. This phase covers the period from the pressurizer relief block valve closure to the start of sustained operation of two high pressure injection pumps. It appears that the major core damage occurred during this phase. Time period: 2:22 to 4:27.

Phase 4 - Sustained High Pressure Injection. This phase covers the period from the start of sustained operation of two high pressure injection pumps to the start of system depressurization. During most of this time the system was being cooled primarily by water flowing into the cold leg injection nozzles and out the top of the pressurizer.
Time period: 4:27 to 7:38.

Phase 5 - Extended Depressurization. This phase covers the period from the start of the sustained depressurization to the beginning of a gradual repressurization of the system. During this phase, a large amount of the hydrogen formed by zirconium/water reaction in the core was vented from the primary system.

Time period: 7:38 to 11:08

Phase 6 - Repressurization and Recovery. This phase covers the period from the beginning of gradual repressurization to the continuous operation of reactor coolant pump 1A, establishing a stable long-term cooling mode via the loop A steam generator. The operation of high pressure injection pumps during the middle of this period, combined with the venting of hydrogen during phase 5, resulted in conditions which set the stage for recovery of the system.
Time period: 11:08 to 15:50

A series of illustrations is included at the end of Section 2 of this appendix, depicting the evolution of the conditions in the primary system and the steam generators during the course of the accident. These illustrations are highly simplified representations of the actual systems and the conditions which existed during the accident. Some of the liquid distributions are judgmental and may be substantially in error. As with the discussion, the philosophy in producing them was to make the best estimates based on currently available information.

1.3 Data Sources

The primary source of data for this evaluation has been the "reactimeter". This monitoring computer provided high-quality digital data on a three-second sampling interval of key plant parameters throughout the accident. The availability of a digital record, which allowed expansion of scales for a more detailed evaluation of key events, was an invaluable aid in the analysis of the accident. Figures TH1 through TH7 were generated from the reactimeter record. A detailed description of the reactimeter is provided in Appendix PDS.

Some key variables were not available from the reactimeter, thus extensive use was made of control room strip chart recorder information. The most notable of these variables were wide-range primary system pressure, containment pressure, and source- and intermediate-range neutron detector signals. Additional information on primary system temperatures from a multipoint recorder was identified and incorporated.

The plant computer was an additional important source of information via the alarm and utility typewriters. This computer, also described in Appendix PDS, provided key pieces of information on state changes occurring in alarm variables, and when data was requested by the operators. Figure TH8 shows a composite plot of primary coolant system pressure pieced together from the best available data extracted from the reactimeter, the plant computer, and the wide-range pressure strip chart.

The reactimeter measurement locations are shown on Figure TH9, and all nozzle locations of interest are shown on Figure TH10. Wide range hot leg temperatures taken from a multipoint recorder are shown in Figure TH20. These data should be used with caution as they are not of the quality of the reactimeter data. Note that the initial temperature is reading approximately 30°F lower than the actual temperature.

2.0 SUMMARY AND CONCLUSIONS

2.1 Summary

The central factor in the accident was the sustained loss of primary system water from the stuck-open electromatic relief valve and through letdown flow established by the operators in an attempt to bring the pressurizer water level back into the normal range. The continuing loss of primary system water led to conditions, approximately two hours into the event, wherein the water level in the reactor vessel dropped below a substantial fraction of the fuel in the reactor core. Under these conditions, the continuing generation of decay heat in the fuel could not be effectively removed by the primary coolant, and core temperatures began to rise.

As the core temperature continued to increase, an exothermic reaction between the zirconium in the fuel cladding and the surrounding steam contributed additional energy, increasing the heatup of the fuel. This reaction produced zirconium oxide and hydrogen gas. Some of the hydrogen dissolved into the water, and the rest of it remained in a gaseous state and collected in the upper sections of the primary system. The presence of large amounts of noncondensible gas in the primary system was a major factor in the subsequent thermal-hydraulic behavior.

A large fraction of the hydrogen collected in the upper portions of the hot legs and steam generator tube bundles on the primary side. This precluded effective heat transfer through the steam generators, with minor temporary exceptions, until the operation of reactor coolant pump 1A began at 15:50.

During the middle time period, a protracted attempt to repressurize and recover the system was unsuccessful due to the noncondensible gas in the hot legs. A sustained depressurization of the primary system allowed a large quantity of the hydrogen to expand and pass out of the system through the open electromatic relief valve at the top of the pressurizer. Approximately eleven and one-half hours into the accident, the sporadic operation of a second high pressure injection pump was terminated for a period of nearly two hours. Except for two brief exceptions, the relief block valve was also closed. Since the gas-blocked steam generators were prevented from removing heat, there was no significant heat removal from the reactor system during this period, and primary system parameters indicate that the system was again approaching conditions which could lead to additional damage to the core. When the primary system was later repressurized, beginning at 13:23 with the start of makeup pump MU-P-1C, the remaining hydrogen was compressed to a small enough volume to allow the operation of a reactor coolant pump. The resulting forced flow of liquid swept the hydrogen from the tops of the hot legs, with some dissolving into the water and some collecting in the upper head of the reactor vessel. Clearing the loops of gaseous hydrogen established a stable mode of cooling with forced flow and steaming through the loop A steam generator.

2.2 Conclusions

The bases for the conclusions identified below are discussed in Section 3. Conclusions are grouped according to the most relevant phase.

Phase 1 - Initiation.

- The electromatic relief valve failed in the fully open position.

- The temporary unavailability of auxiliary feedwater was not a significant direct contributor to core damage.

Phase 2 - Continued Depressurization.

- Comparable amounts of primary water were removed by the stuck-open electromatic relief valve and by continued letdown flow.

- Core flood system actuation may have been prevented by operator closure of isolation valves in the core flood system.

- The primary system began to repressurize before the relief block valve was closed, possibly indicating the initiation of hydrogen evolution in the core.

Phase 3 - Initial Repressurization.

- A manual initiation of safety injection occurred at 3:20.

- Steam generator behavior during this phase indicates the presence of noncondensibles.

- An apparent core collapse is indicated by the response of the primary system at 3:45.

Phase 4 - Sustained High Pressure Injection.

- Core cooling was maintained during this phase by the flow of high pressure injection water into the cold legs, through the reactor vessel, and out the top of the pressurizer.

- The steam generators remained blocked by noncondensibles in the upper regions of the hot legs.

Phase 5 - Extended Depressurization.

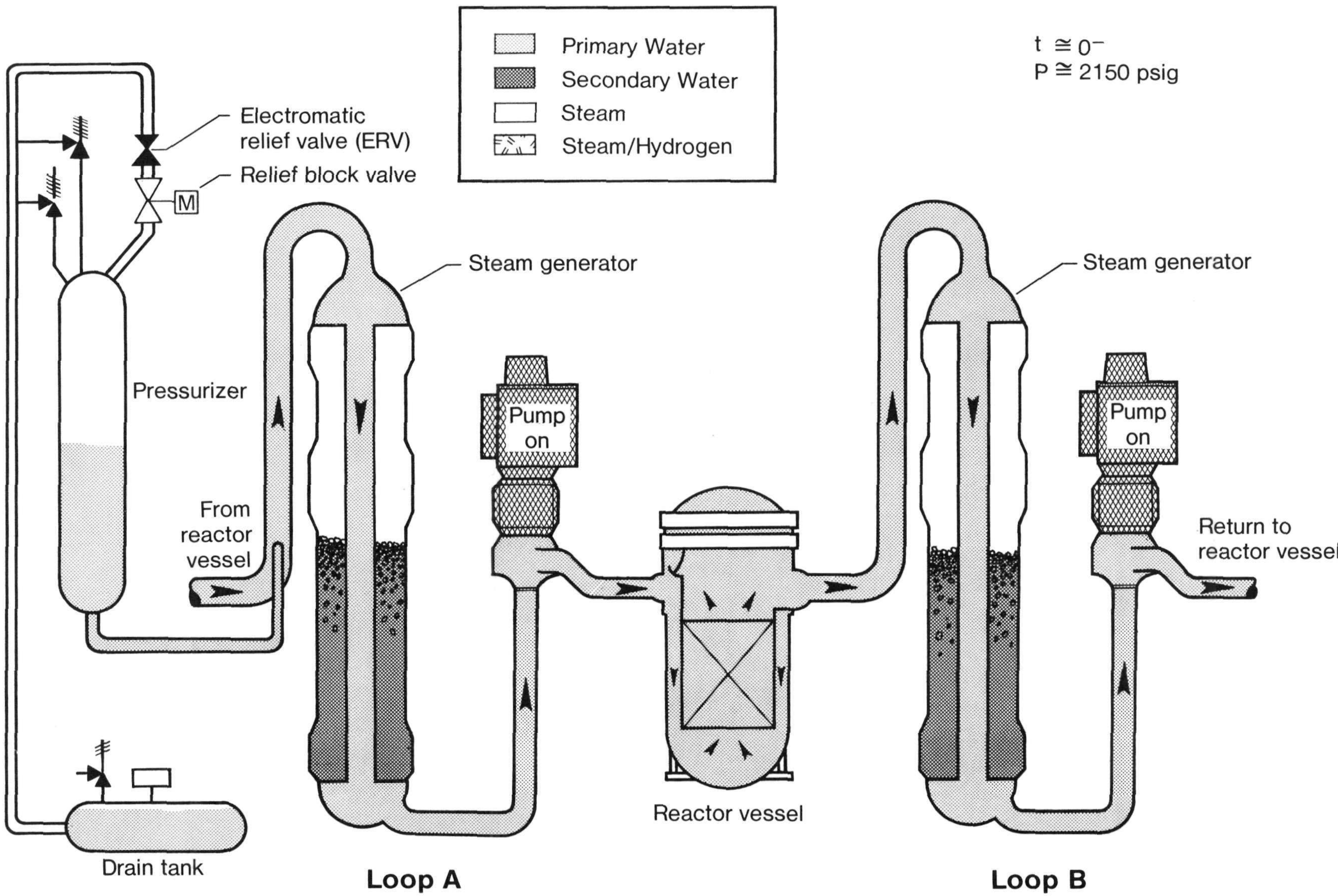
- Core cooling was maintained by intermittent operation of makeup pump MU-P-1C and to a lesser extent by actuation of the core flood system.

- Large amounts of hydrogen were vented to the containment during the depressurization.

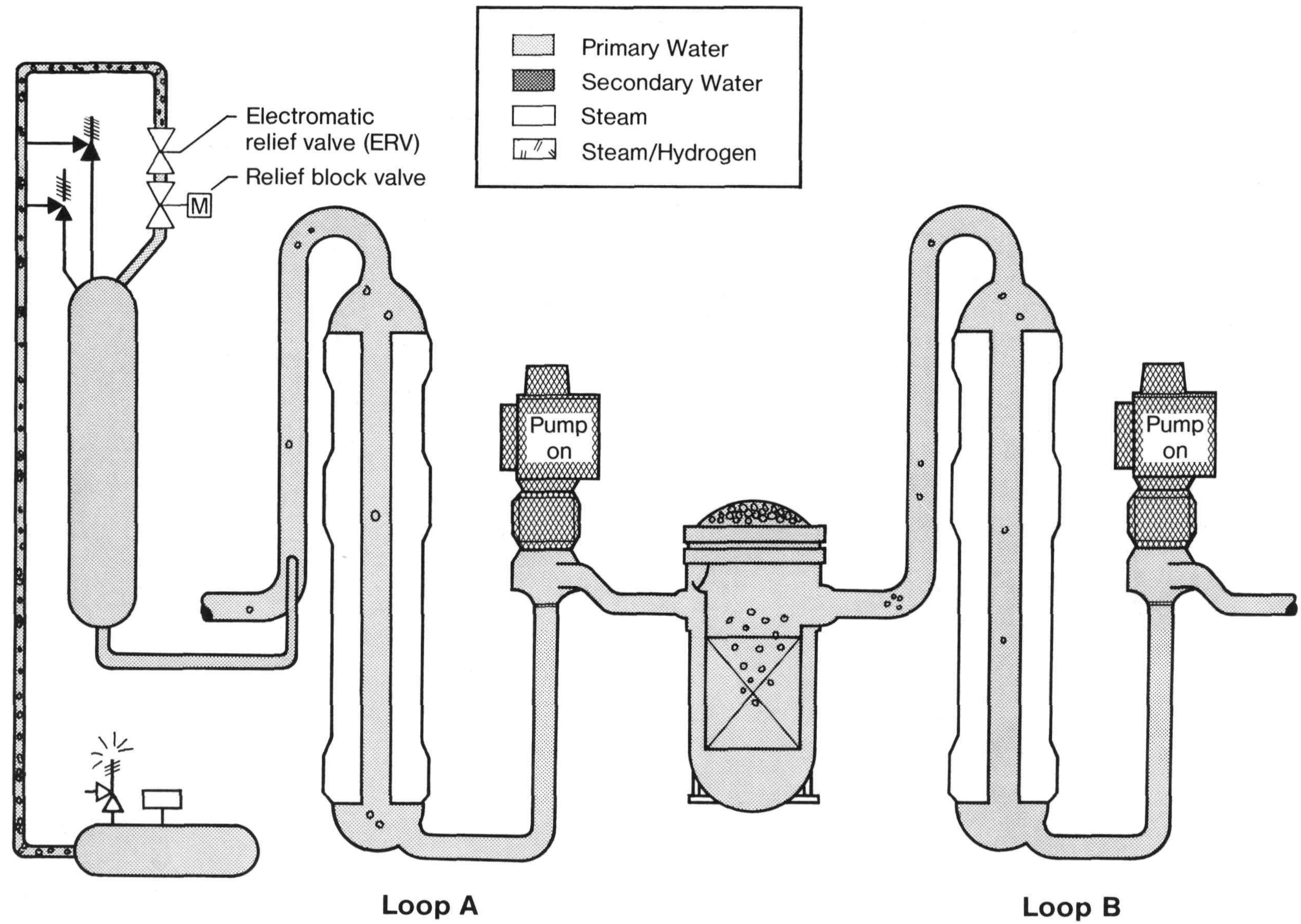
Phase 6 - Repressurization and Recovery.

- There was an extended period (11:35 to 13:23) during this phase with no significant heat removal from the primary system except for two brief openings of the relief block valve.

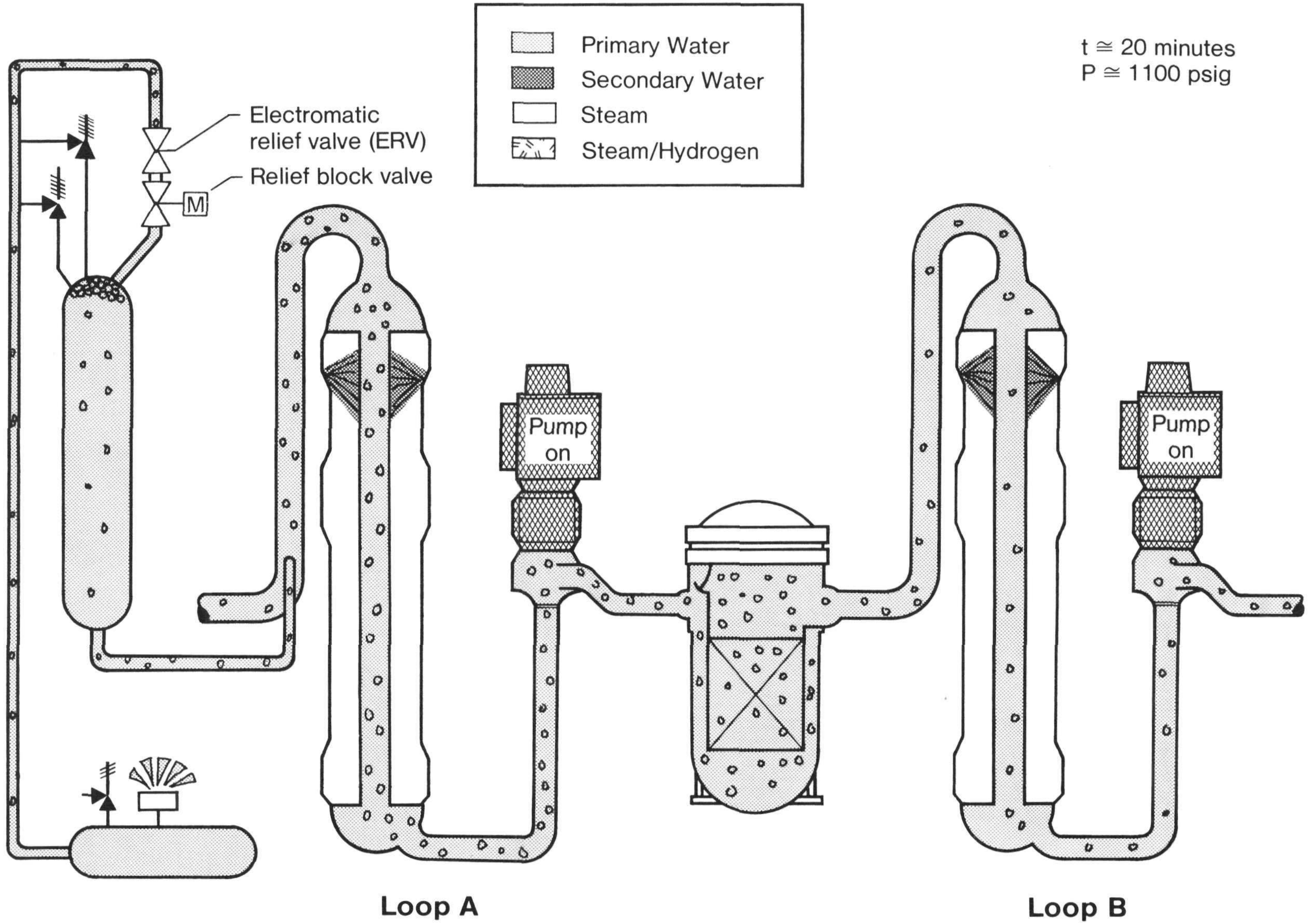
- Repressurization of the primary system by high pressure injection, following the venting of hydrogen in phase 5, allowed reestablishment of heat removal through the steam generators by the operation of a reactor coolant pump.



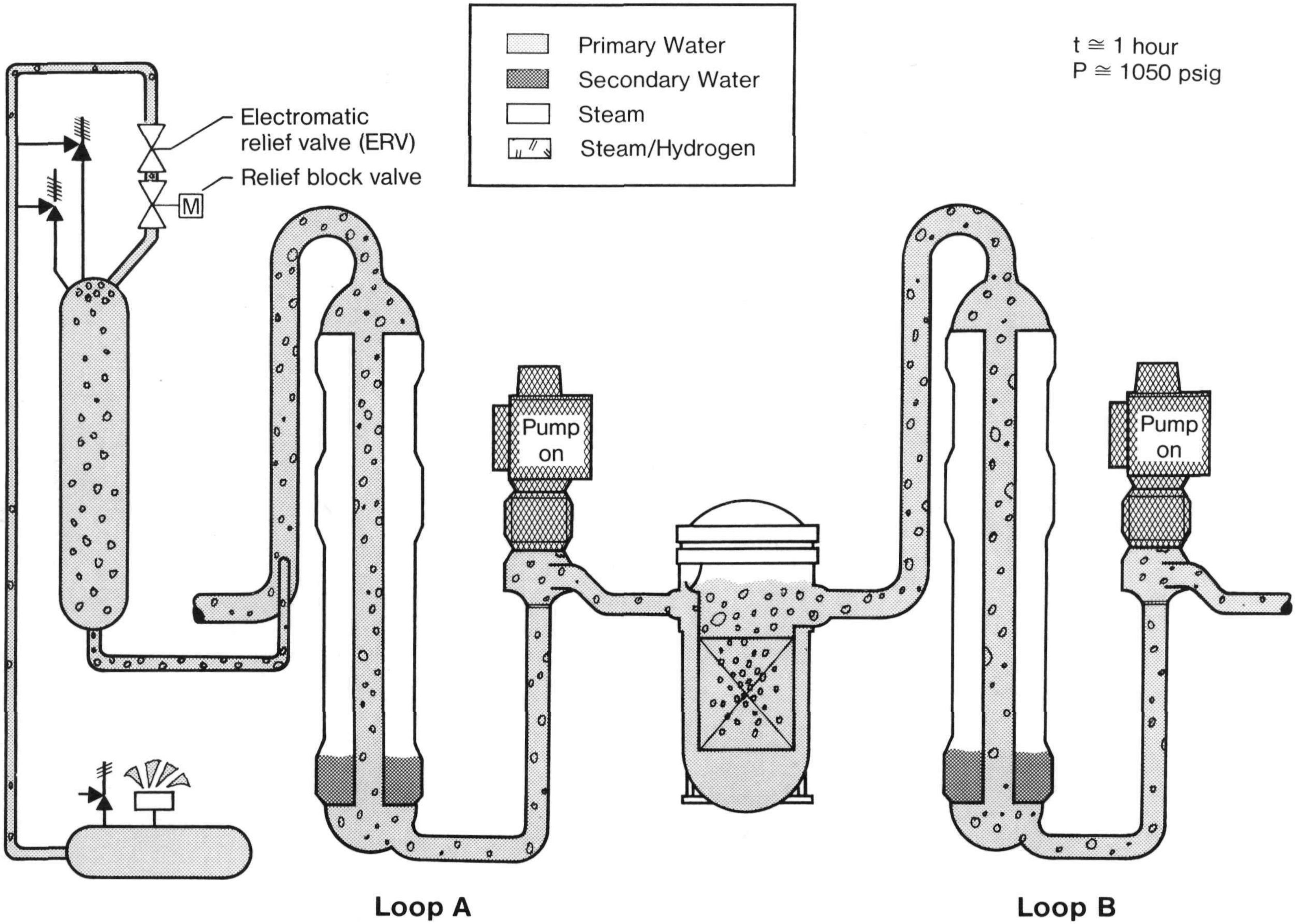
Initial conditions — system steady at 97% power operating conditions.



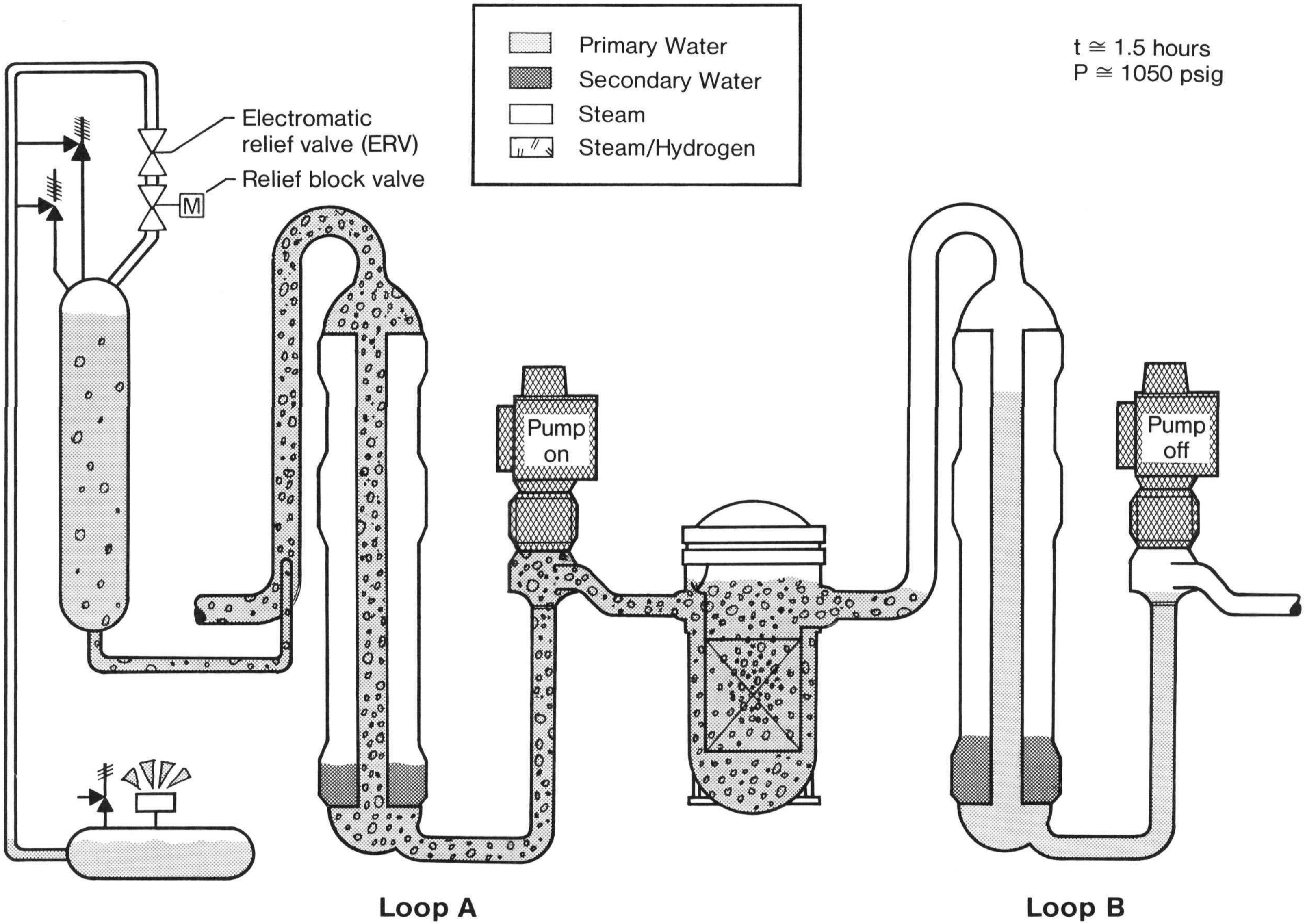
System nearly liquid solid. Liquid expanding with increasing temperatures.



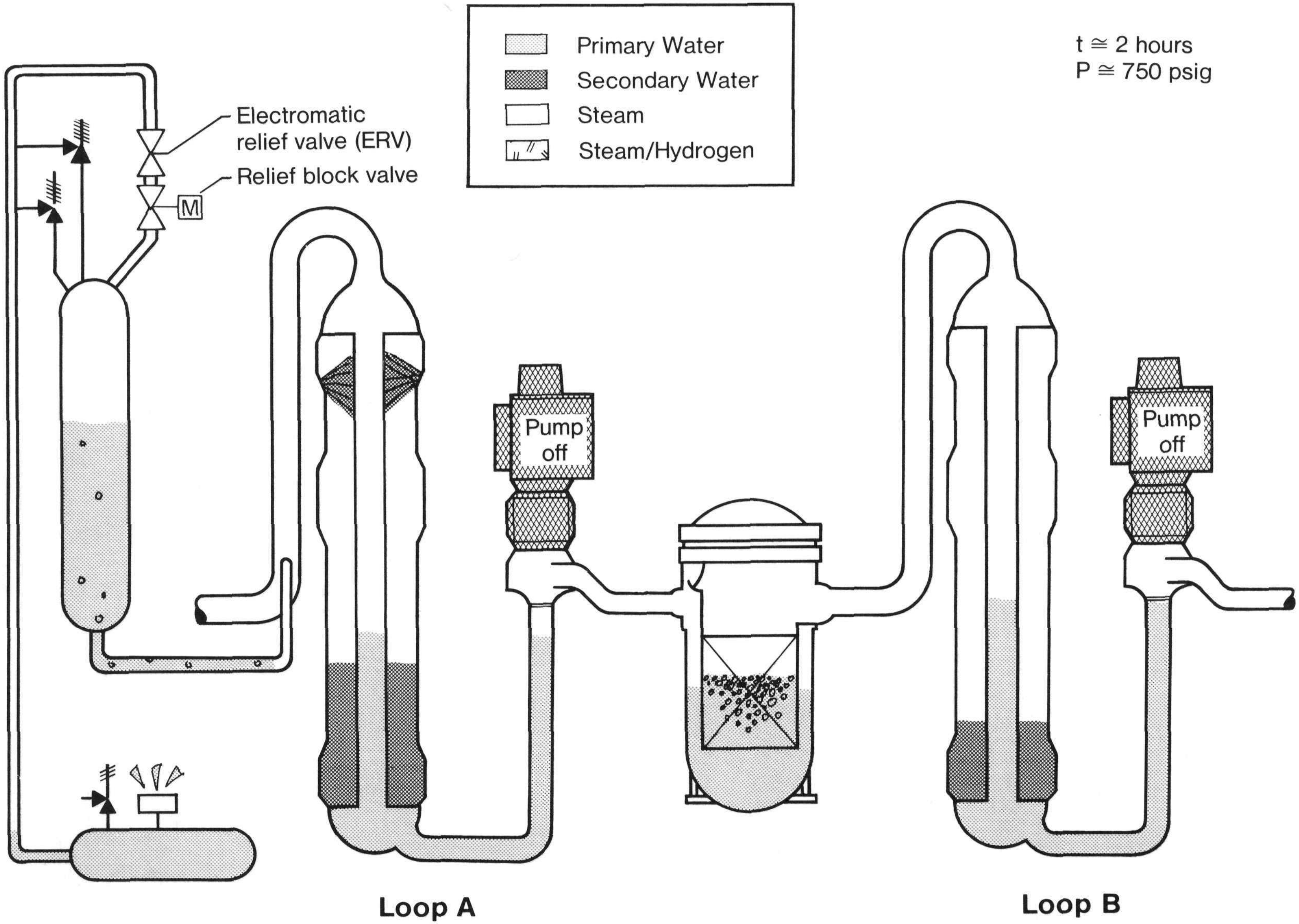
Cooling down via steam generators. Primary system voids increasing.



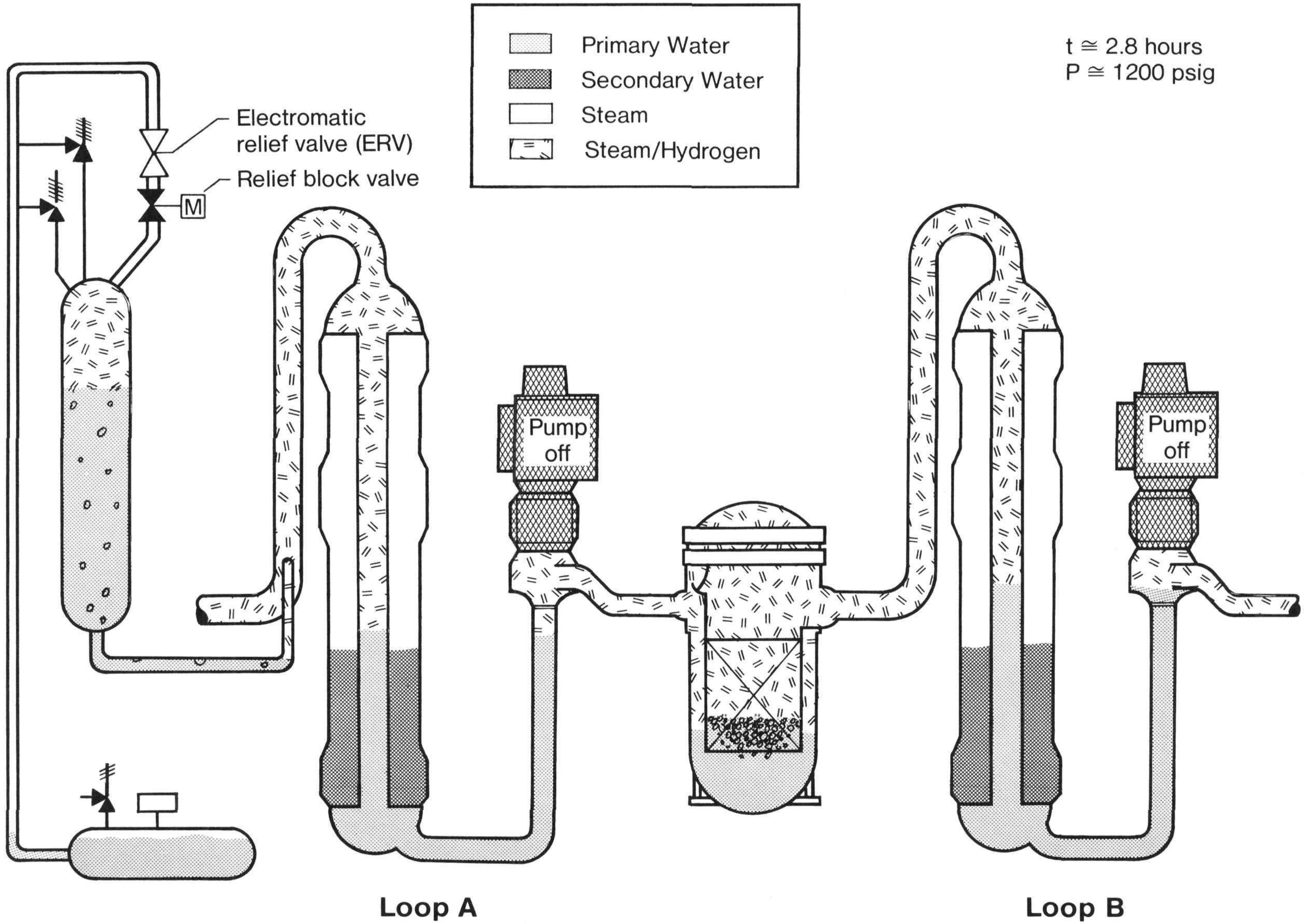
Primary system pressure and temperatures nearly constant following secondary system conditions. Primary voids increasing.



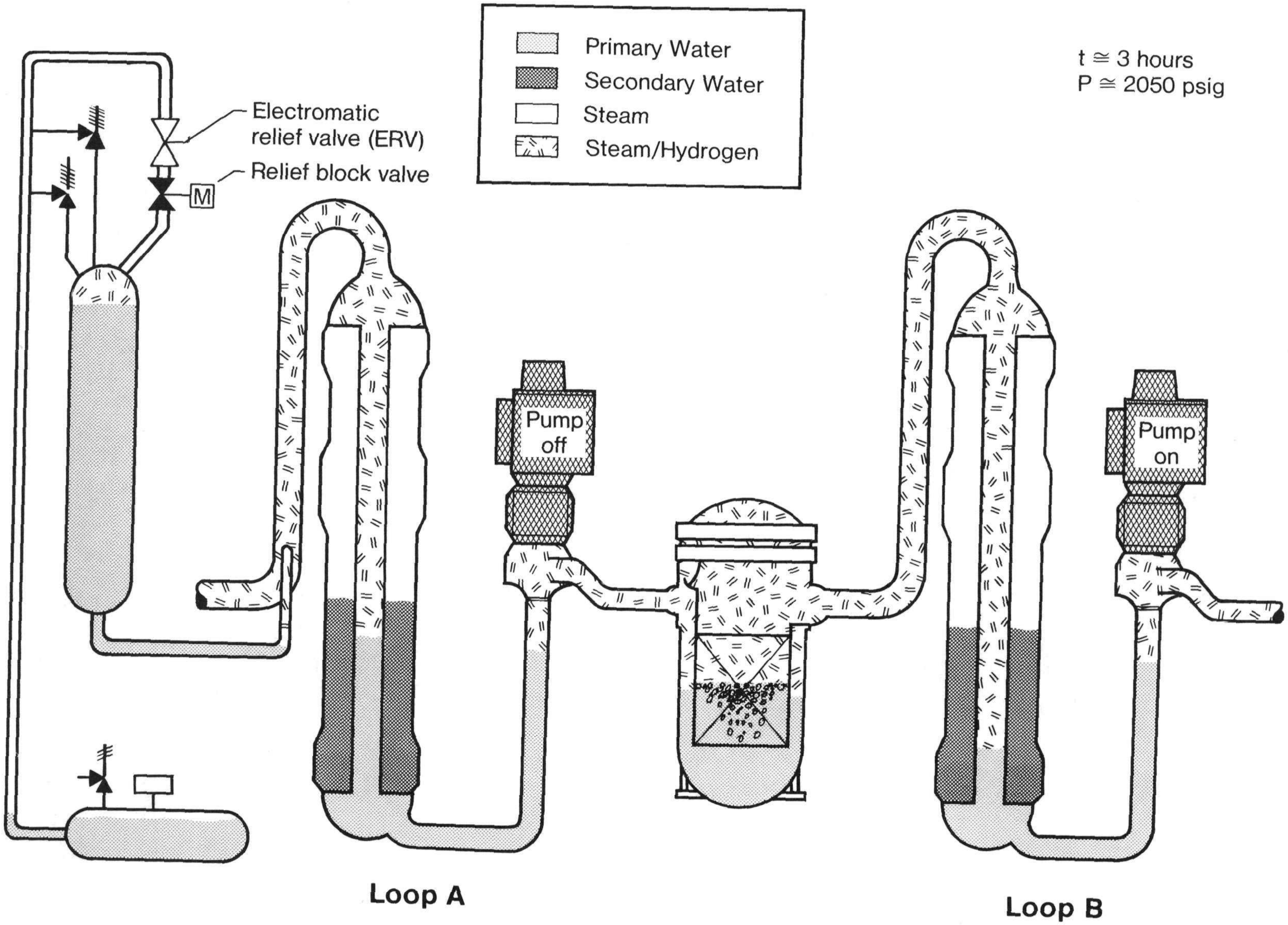
Loop A pumps operating. Loop B stagnant after shutdown of Loop B pumps. Primary voids increasing.



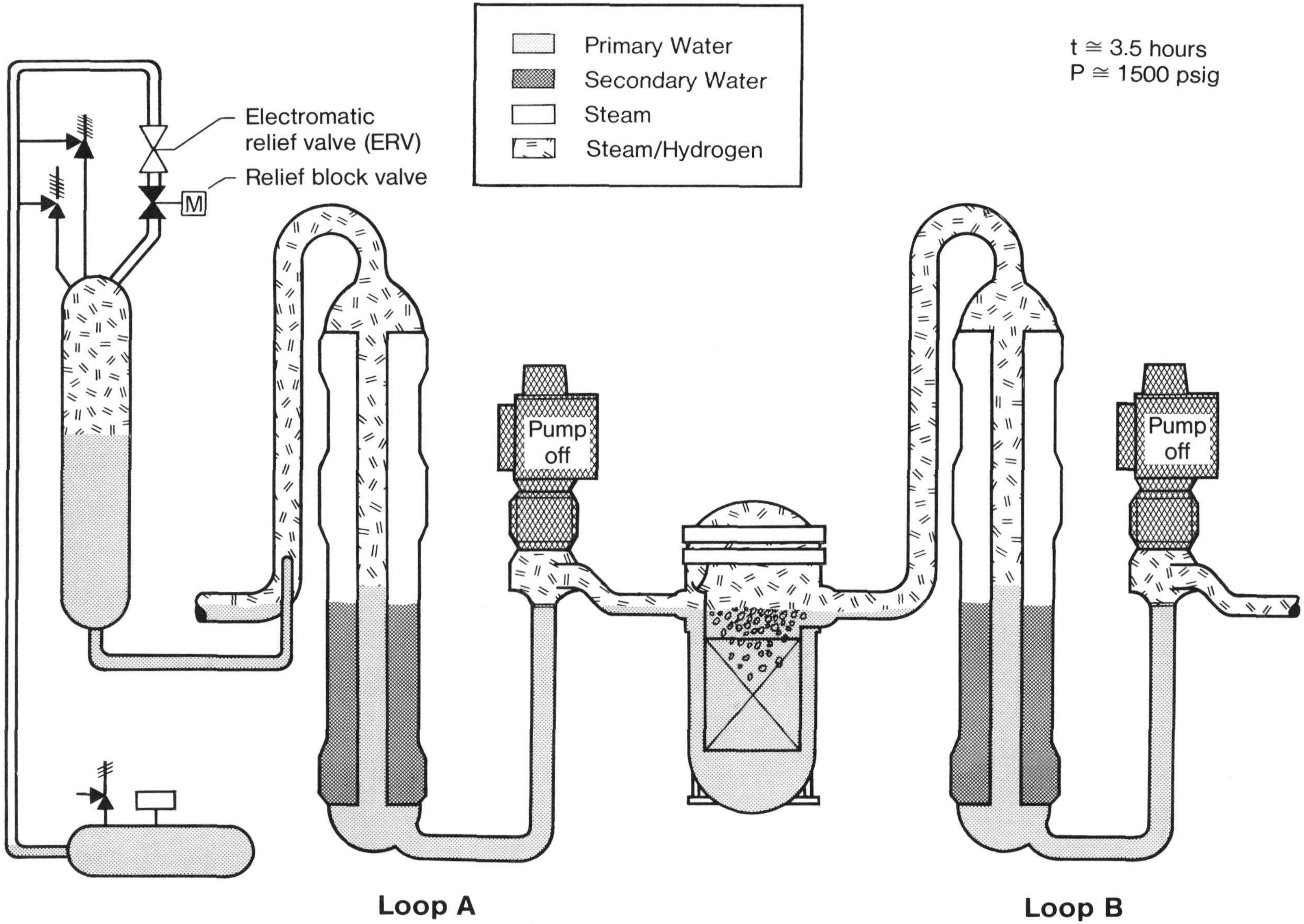
All pumps off. Reactor core drying out and heating up. Superheated steam flowing to pressurizer and to one steam generator and condensing.



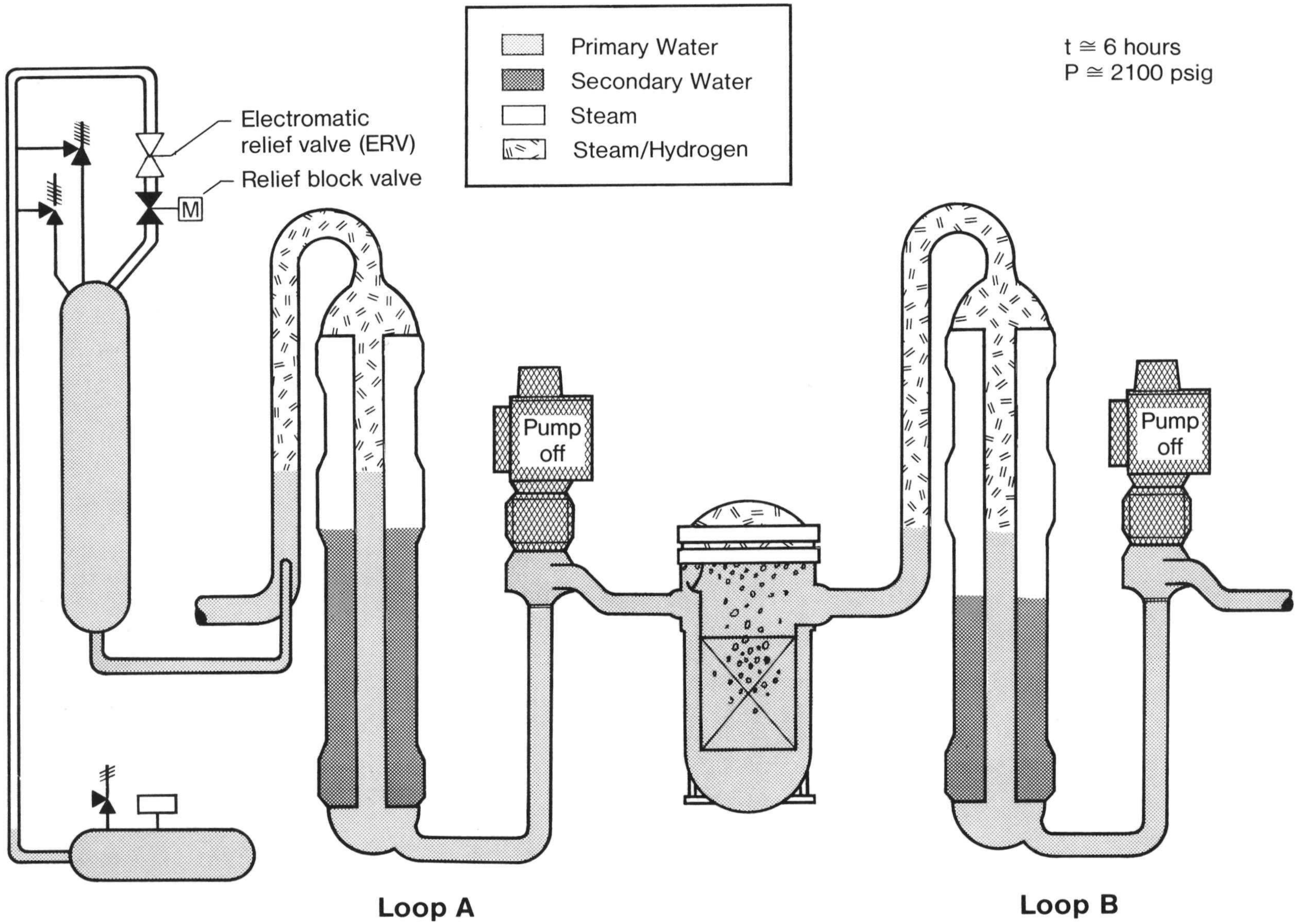
Core dryout and heatup continuing. Hydrogen generation by zirconium/water reactor in hotter regions.



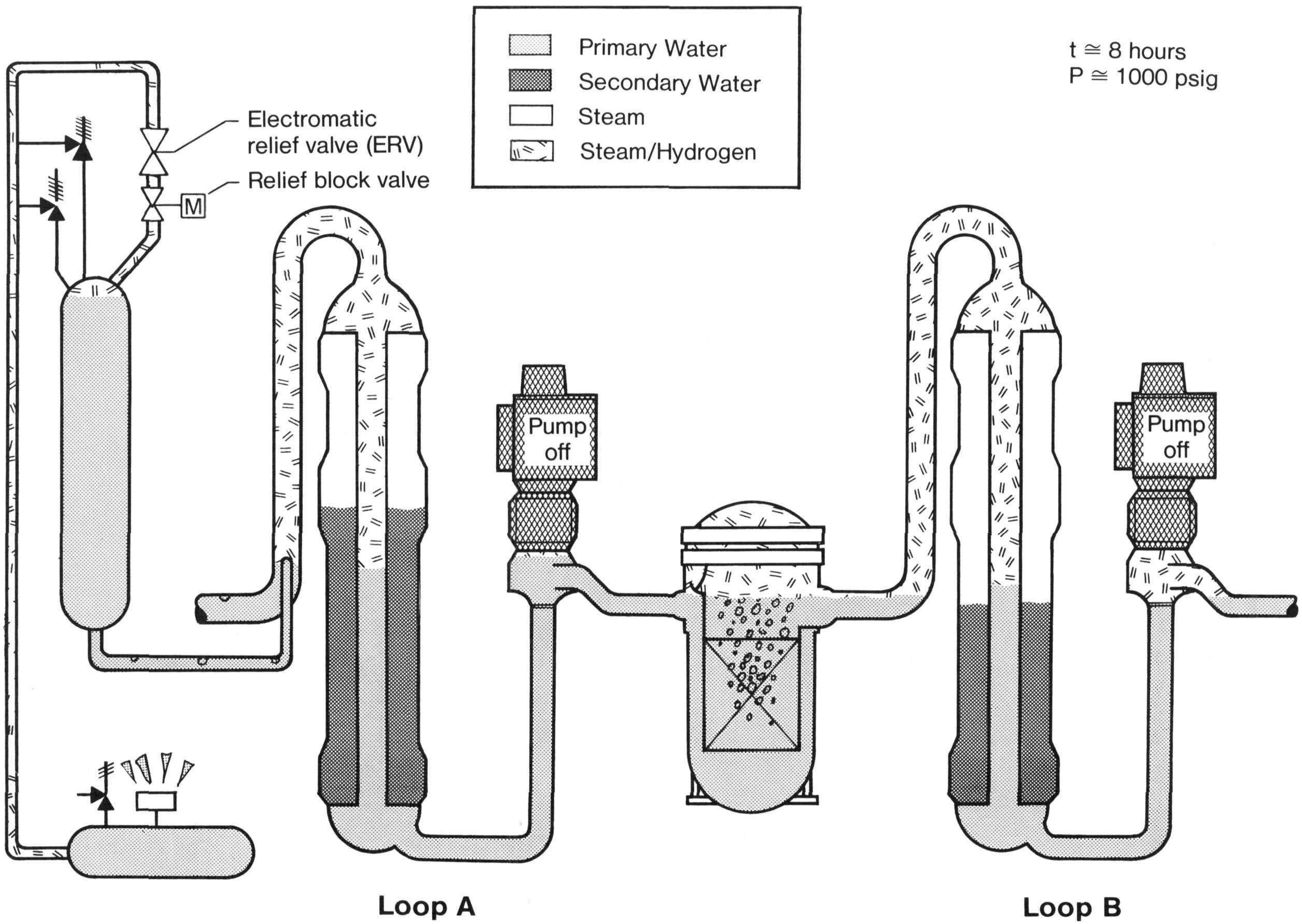
Core partially quenched by fluid during Loop B pump start. Heatup resumes.



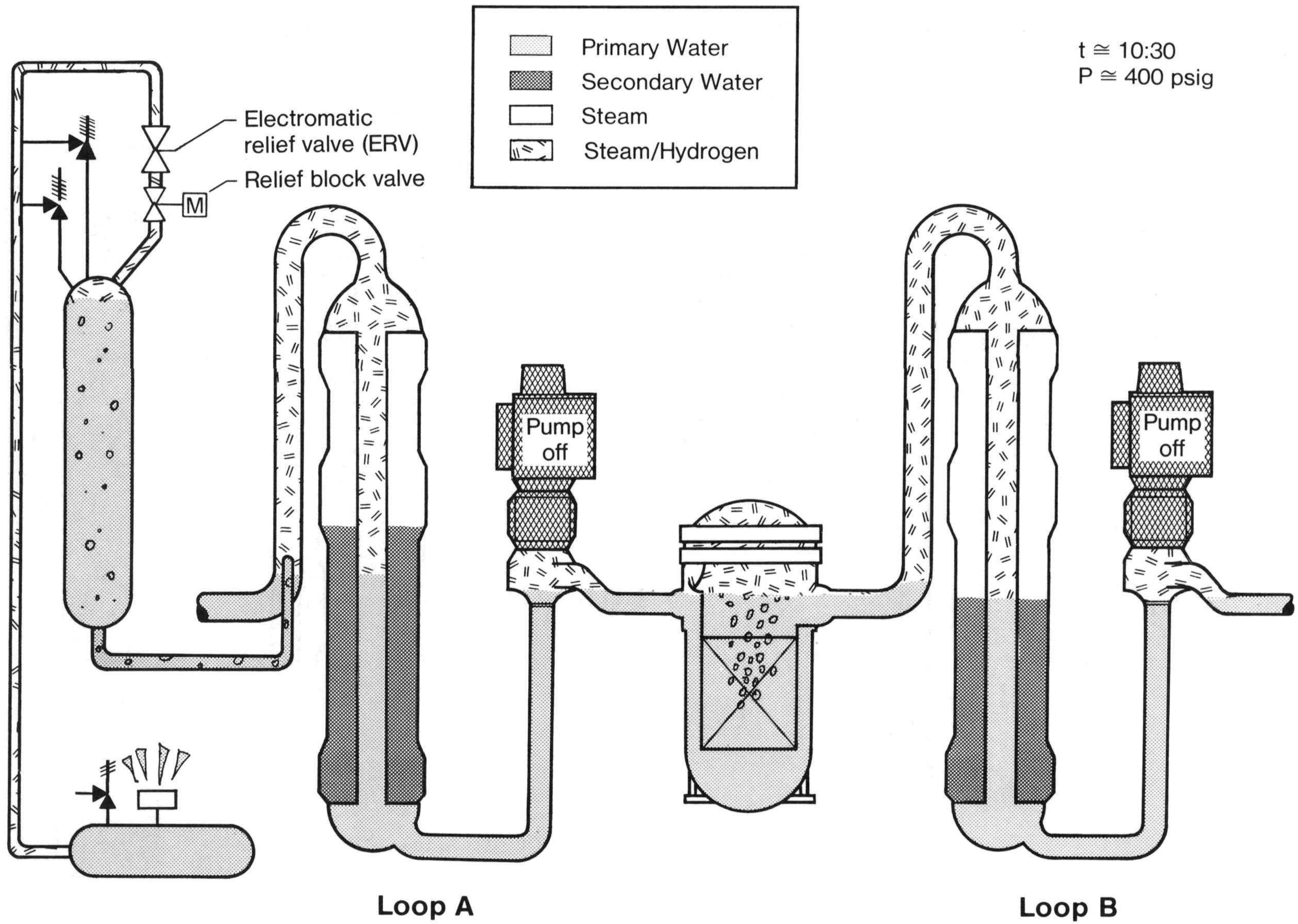
Reactor vessel refilled by manual initiation of safety injection. Core temperatures decreasing.



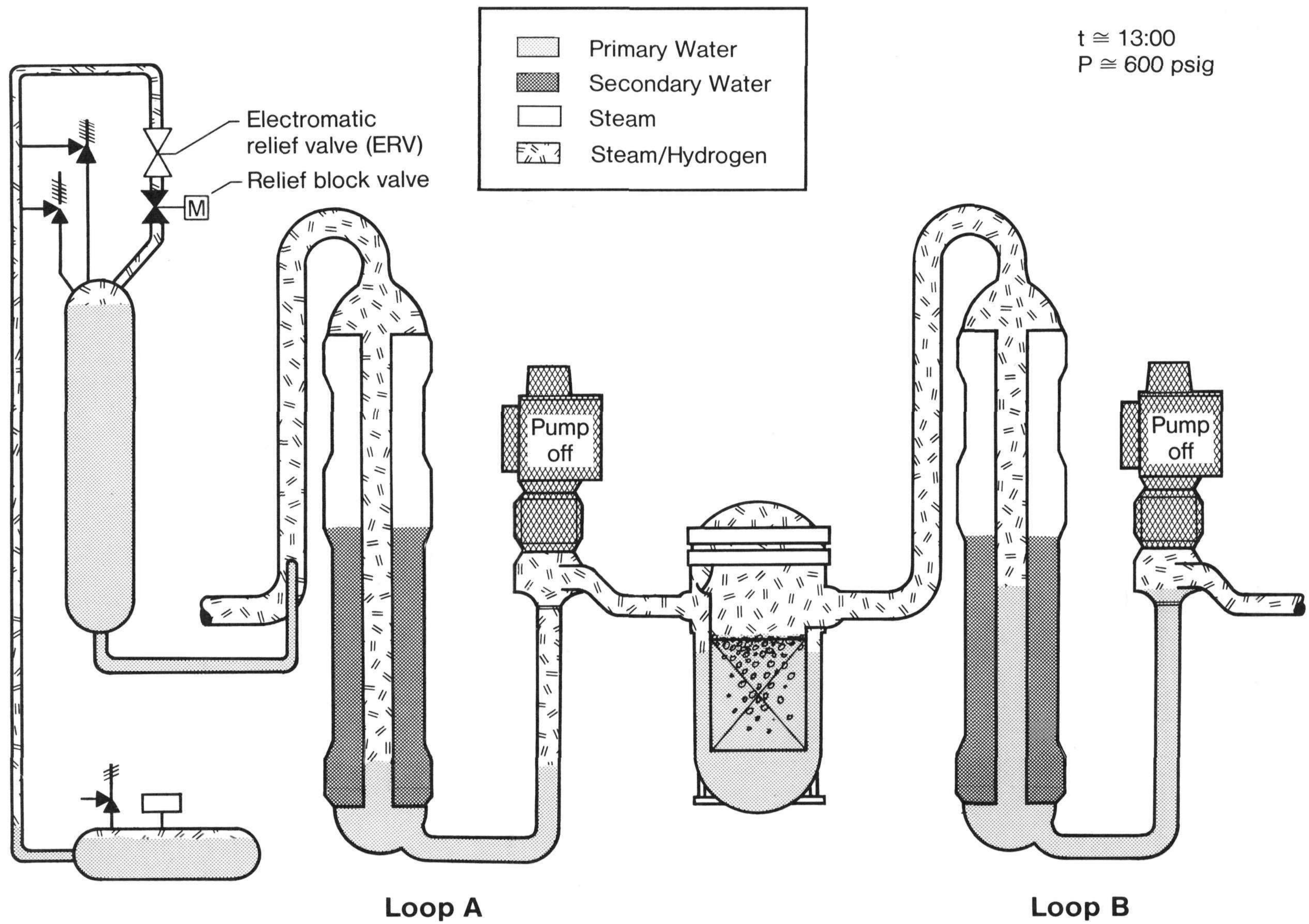
System presssurized by high pressure injection system. Intermittent liquid release through top of pressurizer. Heat removal by heatup of injected water. Steam generator heat transfer blocked by hydrogen.



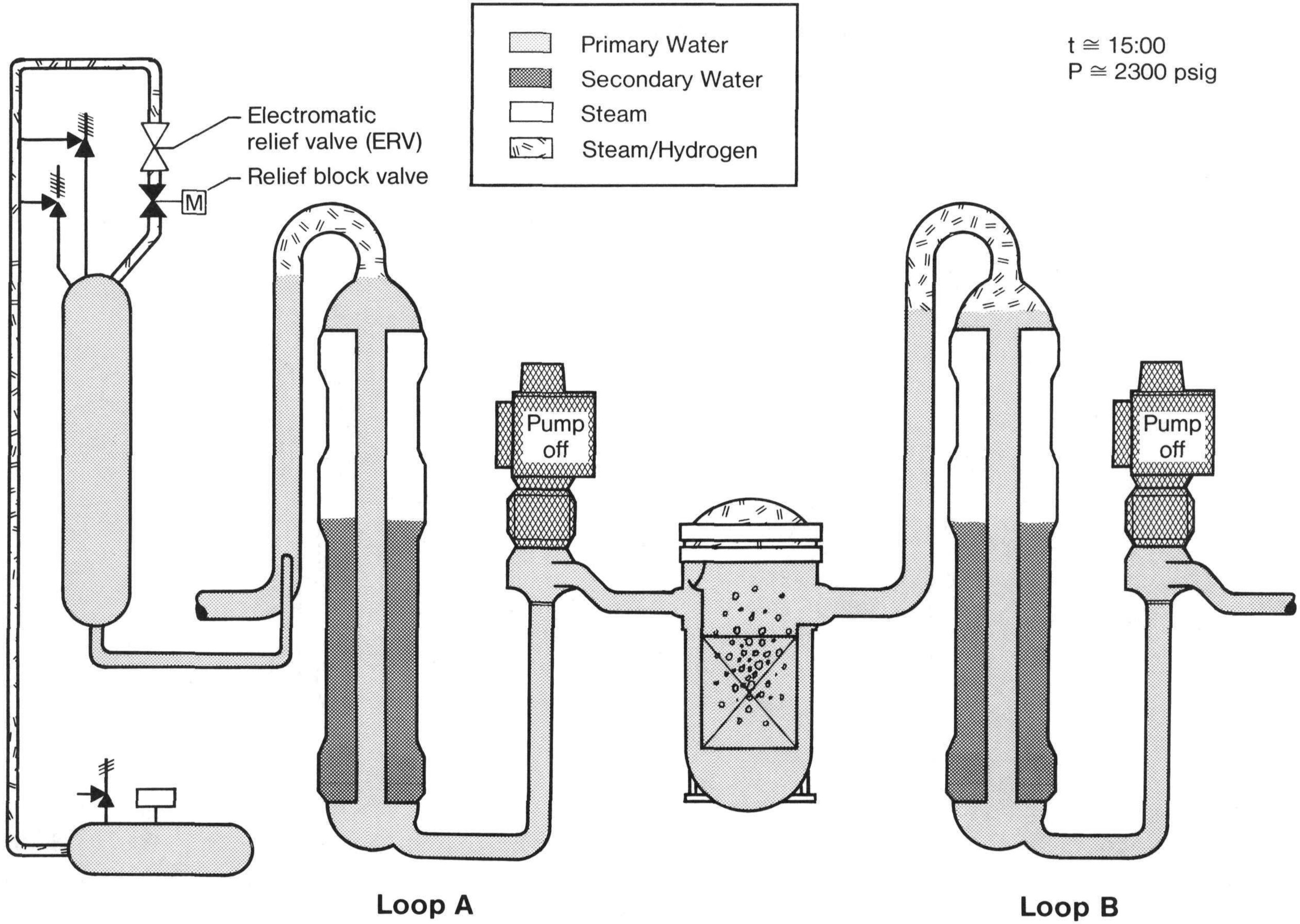
Primary system depressurizing and releasing hydrogen through the pressurizer into the containment.



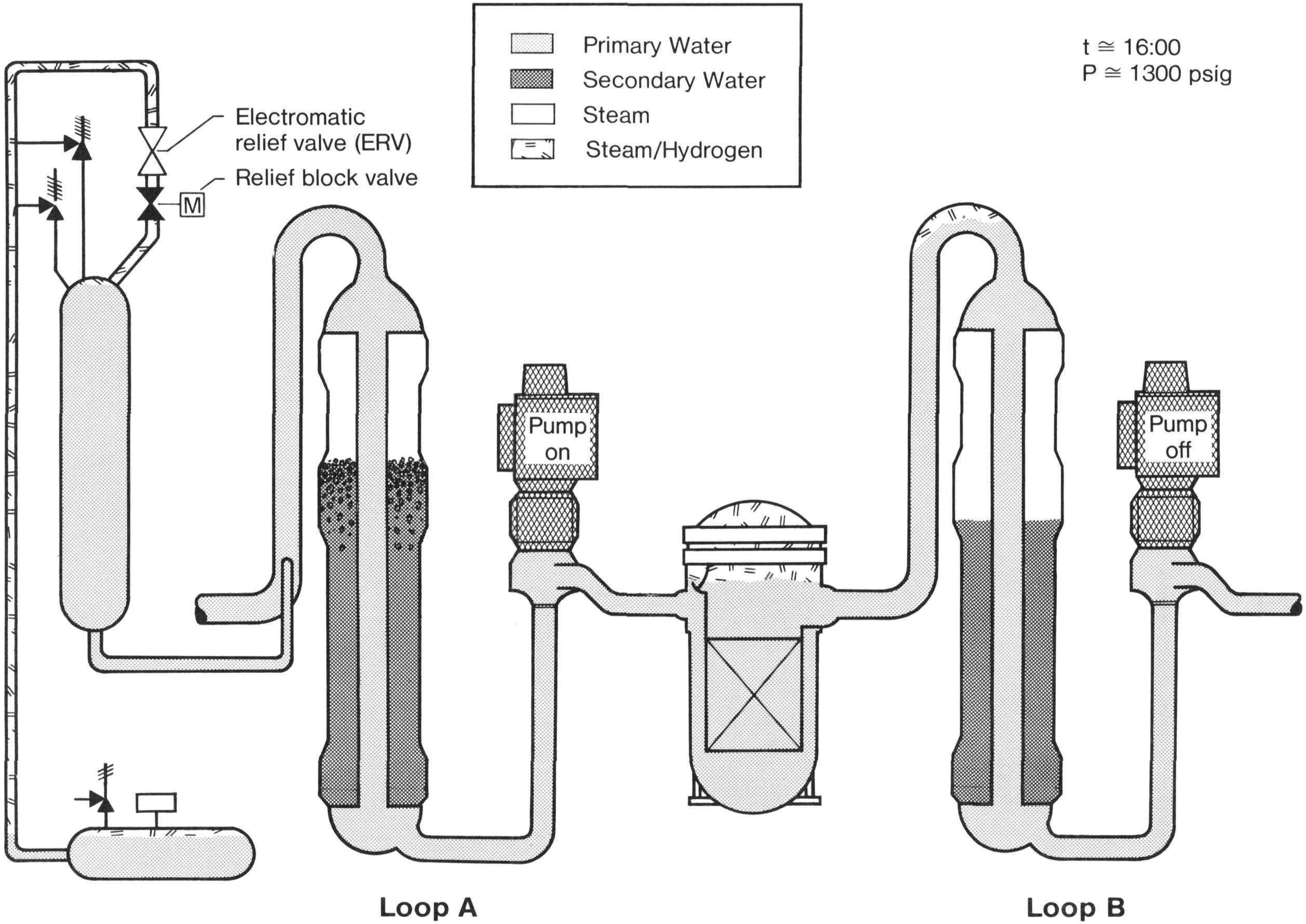
Primary system pressure reaches a miminum as pressurizer temperature returns to saturated condition. Liquid in system maintained during depressurization by high pressure injection and core flood tanks.



Steam generators blocked by hydrogen. Reactor vessel level decreasing. Primary system pressure increasing. Minimal make-up flow.



System repressurized by high injection. Natural circulation to steam generators blocked by hydrogen.



Forced circulation reestablished in Loop A with heat removal via Loop A steam generator.

3.0 SYSTEM THERMAL-HYDRAULIC BEHAVIOR

There were three distinct, abnormal conditions which could be considered in combination as the initiator of the accident:

1. Problems in the condensate polishing system leading to a nearly simultaneous trip of the main feedwater pumps and the turbine.

2. Unavailability of auxiliary feedwater following the reactor trip due to improper positioning of isolation valves in the auxiliary feedwater system.

3. Failure of the electromatic relief valve to reclose when primary system pressure dropped below the reclosure set-point.

The first condition, loss of normal feedwater, falls within the definition of a transient that is anticipated in the design of the plant, with an expected frequency of occurance of approximately once per year. The plant is expected to sustain this event without damage.

The third condition in combination with the first is much less likely, but it has a significant probability of occurance. It had actually occurred three times prior to the accident, once at Three Mile Island Unit 2 (1). This combination of events falls within the design requirements of the plant under the single failure criterion (2). This design criterion requires the plant to be able to withstand the accidents defined within the design basis, in combination with all failures which would be expected as a consequence of the initiating event, plus the additional random failure of any component which would be required to perform an active function. While the combination of the first two events would fall within the guidelines of the single failure criterion, it is not apparent that the criterion was applied to this event combination. This is likely to be a result of the requirement that no credit be taken in the safety analysis for non-safety class systems, such as the electromatic relief valve, whose function would mitigate the course of the accident. Since the analysis procedure does not allow the assumption that the electromatic relief valve would open during the event, the possibility that it would fail to close was not considered in the accident analysis.

The second condition resulted in a temporary unavailability of auxiliary feedwater for a period of almost eight minutes at the beginning of the accident. This factor was not a significant, direct contributor to the damage of the core during the accident. However, it was a key factor in terms of additional problems for the operators, and it may have contributed to the failure of the operators to recognize the course of the event and to take the actions which would have resulted in a controlled shutdown of the plant. The consequences of this combination of initiators are discussed in detail in the following subsections.

3.1 Phase 1 - Initiation

The first phase of the accident is defined as the time period from the turbine trip (00:00) to the shutdown of the two loop B reactor coolant pumps (1:13). The length of this phase is one hour and 13 minutes.

3.1.1 Summary. The accident was originated by problems in the condensate polishing system leading to the nearly simultaneous trips of the main feedwater pumps and the turbine. These events led to increases in primary system pressure and temperature, which caused the pressurizer electromatic relief valve to open. At 7.2 seconds, the reactor scrammed on high pressure indication. After 12 seconds, the primary system pressure had decreased to the electromatic relief valve closure setpoint, but the valve failed to close and remained stuck in the fully open position, thus continuing to depressurize the primary system and allow coolant to escape.

After 30 seconds, the water levels in the steam generators reached the set point for opening the control valves to initiate auxiliary feedwater flow. However, feedwater was not delivered because of earlier improper positioning of the block valves in the auxiliary feedwater system. This situation was not recognized until about eight minutes into the accident. As a result, since there was no secondary side heat sink available, the primary system began to heat up.

Two minutes into the accident, the high pressure injection system came on for 2.5 minutes, which held the primary system temperature at a constant value. At the end of that period, high pressure injection was reduced, resulting in additional heatup of the primary system. At 5.3 minutes, the temperature in the hot leg reached saturation, and at 5.9 minutes two-phase flow was escaping through the stuck-open electromatic relief valve.

At about eight minutes the auxiliary feedwater block valves were opened by the operators, and feedwater was introduced on the secondary side of the steam generators, which decreased primary system coolant temperatures.

About 25 minutes from initiation, a reduction in heat transfer via the steam generators occurred as the primary side pressure approached that of the secondary side, which was being controlled by the turbine bypass system. This condition prevailed up to 73 minutes. During this period, coolant was continuously escaping from the primary system via the relief line, while pressure was holding fairly constant and void fraction kept increasing.

Due to these voids continuously forming in the primary system, the reactor coolant pump performance degraded causing a continuous decrease in loop recirculation mass flow rate. The presence of voids in the pump fluid causes the pumps to shake and vibrate, which may lead to seal and bearing damage. At 73 minutes, because of these concerns, the loop B reactor coolant pumps were turned off, causing flow to stagnate in that loop.

3.1.2 Chronological Narrative of Events. From the standpoint of primary system thermal-hydraulics, the first events of significance were the nearly simultaneous trips of the turbine and the main feedwater pumps. Both of these occurrences reduce the rate of heat removal from the primary system via the once-through steam generators; the turbine trip cuts off the steam flow from the steam generators to the turbine, and the trip of the feedwater pumps stops the feedwater flow to the steam generators. The imbalance between heat added to the primary system fluid by the core and that removed via the steam generators results in an increasing average temperature of the primary system fluid. This in turn causes an expansion of the primary fluid into the pressurizer, compression of the steam bubble and an increase in the system pressure.

The reactimeter sampling interval of three seconds is too crude to resolve the rapid progression of events during the first four seconds. Approximately three seconds into the accident, primary system pressure reached the 2255 psig setpoint of the electromatic relief valve located on top of the pressurizer. By this time, the turbine bypass valves and the main steam safety valves in the secondary system were opening, reestablishing steam flow at a higher pressure. The combined result was to reduce the rate of pressurization of the primary system. At 7.2 seconds, a reactor trip on high reactor coolant pressure (2344 psig) is indicated by the plant computer sequence of events review. The loop B narrow-range primary system pressure recorded on the control room strip chart indicates a peak pressure of 2365 psig.

During the period before the reactor trip, as discussed above, more heat was put into the primary system than what was removed, resulting in increasing pressurizer coolant level and simultaneous primary system pressure rise. A few seconds after the reactor trip, however, when the core was scrammed and thus power generation decreased to decay power, less heat was being generated than removed via inventory boil-off in the two steam generators. This causes a primary system coolant volume decrease resulting in a lowering of pressurizer coolant level and primary system pressure.

Twelve seconds into the accident, the primary system pressure had decreased to the point where the electromatic relief valve should have closed. However, subsequent behavior of the primary system establishes that the relief valve remained in the fully open position (see subsection 3.1.3.3). This resulted in a continuous loss of coolant from the primary system over the next 2.4 hours.

At approximately 30 seconds, the water levels on the secondary side of the steam generators had dropped to the point at which the Integrated Control System was programmed to introduce emergency feedwater and begin controlling water level at 30 inches in the startup level range via valves EF-V11A and EF-V11B. However, feedwater flow was not initiated because emergency feedwater system block valves EF-V12A and EF-V12B had inadvertently been left closed. This resulted in the continued depletion of water in the steam generators and eventual dryout and termination of heat removal from the primary system by the steam generators. After 76 seconds, the primary system cold leg temperatures had begun to increase, indicating that again heat removal via the steam generators was less than the heat being generated in the primary system. After 90 seconds, secondary system steam pressure in both steam generators had begun a steady decrease, indicating they were approaching liquid dryout.

Without the steam generators as an effective heat sink, the primary system temperature continued to increase. The thermal expansion of the system inventory due to this heatup, together with operator actions to provide increased amounts of makeup water to the primary system from the makeup tanks and to reduce letdown flow from the system, resulted in reversing the pressurizer water level decrease. The stuck-open relief valve, however, continued to discharge coolant from the primary system, thus causing it to depressurize. This continued depressurization, coincident with increasing pressurizer coolant level, is a strong indicator of the abnormal nature of the event.

Assuming that the coolant volume in the reactor vessel dome is at the hot leg temperature (605°F) before the transient, its saturation pressure is approximately 1600 psia. This pressure was reached about two minutes after the initiating event. If this volume stagnates due to flow restriction by the plenum cover which separates the plenum from the head, subsequent flashing might occur during the depressurization. However, when four reactor coolant pumps are operating, normal flow across the control rod guide tubes, through the upper head and into the outlet annulus is sufficient to replace one upper head volume in approximately seven seconds. Thus, fluid in the upper head is not stagnant, and flashing would not necessarily occur. In fact, during the first two minutes of the transient, which includes the depressurization to 1600 psia, no evidence of flashing is reflected in the rate of pressurizer water level change. That is, level change is explicable in terms of primary system temperature changes and makeup flow alone.

After about two minutes, the primary system pressure had decreased to the Emergency Safeguards Actuation setpoint, and the emergency high pressure injection system was automatically started. Full injection flow was maintained for approximately two and a half minutes. A calculation of pressurizer coolant level was performed for this period. The results confirmed the indicated pressurizer coolant level.

During this time period, from about two minutes to four and a half minutes, the primary system enthalpy decrease due to coolant loss through the relief valve and heat transfer via the steam generators essentially matched the primary system enthalpy increase due to core decay heat and reactor coolant pump operation heat inputs. The high pressure injection system was operating at this time adding coolant to the system. Thus, there was no significant net primary system heatup, and the cold and hot leg coolant temperatures were essentially equal, because the core decay heat input and the high pressure coolant injection occur between the locations of cold and hot leg coolant temperature measurements. Thus the bulk of the primary system was being maintained subcooled by additional high pressure coolant injection. At four minutes, 38 seconds, this injection rate was reduced, which was reflected by an increase in hot leg coolant temperature after five seconds, followed by a cold leg temperature increase after 11 seconds. The order in which hot and cold leg coolant temperatures changed, i.e., hot followed by cold, establishes that the reduction in high pressure injection flow rate was responsible for the change. Letdown flow was reestablished at about this time, probably at a maximum rate of 140 gpm. The makeup tank level began increasing at a constant rate of about 240 gpm at four minutes, 54 seconds, consistent with 140 gpm letdown flow and 100 gpm bypass flow from the throttled high pressure injection pump.

At five minutes, the pressurizer coolant level reached a high of 377 inches and began to decrease slightly. This decrease can be roughly accounted for by the difference between the decrease in makeup flow and the increase in letdown flow. However, at this time the primary system pressure was decreasing and the coolant temperature was nearing saturation. Void formation in the primary system could also have influenced the pressurizer coolant level to an undeterminable extent. At five minutes, 18 seconds the primary system pressure reached saturation for a large volume of coolant, and the pressurizer coolant level began to increase sharply due to coolant thermal expansion and void formation in many sections of the primary system.

From this time on, the hot leg coolant temperature in the primary system is at saturation, as determined by the saturation pressure. The reactimeter pressure reading is off scale at the low end after two minutes, and computer data are unavailable after 15 minutes. The primary system pressure recorded on reactor control room strip charts has been verified to remain at saturation. At about five minutes, 50 seconds, the pressurizer coolant level reached 400 inches, the measurement scale upper limit. The pressurizer coolant level measurement is confirmed by the reactor coolant drain tank pressure fluctuating widely at five minutes, 36 seconds, indicating that a two-phase mixture was going into the tank. The upper surface of the two-phase mixture in the pressurizer probably reached the top of the pressurizer vessel significantly before the coolant level instrument so indicated (see subsection 3.1.3.1).

The primary system continued to heat up slowly until emergency feedwater addition to the secondary side of the steam generators was successfully established at approximately eight minutes, 18 seconds. This feedwater addition was automatically attempted at 30 seconds but was prevented because block valves EF-V12A and EF-V12B had earlier been inadvertently left closed. This fact was not realized until about eight minutes into the transient, at which time these valves were opened. The coolant temperatures in the cold leg show the effect of heat removal in the steam generators starting approximately three seconds later. However, steam generator water levels did not recover immediately, because at first the feedwater boiled off as it entered the steam generators due to the elevated primary system temperatures and the heat stored in the steam generator hardware. During the following 15 minutes, the primary system temperature decreased. However, the system continued to operate at saturation temperature due to the simultaneous loss of coolant and resulting depressurization through the stuck-open relief valve.

After 25 minutes, the primary system pressure was approaching the pressure in the secondary system. The resulting reduction in heat transfer, due to almost equal pressures and temperatures on the primary and secondary sides of the steam generators, allowed steam generator water levels to recover to near their control setpoint of 30 inches in the startup range. For the remainder of this time period, primary system pressure was determined by the turbine bypass system, which was controlling secondary system pressure around a nominal setpoint of 1025 psig. Primary system pressure remained slightly above secondary system pressure, establishing the slight temperature difference required to transfer a fraction of the decay heat being generated by the core. During this time period, the primary mechanisms for removal of decay heat were the steam generator heat transfer and the boiloff of primary coolant and release of a two-phase mixture through the relief valve. The void fraction in the primary system was gradually increasing, as evidenced by the increasing signal from the source-range neutron detector and the continued decrease of primary system two-phase mass flow. The cause of this decreasing flow is the degradation of pump two-phase head as the fluid continues to form more voids and the density of the fluid decreases.

If auxiliary feedwater flow had been initiated at approximately 30 seconds, as designed for, the increase in primary system temperature during the period from 00:01 to 00:10 would not have occurred. However, safety injection would have occurred on low primary pressure, and the pressurizer would have approached a solid condition, although somewhat later in time. Assuming all other actions remained the same, the near equilibrium between primary and secondary system pressures and temperatures, with continued loss of primary fluid, would still have occurred. Under these conditions, the dryout and damage of the core would also have occurred, although a few minutes later than in the actual case.

3.1.3 Component Behavior.

3.1.3.1 Pressurizer. The pressurizer coolant level is determined by measuring the differential pressure between the head of the water-filled reference leg, which is sub-cooled at containment temperature, and the head of the steam-water mixture in the pressurizer. The indicated coolant level is thus a measure of the equivalent collapsed water level. It does not indicate the true level of the two-phase froth when flashing is occurring in the pressurizer, as was certainly the case. Calculation of pressurizer collapsed coolant level based on mass balance and fluid expansion can be used to confirm hypotheses of system behavior and to infer the formation of steam bubbles elsewhere in the primary system.

The pressurizer coolant level change has been calculated for the one to two minute period based on primary system makeup flow measured by changes in makeup tank level, expansion of the makeup water with heatup to system temperature, expansion of the primary coolant with heatup and depressurization, and blowdown through the stuck-open relief valve. The calculated change in pressurizer coolant inventory is essentially equal to the indicated pressurizer level change during this period, confirming that all significant effects have been accounted for, and that indicated pressurizer coolant level provided an accurate representation of the collapsed water level in the pressurizer.

Based on these calculations and analyses of available recorded information, it is believed that the differential pressure cell providing pressurizer level measurements was functioning correctly during the transient.

3.1.3.2 Reactor Coolant Pumps. During the early portion of the transient, the primary system coolant flow remained close to the nominal value, indicating that the four reactor coolant pumps were providing sufficient head to circulate the subcooled coolant through the system.

As the primary system depressurized, steam voids began to form in the coolant. The pump suction inlet coolant started to experience first cavitation bubbles and later two-phase inlet line flow. The existence of vapor bubbles in the coolant causes the pump head to degrade, which means that less head is available to circulate the coolant, thus causing the mass flow rate to decrease. Another factor contributing to the flow rate decrease is the increased resistance of the piping system to two-phase flow as compared to its resistance to single-phase coolant flow.

When the two-phase flow in the pump suction line between the steam generator outlet and the pump inlet enters the pump, the pump starts to vibrate and shake. This was indicated by increased pump seal leakage and measured oil level fluctuations in the oil sump of the pump high-pressure lubrication system.

Pump discharge pressure fluctuations and variations in pump motor current were also strong indications that the pump was cavitating and/or experiencing two-phase inlet line flow.

Severe pump vibration and shaking can also damage primary system piping and/or pipe supports, possibly resulting in a primary system leak. At 73 minutes, the loop B coolant pumps were shut off to avoid such possible damage.

3.1.3.3 Drain Tank Response and System Mass Discharge. As discussed in Appendix ERV, the drain tank serves to quench a limited quantity of steam discharged from the pressurizer during the primary system high pressure transients. The behavior of the tank was studied during the period from initiation (00:00) to the time of tripping the loop A pumps (1:40). The purpose was to estimate the mass discharged from the primary system and to attempt to provide correlations which would allow determination of the release rates of liquid and gas (steam plus noncondensibles) during later periods.

The study resulted in a reasonably good estimate of the mass discharged through the relief valve up to the time of the rupture disc blowout at approximately 00:15. A reasonable understanding of the early drain tank behavior was achieved. With the inclusion of pressurizer activity and primary system response up to 1:40; a reasonably accurate upper bound of the mass release out through the relief valve was calculated. Correlations for determining later activity were also developed. However, such correlations were found to be of limited value due to the inability to accurately estimate steam/noncondensible gas concentrations and temperatures.

The approach used to estimate the mass escape from the open relief valve is a combination of mass and energy balances written for the drain tank, and tables of critical flow based upon primary system pressure and pressurizer conditions. The approach is discussed below by time period.

Initiation to 206 seconds. During this period the alarm printer has two data points of the drain tank temperature. The approach is similar to that discussed in Appendix ERV except that the mass release through the open drain tank relief valve is accounted for. The major assumptions are:

1. The homogeneous fluid mixture and the initial nitrogen blanket are completely separated within the drain tank.

2. The unquenched steam volume is small relative to the initial nitrogen volume in the drain tank.

3. Discharge from the pressurizer to the drain tank is isenthalpic.

4. The drain tank cooler is on continuously, removing a constant 2183 Btu/second.

The first three assumptions would tend to underpredict incoming mass flow through the relief valve when mass flow is calculated from drain tank behavior since less mass and energy influx is required to affect changes in the tank. The drain tank cooler heat removal rate, if overestimated by the last assumption, would tend to overpredict calculated relief valve mass flow. However, based upon the observed drain tank behavior for the first 206 seconds, these assumptions do not adversely affect estimates of relief valve mass and energy discharge rates.

The mass and energy discharge versus time are calculated based on Moody critical mass flux data, adjusted first to the valve rated capacity and further adjusted by an energy balance similar to that presented in Appendix ERV.

The following conditions existed in the drain tank:

At t_1 = 86 sec: T_1 = 85.5°F
P_1 = 17 psig (31.7 psia)
h_1 = 54.15 Btu/lbm (subcooled)

At t_2 = 206 sec: T_2 = 127.7°F
P_2 = 121 psig (135.7 psia)
h_2 = 99.3 Btu/lbm (subcooled)

The adjusted energy balance includes the energy discharged from the tank due to liquid expansion as

$$\frac{\partial M}{\partial t} = - \frac{V}{v^2} \frac{\partial v}{\partial T} \frac{\partial T}{\partial t} \qquad (1)$$

where M = mass
t = time
V = volume
v = specific volume
T = temperature

The energy balance is then rewritten as:

$$\dot{M}_i = \frac{M_3(h_2 - h_1) + \dot{Q}_c \Delta t}{\Delta t \ (h_s - h_t)} \qquad (2)$$

where $\dot{M}_i$ = mass flow into drain tank
M_3 = mass in drain tank at 192 seconds when drain tank relief valve lifted.
$\dot{Q}_c$ = drain tank cooler capacity, 2183 Btu/second.
Δt = time interval.
h_s = enthalphy of steam.
h_t = average enthalpy of contents in drain tank.

This results in:

$$\dot{M}_i = 0.0417 \ M_3/\Delta t + 2.02 \qquad (3)$$

With the pressurization rate following the volume change of the initial nitrogen space, a combination of the gas state law (Pv^k = const.) and a drain tank volume of 920 ft^3, a relation for the volume of gas at 86 seconds, is written as follows:

$$V_{g1} = \frac{V \left[v_{\ell 1} - v_{\ell 3} \ (1 + 0.0417) \right] - 2.02 \ v_{\ell 3} v_{\ell 1} \Delta t}{v_{\ell 1} \ (P_1/P_3)^{1/k} - v_{\ell 3} \ (1 + 0.0417)} \qquad (4)$$

where V = drain tank volume
$v_{\ell 1}$ = specific volume of liquid at 86 seconds.

$v_{\ell 3}$ = specific volume of liquid at 192 seconds.
P_1 = drain tank pressure at 86 seconds.
P_3 = drain tank pressure at 192 seconds.

This relationship, when combined with equation (1), yields a gas volume at 86 seconds of 69 ft^3 and $\dot{M}_i$ of 20.4 lbm/sec (between 86 and 206 seconds).

Critical flow tables are used in later time periods and an orifice area for the ERV must therefore be calculated. First, the rated mass flow is correlated to the Moody critical mass flux

$$A_e = W_r/G_c$$

where A_e = equivalent orifice area, 5.9×10^{-3} ft^2

W_r = valve rated capacity at 2255 psia, 31.1 lbm/sec

G_c = Moody critical flow at set current pressure, lbm/sec-ft^2

Strict use of the Moody critical flow tables for 18 data points just prior to 86 seconds yielded a critical mass flow rate M_c of 21.8 lbm/sec, and thus an additional scale factor A_3 is applied to all following data, with

$$A_3 = A_e \ (\dot{M}_i/\dot{M}_c) = 5.9 \times 10^{-3} \ (20.4/21.8) = 5.5 \times 10^{-3} \ ft^2 \quad (5)$$

Using these relationships results in a total mass within the drain tank of 53,200 lbm and a total energy content of 5.63×10^6 Btu at 206 seconds.

206 to 351 seconds. For the period 206 to 351 seconds, the relief valve discharge is dominated by steam and high quality discharge as shown on Figure TH11. Because drain tank temperature is not provided beyond 206 seconds, an iterative procedure

to 351 seconds is used. The mass discharge from the tank is written in terms of tank energy change as per equation (1) as

$$\dot{M}_t = \dot{M}_i - \dot{M}_o = - \frac{V_3}{C_p v^2} \frac{\partial v}{\partial T} \frac{\partial h}{\partial t} = - \frac{1}{C_p v} \frac{\partial v}{\partial T} \frac{\partial E}{\partial t} \quad (7)$$

where V_3 = mass volume in drain tank at 192 seconds.
E = energy content in drain tank
C_p = specific heat

Using equation (7) the tank energy equation is rewritten as

$$\frac{\Delta E}{\Delta t} = (1 + \frac{h_t}{v} \frac{\partial v}{\partial t} \frac{1}{C_p}) = - \dot{Q}_c + \dot{M}_i (h_i - h_t) \quad (8)$$

State variables in the equations are calculated at average values over the time period.

Iteration between these equations is accomplished to:

(1) estimate h_t.

(2) find $\Delta E/\Delta t$ from equation (8).

(3) find ΔE and M_t from equations (7) and (8) and reestablish h_t as E/M_t.

(4) find the average tank temperature from h_t and use this to calculate v_t.

(5) With the data from steps (3) and (4) repeat the steps from (1) until h_t changes within less than 1%.

This energy balance is used for later calculations and assumes a high quality discharge until 351 seconds with critical flow based upon adjusted Moody choking. This assumption is believed to be valid because of its prior use for high quality discharge calculations, the adjustments to the equivalent area, and the moderate change of critical flow values with discharge quality beyond 50%. Through this period then, an average relief valve energy discharge rate of 23.9×10^3 Btu/sec (h_p = 1157 Btu/lbm, G_c = 3700 lbm/sec-ft^2) is used.

This process of iteration yields a total tank energy of 8.46×10^6 Btu, temperature near 180°F, and a tank mass inventory of 5.57×10^4 lbm at 351 seconds. The total discharge through the relief valve to 351 seconds is 7,600 lbm.

351 to 900 seconds. Beyond 351 seconds and up to the shutdown of the loop A pumps, drain tank pressure and relief valve discharge are controlled by a low quality discharge. At approximately 705 seconds (00:12) the drain tank looses its cooling/quenching capacity and approaches saturation temperature as the drain tank pressure decreases (Figure TH11). The low quality mixture entering the drain tank through the relief line cannot be relieved fast enough and drain tank temperature rises.

The energy coming into the drain tank via the open relief line can be estimated through equations (7) and (8) for the period between 351 and 705 seconds. The energy change rate is estimated as

$$\frac{\Delta E}{\Delta t} = \frac{E_{705} - E_{351}}{\Delta t} \qquad (9)$$

Using an iteration process similar to that discussed previously, the energy flow out through the relief valve is calculated to be $\dot{M}_i h_i$ = 46,400 Btu/sec resulting in an unrealistically high average mass flow rate of approximately 76.5 lbm/sec. Using 10 data points of mass flow rate history based upon a pressurizer discharge quality of less than 1% yields average mass flow rates of 52 lbm/sec using the Moody tables, 39 lbm/sec using the homogeneous equilibrium model (HEM) choked flow, and 63 lbm/sec using the Henry-Fauske model.

The predicted very high mass flow rate and its apparent closeness to the value calculated using the Henry-Fauske model are evidence of the breakdown of the assumed homogeneous conditions in the drain tank. Based upon pressurizer level, primary system pressure and loop A hot leg temperature, it should not be necessary to assume subcooled or saturated liquid discharge throughout the time period. Low quality fluid discharge rates should be comparable to the Moody or HEM model. With the lack of drain tank homogeneity, the liquid temperature stratifies and hotter liquid remains near a short-circuited path from the sparger to the drain tank relief valve. These phenomena are also observed to a great extent in the suppression pool of various boiling water reactor containment designs experiencing jet injection.

The effects of the short-circuiting and thermal stratification were estimated assuming 75% and 50% of the tank mass mixing with the incoming relief valve discharge. This resulted in relief valve discharge rates of 59 and 41 lbm/sec for 75% and 50% mixing, respectively.

The mass inflow through the period 351 to 705 seconds will therefore be estimated at an average of 47 lbm/sec resulting in a total relief valve discharge to 705 seconds of 24,000 lbm. Because pressurizer level and primary system pressure vary only slightly to 900 sec, the continuation of 47 lbm/sec is assumed yielding a total discharge through the relief valve of 33,200 lbm up to 900 seconds. It should be noted that these mass flow estimates are intended to be reasonable and are believed to be considerably on the high side.

900 to 6000 (loop A pump shutdown) seconds. Beyond the point of rupture disk breakage at 14 minutes, 48 seconds, the drain tank pressure is dominated by the release of the incoming two-phase mixture and flashing of a great deal of tank water with depres-

surization. Estimates of relief valve discharge beyond this point must thus rely on other means.

Until the point of the loop A pump shutdown, mass and energy balances of the pressurizer are used.

$$\text{Mass:} \quad \dot{M}_i - \dot{M}_e = \frac{A_p}{Vp} \frac{\partial L_p}{\partial t} - \frac{V_p}{2} \frac{\partial v}{\partial P} \frac{\partial P}{\partial t} \tag{11}$$

$$\text{Energy:} \quad \dot{Q}_{ph} + \dot{M}_i h_i - \dot{M}_e h_p = \frac{V_p}{v} \frac{\partial h}{\partial P} \frac{\partial P}{\partial t} + \frac{hA_p}{v} \frac{\partial P}{\partial t} - \frac{V_p h}{v^2} \frac{\partial v}{\partial P} \frac{\partial P}{\partial t} \tag{12}$$

where A_p = pressurizer cross-sectional area.
V_p = pressurizer liquid volume, A_pL_p
L_p = pressurizer liquid level
$\dot{Q}_{ph}$ = energy input from pressurizer heaters.
h_i = enthalpy of coolant entering pressurizer.
h_p = average enthalpy in pressurizer.

Over the period from 900 seconds and to the time of the shutdown of the loop A pumps (1:40), the pressurizer level varies only moderately and primary system pressure decreases from approximately 1175 to 960 psia. Throughout this period it is reasonable to assume an average pressurizer level and system pressure, such that dP/dt and dL_p/dt are approximately zero and equations (11) and (12) reduce to simply

$$\text{Mass:} \quad \dot{M}_i - \dot{M}_e \simeq 0 \tag{13}$$

$$\text{Energy:} \quad \dot{Q}_{ph} + \dot{M}_i h_i - \dot{M}_e h_p \simeq 0 \tag{14}$$

These are then combined and rewritten as

$$h_p \simeq \frac{\dot{Q}_{ph}}{\dot{M}_e} + h_i \tag{15}$$

where $\dot{M}_e = G_c A_3$ (16)

While this method will not address short term variations in the parameters, in the long term it should be a very good representation of the average pressurizer outlet conditions. This is because the integrated effect of inlet and outlet mass and energy flows must cancel, regardless of the microscopic transient behavior of the pressurizer.

To estimate $\dot{M}_e$ through this time period, a constant pressurizer heater energy of 1.05 MW is assumed and the input enthalpy varied for conditions between 0 and 50% void fraction. An average primary system pressure of 1000 psia is used in one case, and 1050 psia in another. The process of solution for $\dot{M}_e$ is to first choose the incoming hot leg quality and, therefore, h_i. Equation (15) is then solved through iteration using Newton's method as

$$f = h_p - h_i + \frac{\dot{Q}_{ph}}{\dot{M}_e} \qquad (17)$$

$$h_p(2) = h_p(1) - \frac{\dot{Q}_{ph}}{\dot{M}_e} \; \frac{1}{1 + \dfrac{\dot{Q}_{ph}}{\dot{M}_e^{\,2}} \; \dfrac{\partial \dot{M}_e}{\partial h_p}} \qquad (18)$$

For the first iteration, $h_p = h_i$ and $\dot{M}_e$ is found from tabled values of either Moody or HEM critical flow tables (taken from Reference 3). A new estimate for h_p is found from equation (18) and iteration continued until f is within 1 Btu/lbm. The results of this process are summarized below.

The following mass discharges through the open relief valve have been calculated:

(1) for the period from initiation to 900 seconds: 33,200 lbm

(2) for the period from 900 seconds to loop A pump trip:

Inlet Quality/ Void Fraction (%/%)	0/0	2/30	4.6/50
Using Moody Model (lbm)	210,600	202,000	190,000
Using HEM Model (lbm)	130,200	123,000	117,000

The sensitivity of total mass discharge to average primary system pressure was investigated by running the above methods with 1075 psia and 2% quality. This resulted in approximately 2%, or 4,000 lbm, additional mass for the Moody model and only 1%, or 1,000 lbm, for the HEM critical flow model.

Further study regarding the loss coefficient for the rupture disc yields

$$\Delta P = Cv\dot{M}^2 \qquad (19)$$

where
P = pressure, psid
C = coefficient
v = specific volume, ft^3/lbm
$\dot{M}$ = mass flow rate, lbm/sec

The usefulness of such an equation is in doubt, because differential pressure readings between tank and reactor building generally are very small.

3.1.3.4 Letdown Flow. The letdown flow system is part of the high pressure injection system and is discussed in Appendix HPI.

Prior to accident initiation, a normal letdown flow rate was maintained utilizing only one of the two letdown coolers. It is inferred that the second cooler was put into service at about five minutes from the start of the transient, by noting the similar behavior of the record outlet temperatures of both coolers after that time.

There were many abrupt changes in the rise and fall of the outlet temperatures as a function of time observed in the multipoint recorder data during the first two hours of the accident. The assumption was made that the letdown flow decreased to the block-orifice level at those times that the outlet temperatures suddenly changed from rising to falling. The flow rate during the periods of falling temperatures was taken equal to the normal value of 45 gpm.

At the times when the outlet temperatures reversed from decreasing to increasing, it was inferred that the maximum letdown flow was started by opening the block-orifice bypass valve. The nature of the temperature rise appears to be asymptotic to a steady-state value. The asymptotic outlet temperature recorded at each time period of rising temperatures was used to evaluate the flow rate during that period of time. A steady-state heat exchanger equation was used to model the letdown cooler.

A calculation of letdown flow was performed for the first two hours of the event. The results and details of that calculation are described in the following paragraphs.

The total integrated letdown flow from both coolers for the time period 00:00 to 2:04 was computed to be 150,500 lbm.

In making the calculation, the multipoint recorder data provided letdown cooler outlet and inlet temperatures (the latter assumed to be equal to the cold leg temperature). The cooling water conditions were assumed to be identical to the design specifications of the letdown cooler.

The heat exchanged in the letdown cooler was calculated from equation (1) for a counterflow heat exchanger.

$$q = \frac{UA\ [(T_{h_{in}} - T_{c_{out}}) - (T_{h_{out}} - T_{c_{in}})]}{\ln [(T_{h_{in}} - T_{c_{out}}) / (T_{h_{out}} - T_{c_{in}})]} \qquad (1)$$

where q = heat exchanged, Btu/hr
UA = conductance, Btu/hr-°F
$T_{h_{in}}$ = letdown cooler inlet temperature, °F (tube side)
$T_{h_{out}}$ = letdown cooler outlet temperature, °F (tube side)
$T_{c_{in}}$ = cooling water inlet temperature, °F (shell side)
$T_{c_{out}}$ = cooling water outlet temperature, °F (shell side)

A heat balance on the cooling water results in equation (2).

$$T_{c_{out}} = T_{c_{in}} + Q/\dot{M}_c\ C_{p_c} \qquad (2)$$

where $\dot{M}_c$ = cooling water mass-flow rate, lbm/hr
C_{p_c} = cooling water specific heat, Btu/lbm-°F

An iteration of $T_{c_{out}}$ was performed with equations (1) and (2), first assuming the design specification for $T_{c_{out}}$. Convergence was rapid (within a few iterations) and showed that q is predicted within a few percent of the converged answer by using the design specification for $T_{c_{out}}$.

The letdown mass flow rate is given by equation (3) resulting from a heat balance on the letdown flow.

$$\dot{M}_h = q/C_{p_h}\ (T_{h_{out}} - T_{h_{in}}) \qquad (3)$$

where $\dot{M}_h$ = letdown mass flow rate, lbm/hr

C_{p_h} = letdown water specific heat, Btu/lbm-°F

The time history of $\dot{M}_h$ for the time period under consideration is shown in Figure TH12. The integrated letdown flow from a single cooler was obtained by integration of the curve in Figure TH12. Based on the small variation between the recorded two letdown cooler outlet temperatures and the demonstrated insensitivity of $\dot{M}_h$ to $T_{c_{in}}$ (see next paragraph), it was assumed that each cooler passes equal amounts of letdown flow.

A sensitivity study was performed on all inlet and outlet temperatures. The results of this study show that the variation of letdown mass flow rate over a wide range of cooling water inlet and outlet conditions is between +20% and -35% of the rate at design conditions.

3.2 Phase 2 - Continued Depressurization

The second phase of the accident is defined as the time period from the shutdown of the two loop B reactor coolant pumps (1:13) to the closure of the relief block valve on the pressurizer (2:22). The length of this phase is one hour and nine minutes.

3.2.1 Summary. The shutdown of the loop B coolant pumps caused flow stoppage and phase separation in loop B. The loop A pumps continued to pump two-phase coolant with increasing void formation. Heat is being removed from the primary system via coolant evaporation and release through the stuck-open relief valve, and via the loop A steam generator. Due to continued indications of severe pump vibration and deteriorating mass flow caused by increasing void fraction in the primary system, the loop A coolant pumps were shut down. Shortly thereafter, the loop A hot and cold leg temperatures show a significant difference due to superheated steam entering the hot leg. The loop B steam generator was isolated on the secondary side at this time.

While auxiliary feedwater was injected into the loop A steam generator, the core was being cooled via boil-off of water and subsequent condensation in the loop A steam generator or release through the relief valve. However, due to continuous loss of coolant through the stuck-open relief valve and insufficient injection of additional coolant into the primary system, the system was boiling, partially uncovering the core and causing it to heat up.

3.2.2 Chronological Narrative of Events. Due to the almost continuous depressurization of the primary coolant system during the first hour of the transient, the thermal-hydraulic characteristics of the primary system went from normal subcooled to saturated two-phase flow. The performance of the reactor coolant pumps is highly sensitive to the condition of the coolant and deteriorates rapidly as the coolant void fraction increases.

As discussed earlier (see 3.1.3.2), the reactor coolant pump performance deteriorates continuously almost from the initiation of the transient, and at approximately one hour, 13 minutes into the accident, the pumps in loop B were turned off. Under normal reactor operating conditions this would result in a flow reversal in the shutdown loop due to continued operation of the pumps in the other loop. However, loop parameters indicate that flow stopped in loop B following the shutdown of the loop B pumps.

This stoppage of flow can be established from the reactimeter data on loop B temperatures and steam pressure. Immediately following the shutdown of the loop B pumps, the secondary side steam pressure in loop B started to drop rapidly. This indicated a substantial reduction in primary to secondary side heat transfer. This decreasing steam pressure would result in falling temperatures within the steam generator. If there were significant primary system flow in either direction, the primary loop temperature in that direction would indicate a substantial decrease. Instead, the loop primary system temperature continued to track primary system saturation temperature. From these observations it appears that phase separation occurred on the primary side of loop B following the shutdown of the loop B pumps. A condition resulted, where the difference in elevation head of liquid between the level in the loop B steam generator and that in the cold leg balanced the pressure drop across the reactor core. This resulted in the stoppage of flow through loop B.

The loop A primary pumps were left running at this time and continued to circulate a two-phase mixture of continuously increasing void fraction, while loop B remained stagnant. Primary system pressure and temperature variations during this period were mainly in response to feedwater and steam flow variations in the loop A steam generator. The primary coolant system is removing heat via two parallel pathways: 1) evaporation of primary coolant and release of a two-phase mixture through the stuck-open electromatic relief valve and the letdown system, and 2) heat transfer via the loop A steam generator. If an imbalance occurs, the primary coolant system will move toward an equilibrium condition, where the total heat removal via these two paths balances the primary system heat generation rate. Thus, if heat removal through the loop A steam generator decreases, primary system pressure will rise to increase the flow rate through the electromatic relief valve. On the other hand, if more heat is transferred via the loop A steam generator the primary coolant system pressure will decrease, and less two-phase flow will be expelled via the relief valve.

A sharp decrease in loop A secondary system pressure at approximately one hour, 30 minutes is attributed to the opening of the atmospheric dump valves MSV-3A. The purpose was to halt the pressure rise in the secondary system, and to avoid steam dump to the condenser. At this time, flow into the condenser was not desirable because of high condenser level and temperature. A switch to atmospheric dump at a controlled depressurization rate is indicated.

Approximately ten minutes prior to the shutdown of the loop A pumps, the loop A steam generator parameters indicate a sharp increase in secondary system steam flow, which last for approximately forty seconds. At this time, auxiliary feedwater flow was initiated in the loop B steam generator and continued for about two minutes, during which the loop A steam generator dried out. Feedwater flow was then switched to the loop A steam generator, and primary system temperature and pressure began dropping. Based on these records it appears that the auxiliary feedwater was initiated, by mistake, to the loop B steam generator rather than to the one in loop A.

The reduced primary system pressure in the loop A steam generator resulted in rapid boil-off of the feedwater as it was introduced into the steam generator, and thus steam generator level did not recover until the loop A pumps were tripped, reducing heat transfer in the loop A steam generator.

Throughout this period, the indicated primary system flow rate in loop A continued to deteriorate due to increasing void formation and subsequent degradation of pump performance. The flow rate started decreasing more rapidly during the period of approximately five minutes prior to the trip of the loop A pumps, when primary system temperature and pressure were decreasing rapidly. Approximately one hour, 40 minutes following the turbine trip, both reactor coolant pumps in loop A were shut down.

At this time, the indicated flow in loop A was approximately 40% of its initial value. This is to be compared with its value under normal, subcooled conditions when it would have been about

110% of its initial value due to the shutdown of the loop B pumps.

Less than 20 seconds following shutdown of the loop A pumps, the indicated primary coolant system flow went to zero. About two minutes later, a significant difference between the loop A hot and cold leg temperatures is evident in the reactimeter data. This is probably caused by a small amount of superheat due to heat transfer to the steam from the hot metal surfaces of the upper internals and hot leg piping. At about this same time, the loop B steam generator was isolated as indicated by a constant water level and increasing steam pressure. The resulting decrease in heat removal from the primary system was reflected in a temporary interruption of the decrease in primary system temperatures and pressure. About ten minutes following the trip of the loop A pumps, the temperature difference between the hot and cold legs in loop A began to increase rapidly, suggesting that by this time the upper part of the fuel had dried out, and the core was generating superheated steam.

During the period from several minutes before, until about 25 minutes after the trip of the loop A pumps, the steam generator in loop A was being filled to 50% of the operating range level with auxiliary feedwater to promote natural circulation. The auxiliary feedwater system injects ambient (70°F) water directly onto the upper portion of the steam generator tube bundle. Thus, steam entering the tube bundle on the primary side is rapidly condensed during times when auxiliary feedwater is injected. On the secondary side, part of the incoming feedwater is boiled off and the remainder is heated and collects at the bottom. Assuming the auxiliary feedwater flow rate is approximately constant during this period, the rate of change of steam generator water level is an indication of the amount of energy being removed from the primary system via secondary side boil-off.

About 15 minutes after the loop A pump trip (1:55), primary system pressure and cold leg temperatures began dropping more rapidly, while the rate of secondary system level rise in the loop A steam generator increased. The pressure drop indicates an increase in net heat removal from the primary system fluid, while the level increase indicates a decrease in the amount of heat being removed from the primary system fluid in the steam generator. This anomaly is due to a substantial decrease in heat transfer from the core to the primary system fluid, indicating that by this time a substantial fraction of the core had dried out and the excess energy generation was increasing fuel temperatures. At about this same time, loop B hot leg temperature began to decrease. This occurred because the depressurization resulting from flow out through the electromatic relief valve and the continuing condensation of primary system steam in the loop A steam generator was causing the colder primary system steam residing in the loop B steam generator to expand out into the loop B hot leg piping.

The temperature of the hot leg in loop A began climbing rapidly (8°F/minute) until approximately 24 minutes after the shutdown of the loop A pumps (2:04), when it stopped increasing and dropped about 9°F. This temporary reversal of the loop A temperature coincides with the termination of feedwater flow to the loop A steam generator. Thus, it appears that termination of feedwater flow interrupted condensation and flow of primary system steam in loop A, and since the piping at this point was cooler than the steam, the measured steam temperature decreased temporarily. About five minutes later (2:09), auxiliary feedwater flow was on again for approximately five minutes controlling secondary side water level in the loop A steam generator. At this time, the indicated hot leg temperature increased very rapidly (30°F/minute) and went off scale at 620°F. The initiation of auxiliary feedwater flow appears to have restarted steam flow in the loop A hot leg through condensation of primary system steam in the steam generator, drawing the now hotter steam from the reactor vessel into the loop A hot leg. Shortly after feedwater flow was stopped again, the loop A hot leg temperature came back on scale for a few minutes, then drifted back off scale.

The behavior of the primary coolant system following the trip of the loop A pumps indicates that steam occupied a substantial fraction of the system volume at that time. The behavior also illustrates that heat was effectively removed from the primary system by steam condensation on the primary side of the steam generators when the auxiliary feedwater was flowing. However, the continued loss of fluid from the primary system through the open electromatic relief valve resulted in a condition where the amount of coolant injected into the system was insufficient to maintain this mode of heat removal, and the core began to heat up because it was drying out. The primary system pressure responded as would be expected to the manipulations of feedwater during the period after the trip of the loop A pumps. System pressure decreased during the period when the loop A steam generator was being fed, from about 1:35 to 2:04, and remained constant for the invervening interval 2:04 to 2:09. However, when feedwater was injected again, from 2:09 to 2:14, the system pressure began to rise unexpectedly at about 2:10. Until this time, cold leg temperatures had been tracking saturation temperature, but in response to this pressure rise, they did not increase. It is thus concluded that there was an actual reduction of steam generation in the loops. There could be two reasons for this rising pressure: (1) increased steam generation in the core, and (2) the evolution of gaseous hydrogen from the zirconium-water reaction. The first reason seems unlikely, since there is no evidence of an increasing liquid coolant level in the downcomer, which would be necessary to increase core heat removal via increased steam generation. For the previous 40 minutes the primary system was depressurizing at rates varying with the effectiveness of secondary heat transfer, small amounts of coolant boiling within the core, and the two-phase discharge rate through the open relief valve. At approximately 2:10 the primary system pressure begins to rise. Within 4 minutes of the start of this pressure rise, still prior to relief block valve closure (occurring at approximately 2:22), reactor building particulate, gas and iodine channel radiation readings increase and eventually go off scale high. The pressure increase is believed to be hydrogen gas generation and expansion. This gas generation, which turns around the primary system pressure and affects reactor building radiation monitors, therefore indicates the start of the zirconium-water reaction. Large amounts of hydrogen and other noncondensible gases are released through this reaction. The failure of fuel cladding releases radioactive materials into the primary system and further via the pressurizer, the open relief valve, and the drain tank into the reactor containment building. This large initial release lasts until the relief block valve is closed at 2:22. The drain tank continues to vent steam. The releases to the reactor containment building are reinitiated later on in the transient with reopenings of the relief block valve.

3.2.3 Component Behavior

3.2.3.1 Primary System Liquid Inventory. The loss of fluid from the primary system was the key factor leading to core damage. Unfortunately, a detailed accounting of the primary system inventory is not possible given the limited data available on the operation of the primary coolant makeup system. However, the net fluid added or removed can be estimated by an analysis of primary system liquid inventory.

The primary system inventory analysis is keyed to the time interval between the turbine trip and the shutdown of the loop A primary coolant pumps. Two estimates of integrated mass flow out the relief valve based on different critical flow models were combined with an upper and lower bound estimate of remaining system liquid inventory for a total of four cases.

The estimates of mass flow out through the relief valve are based on the use of a critical flow homogeneous equilibrium model (HEM) for one analysis, and the Moody critical flow model for the other. These analyses are discussed in subsection 3.1.3.3. Presently available data on two-phase flow through relief valves are insufficient to make a definitive judgement regarding the relative validity of the two models. The actual mass flow rate probably falls somewhere between the two results, most likely closer to the results of the homogenous equilibrium model.

The behavior of the primary system following shutdown of the loop A primary coolant pumps provides strong indications of remaining primary system inventory. The excore source range neutron counters indicate that the downcomer level began dropping below the top of the core within a few minutes after pump shutdown. In addition, the hot leg temperatures began indicating superheated steam conditions approximately two minutes after pump shutdown, and temperatures began increasing rapidly approximately ten minutes after pump shutdown. From these indications, two scenarios were calculated in an attempt to bound the remaining primary system inventory.

A maximum inventory estimate is based on the liquid distribution shown in Figure TH13. The steam generators in both loop A and B are assumed filled with liquid to the level of the pump discharge nozzles, the maximum level attainable without spillover into the reactor vessel. The reactor vessel is assumed filled to the bottom of the hot legs, resulting in approximately 350 ft^3 of liquid above the top of the core. For the estimated decay heat level of 32 MWt, boiloff of 350 ft^3 of liquid would require approximately 6 minutes.

A minimum inventory estimate is based on the liquid distribution shown in Figure TH14. Loop B is assumed filled to the pump discharge nozzle because it was probably full above this level prior to pump shutdown (see system illustration in Section 2 for t = 1.5 hours). The level in loop A is calculated assuming the collapse of an 80% void fraction mixture as a lower bound. The level in the reactor vessel is assumed to be at the top of the core. In both the maximum and minimum mass calculations the pressurizer mass was based on the reactimeter pressurizer level at the time of loop A pump shutdown.

The loss of primary system fluid was determined by estimating the initial inventory, adding the liquid injected during the two and a half minutes of safety injection, and subtracting the estimated residual inventory. Both steam and water masses are included in the calculations. The results are summarized below:

Initial inventory	502,300 lbm
Safety injection	17,700 lbm
Maximum Inventory (100 min)	276,200 lbm
Minimum Inventory (100 min)	233,700 lbm

The changes in inventory are:

Minimum Loss	231,500 lbm
Maximum Loss	272,200 lbm

Combining these results with the two estimates of integrated mass flow out through the relief valve from subsection 3.1.3.3 results in the following estimates of net makeup (i.e., flow out the relief valve minus inventory loss):

<u>Case 1:</u> Moody flow model, minimum loss

$$M = 7500 \text{ lbm}, \ \dot{M}_{net} = +9 \text{ gpm}$$

Case 2: Moody flow model, maximum loss

$$M = -\ 33{,}200 \text{ lbm}, \quad \dot{M}_{net} = -41 \text{ gpm}$$

Case 3: Homogeneous equilibrium model, minimum loss

$$M = -\ 74{,}500 \text{ lbm}, \quad \dot{M}_{net} = -91 \text{ gpm}$$

Case 4: Homogeneous equilibrium model, maximum loss

$$M = -115{,}200 \text{ lbm}, \quad \dot{M}_{net} = -141 \text{ gpm}$$

The results indicate that uncertainties in the calculation of critical flow out the relief valve dominate the analysis of primary system inventory. Nonetheless, it appears likely that there was a net letdown of primary system fluid, probably in the range of 40 to 90 gpm average during the first hundred minutes. The letdown cooler outlet temperatures indicate that high letdown flows continue until approximately two hours after the turbine trip.

3.2.3.2. Core Flood Tanks. During this time period, the primary coolant system pressure was very close to the pressure required to initiate liquid injection from the core flood tanks. The reactimeter data on the loop A cold leg temperature indicate a minimum system pressure of about 605 psig. The pressure in the core flood tanks is controlled to 600 + 25 psig. The initiation of cold water injection into the reactor vessel downcomer would have condensed the steam residing there, further dropping system pressure and thus drawing in more water from the core flood tanks. Based on system behavior later in the accident, this depressurization would also have allowed pressurizer water to be drawn back into the hot leg, further increasing the liquid available to cool the core. As the added liquid increased core heat transfer, the primary system would repressurize to an extent dependent upon the temperatures in the core at the time, limiting the amount of injected liquid. Further analysis would be required to determine the effect of core flood tank actuation. Operator interviews performed after the accident indicate that the core flood tanks may have been isolated prior to this period. It is not clear whether this would have prevented the initiation of core flood injection during this phase of the accident.

3.3 Phase 3 - Initial Repressurization

This phase extends from the initial closure of the relief block valve at 2:22 to the start of sustained operation of two high pressure injection pumps at 4:27. The length of this phase is two hours and five minutes.

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

During the first half of Phase 3 portions of the core continued to dry out and heat up. The running of one reactor coolant pump, and the later manipulation of the high pressure injection system, served only to slow or delay core heatup. A substantial part of the actual core damage occurred during this period. Primary system behavior at 3:45 indicates an apparent collapse of core material as a result of damage which occurred prior to this time.

3.3.2 Chronological Narrative of Events. Just prior to closure of the relief block valve, the loop A hot leg temperature measurement went off scale (above 620°F) indicating that superheated steam had already formed in the top of the core and the reactor upper plenum. The four reactor coolant pumps had at this time (2:22) been shut down for approximately 42 minutes. The system saturation temperature was 486°F. At about 2:22 the relief block valve was closed in an attempt to regain reactor coolant system pressure. Shortly after the relief block valve had been shut, the hot leg fluid temperature in loop B went off scale (above 620°F). The temperature indicated in loop B was lagging that of loop A for two principal reasons: (1) the loop B steam generator was isolated, tending to stagnate flow; and (2) the pressurizer connection is to loop A, increasing communication with core outlet conditions in that loop.

During the time period from 2:22 to 2:54, the primary system pressure was rising continuously as a result of the relief block valve closure. Pressurizer level remained constant, probably because there was no liquid available with which to fill the pressurizer via condensation. Only superheated steam was present.

In response to increasing neutron levels on the source- and intermediate-range excore instrumentation and reactor coolant sample boron concentrations, emergency boration of the core was initiated. In retrospect, the activity increase observed in the excore instrumentation is known to be due to fluid density decreases in the downcomer and core regions. These density decreases were caused by the continuous boiling and steam generation in the core region, depleting vessel liquid inventory and causing a lowering of downcomer level and partial core uncovery. The speculation of core heatup prior to this point is substantiated by the radiation alarms in the control room at about 2:45, indicating fuel rod cladding failure and release of radioactive gases to the primary coolant system.

At approximately 2:35 the loop A cold leg temperature begins to drop, primary system pressure begins to rise more rapidly, and the excore source-range neutron detector indication begins to fall more rapidly. This response can be explained by the initiation of water injection into loop A by the start of makeup pump MU-P-1C. This cannot be verified by the alarm typewriter because alarm indications are not available at this time. The pump was apparently stopped sometime before the alarm information became available at 2:48.

Early in this interval, auxiliary feedwater was introduced into the loop B steam generator to attempt to promote heat removal from the primary system and induce natural primary coolant circulation. However, there was little heat transfer to either steam generator from 2:22 to 2:50, as indicated by the dropping pressures on the steam generator secondary sides at about 1:42. The loop B steam generator, which earlier had been isolated, was returned to service at 2:45 by opening the loop B turbine bypass valve. This action was probably taken in preparation for attempts to restart the reactor coolant pumps, since natural circulation had not been achieved.

During the next 10 minutes attempts were made to start each of the four reactor coolant pumps. At 2:55 reactor coolant pump 2B was successfully started. Some coolant flow was achieved in loop B for a short time, as indicated by the following changes:

(1) Loop B hot leg flow instrumentation indicated forward flow over a 6 to 9 second interval, beginning at 2:54.

(2) Excore nuclear instrumentation measurements decreased sharply, indicating the presence of a higher level of fluid in the reactor vessel downcomer and a higher density of fluid in the core regions.

(3) Cold leg temperatures in loops A and B increased substantially over a period of 15 to 18 seconds.

(4) Pressurizer level increased rapidly.

(5) Primary system pressure increased rapidly.

(6) The loop B steam generator level dropped, and its pressure rose from 140 to 700 psig.

An analysis of coolant flow in the system caused by the operation of coolant pump 2B can explain each of these indications.

The geometry of the reactor primary coolant system can be characterized as containing three different U-tube sections; one is the loop B lower steam steam generator tubes and pump suction piping, a corresponding one in loop A, and one in the vessel consisting of a level in the downcomer balanced by a two-phase mixture in the core. Upon operation of coolant pump 2B, water from the loop B U-tube will be passed to the downcomer causing: (1) increased pressure in the downcomer, (2) an increasing level in the downcomer, (3) a decreasing level on the primary side of the loop B steam generator, and (4) a decreasing level in coolant pump 1B suction piping. The level decrease on the primary side of the loop B steam generator will draw steam through the loop B hot leg to the tubes. Condensation of that steam in the tubes will enhance the flow through the loop B hot leg for a short time until sufficient concentration of non-condensibles is re-established to cease heat transfer. The increased downcomer pressure, coupled in loop B with the decreasing water level, will cause backflow of coolant in the 1B, 1A, and 2A loop cold legs. This backflowing coolant may be either steam or liquid, but in any case is warmer fluid since its source was the downcomer. This phenomenon explains the increased temperatures in loops 1A and 1B. The increased downcomer level and pressure will raise the level of the two-phase mixture in the core resulting in the liberation through boiling of heat stored in the core and a rapidly increasing system pressure. This will result in compression of the gas volume in the pressurizer, drawing fluid from the lower regions of the loop A hot leg into the pressurizer. The steam portion of that fluid will be condensed within the pressurizer liquid, which is now subcooled since the pressure has increased, and the condensate and any liquid entrained in the surge line will cause the pressurizer level to increase.

Since liquid flow was not established in loop B, pump 2B soon ran dry, and continued operation of the pump was ineffective. Intermediate range nuclear instrumentation confirms that the core was undergoing substantial heatup and voiding between 3:05 and 3:19 due to the lack of heat removal from the primary system. Pump 2B was stopped at 3:13. Removal of the dynamic head of this pump apparently permitted a small amount of flow to pass the temperature indicators in the other three loops in a forward direction, resulting in a temperature reduction at those points. Apparently the sole effect of operation of coolant pump 2B, after the initial 6 to 9 seconds of flow, was a minor pressurization of the downcomer, causing an offset in the U-tube in each of the cold legs.

At approximately 3:12, the relief block valve was opened as indicated by reactor building pressure and temperature. Following valve opening, system pressure decreased sharply, pressurizer level dropped and the cold leg temperatures in loops A and B were slightly depressed. The rate of primary system depressurization flattened out in the range of 1925 psig apparently due to relief block valve closure at 3:17. However, immediately thereafter, at approximately 3:20, makeup pump MU-P-1C was manually actuated, resulting in a continued decrease in reactor coolant pressure to a value of approximately 1450 psig. This depressurization was driven by condensation of steam due to the injection of makeup flow. Within one minute of startup of makeup pump MU-P-1C, cold leg temperatures began to fall. Loop B cold leg temperature decreased more rapidly, suggesting that some of the cold makeup water was sprayed into the pump suction piping. Makeup pump MU-P-1C was stopped at 3:37, creating a small system pressure rise, probably resulting from discontinuation of steam condensation and continued boiling of the makeup volume previously injected.

At approximately 3:40 the relief block valve was re-opened.

Excore source-range instrumentation indicated a sharp increase in count rate at 3:46, possibly due to a core configuration change. Simultaneously, cold leg temperatures increased in loop A significantly more than in loop B, and reactor coolant pressure increased from 1400 to 1675 psig. These observations are consistent with the belief that a reconfiguration of the core occurred, where hot material was transferred to the coolant available in the lower core regions.

At 4:08 reactor coolant pump 1A was started and shut down within one minute. This startup slightly decreased the loop A cold leg temperature, measured at the pump 1A suction side, which then resulted in a temperature increase as the hot downcomer fluid drained back into cold leg 1A. This directionality of mass movement is consistent with expected system response, and is similar to the behavior upon shutoff of reactor coolant pump 2B.

Makeup pump activity similar to that of the period from 3:20 to 3:37 occurred after 4:15, as is evident in the reactor coolant pressure response and cold leg temperature responses. Sustained high pressure injection via makeup pump MU-P-1C was begun at 4:27 and resulted in a positive rate of pressure change, which indicated that the liquid level was above the high pressure injection nozzles, thereby charging the reactor coolant system rather than condensing steam.

During this entire two hour period, cold leg temperatures in loops A and B trended together in response to a large number of high pressure injection system and reactor coolant pump status changes.

3.4 Phase 4 - Sustained High Pressure Injection

The fourth phase of the accident is defined as the time period from the start of sustained operation of two high pressure injection pumps at 4:27, and lasting to the start of primary system depressurization at 7:38. The length of this phase is three hours and eleven minutes.

3.4.1 Summary. The significant events during this period are the closure of the relief block valve at 5:18, with a subsequent primary system pressure rise, followed by cycling of the relief block valve to maintain pressure between 2000 and 2200 psig. Releases of liquid, gas, and liquid/gas mixtures from the primary system occur throughout the periods of relief block valve opening. Core decay heat is being removed by heatup of high pressure injection water flowing into the cold leg injection nozzles and out the top of the pressurizer.

The loop B secondary side is isolated during this period, and loop A secondary side steaming is switched from the condenser via the turbine bypass to the atmospheric dumping mode through the power-operated emergency main steam dump valve at 4:31 when condenser vacuum is lost. The loop A steam generator secondary side operating level is increased to nearly 100% through apparent feedwater flow through the feedwater startup control valves.

3.4.2 Chronological Narrative of Events. At the beginning of this phase a large gas volume consisting of superheated steam and noncondensible gases exists within the primary system. This mixture is distributed within the upper steam generator regions, the hot leg piping, and the vessel upper plenum. A summary chronology is shown in Figure TH15 for a portion of this period.

Core cooling is accomplished through water addition via two makeup pumps which force the coolant through the core. Primary system pressure is increasing slowly, although the relief block valve is still open. Loop B secondary system is isolated, and loop A secondary system is venting to the condenser.

At 4:31 condenser vacuum was lost, condenser dump valves closed and secondary side steam relief could be accomplished only by opening the emergency main steam atmospheric dump valve MS-V-3A. As a result, secondary side pressure in the loop A steam generator reversed its downward trend and increased slowly for approximately 12 minutes, after which it held constant. This constant pressure plus the stabilization of the coolant level in the loop A steam generator indicate the loss of vacuum (see Figure TH16).

At 4:42, delivery of emergency feedwater to the secondary side of the loop A steam generator is discontinued due to the trip of pump EF-P-2A, which has been operating since 3:35. As a result, heat transfer from the hot gases on the primary side to the steam on the secondary side decreased, causing a rapid increase in primary system pressure and also an increase in secondary side steam pressure. The concurrent slight increase in loop A steam generator operating level is believed to be due to the change of relative levels between primary side hot gas/liquid interface and secondary side level (See subsection 3.4.3.1). Steam generator A steam pressure continues to rise as primary system pressure becomes constant. This indicates a balance between heat sink capability of the secondary side, heat generation of the primary side, mass injection by the makeup pumps and mass ejection through the relief block valve. The rate increase and actual rise of secondary side water level in the loop A steam generator at 4:50 is believed to be the initiation of feedwater flow through the loop A feedwater startup control valve and subsequent level control.

Primary system pressure decreases as steam pressure in the loop A steam generator increases until approximately 5:18, when the relief block valve is closed. This is evidenced by a sharp drop and subsequent rise in loop A cold leg temperature. At this time, the high pressure injection pump pressurizes the inlet nozzle and delivers water to the primary side of the loop A steam generator. This results in a rise of the gas/liquid interface level until the primary side level is balanced, raising the primary system pressure and compressing the gas space, which results in an increased gas temperature. Cold leg temperatures oscillate slightly until approximately 5:30, due probably to redistribution of bottled-up gases during compression with the high pressure injection system. The cold leg temperature in loop A then increases as secondary side pressure decreases, indicating the rise of primary side hot gas/liquid interface level (with gas compression) because of inadequate heat transfer to the secondary side. Note that the pressurizer, through which the primary system flow was venting, is located on the loop A hot leg, and some redistribution of fluid/gas level is expected when the relief block valve is closed.

The cold leg temperature drop in loop B lags the response of the cold leg temperature in loop A by approximately 7 minutes, beginning at 5:25, indicating the above mentioned redistribution of primary side gas spaces. The fact that these cold leg perturbations in loop B are not observed in the steam generator secondary side pressure or water level indicates that no significant heat transfer to the secondary side takes place (see subsection 3.4.3.1).

At approximately 5:40, the primary system pressure rise is halted and maintained in a range between 1900 and 2150 psig by the first of a series of more than ten opening/closing cycles of the relief block valve (or possibly pressurizer vent valve). For the first half hour of this cycling, rapid changes are observed in the loop A cold leg temperatures corresponding to the frequency of relief block valve cycling. These variations occur in loop A due to the proximity of the loop A steam generator to the pressurizer surge line.

The secondary side water level in the loop A steam generator, which has been rising very slowly, following the decreasing secondary steam pressure and the rise of primary side gas/liquid interface, rises rapidly in two steps, a small step at 6:04, and a larger step at 6:08, followed by a half hour long continuous increase. These intentional increases of coolant flow through the feedwater startup control valves with a condensate and condensate booster pump pair in service (note that the integrated control system feedwater latch system must be bypassed to achieve such performance). The feeding by this process allows cooling and collapse of the upper downcomer annulus voids. This upper downcomer annulus space is approximately 4 feet above the 100% operating range level and is exposed to hot steam. Pressure response to cold water injection through normal feed is first observed as a slight pressure increase with operating water level increasing and leveling out as the fluid is boiled off. The continued increase allows exposure of the secondary side steam to the now colder annulus fluid and subsequent continued pressure drop which is aggravated by the rise of the primary side gas/liquid interface (see subsection 3.4.3.1). Loop A steam generator operating level continues to rise and approaches the primary side hot gas/liquid interface and boils off (6:30). As feeding continues, level rises to the point where the times of boil-off (relative change between primary and secondary sides) are controlled by the oscillation of primary side hot gas space. Loop A steam generator pressure continues to fall due to continued atmospheric release of any new steam generated.

At 7:08 the steam generator in loop A is fed for approximately 9 minutes with pump EF-P-2A. Steam pressure is low during this period and the water level is high, and only limited primary to secondary heat transfer takes place. However, as expected, the effect of this feed shows up in a slight pressure decrease, through steam quenching, and rapid level oscillations (see subsection 3.4.3.1).

During this period, the combination of sustained high pressure injection flow due to the cyclic opening/closing of the relief block valve and compression of the gas spaces with increasing primary system pressure appears to have provided core cooling largely missing from earlier periods. The variations of secondary side water levels and pressures with primary system pressure variations indicate that towards the end of this period the hot gas/liquid interface in the primary system is generally well above the water level in the loop B steam generator and rising just near the 100% operating level. The loop A steam generator is steaming and continues to do so.

At 7:13, the plant computer indicates a pressurizer temperature just above the pressurizer heaters of 342°F, well below saturation temperature at the current primary system pressures (greater than 2000 psig). These temperatures are believed to be representative of the reactor core outlet temperature at this time. The effective flow path is now from the high pressure injection nozzles through the core, the hot leg in loop A and via the pressurizer out through the open relief block valve, rather than via the steam generators. Thus, the hot leg temperatures above 620°F further indicate that the hot gases are trapped and compressed in the upper hot leg regions and prevent flow of primary coolant through the steam generators. The wide range temperature in the loop B hot leg indicates that after 2:30 temperatures vary with pressure in a direction which further confirms the continued presence of gas (see subsections 3.2.2 and 3.5.2 for further discussion of hot leg temperatures). The cold leg temperatures, measured in the loop seal region of the cold leg, indicate that the water is well subcooled and is slowly decreasing in both loops. This indicates that a portion of the cold injection fluid is being distributed to the lower portions of the cold legs, where it mixes with and slowly cools the existing fluid. There could be partial gas blockage in all cold legs at the slightly elevated pump discharge region.

3.4.3 Component Behavior.

3.4.3.1 Steam Generator Response with Primary Side Gas Blockage. When a large volume of noncondensible gas and steam have formed in the primary system and the reactor coolant pumps are shut off, the hot gas and steam will separate out and collect in the elevated sections of the system, such as the core upper plenum, the upper hot legs and the steam generators. Until sufficient pumping action is available to overcome the difference in elevation head of the liquid, this gas will remain. Such an accomplishment requires operation of a reactor coolant pump since high pressure injection or normal makeup is inadequate. While in the gas-blocked state, the contained gas and liquid are thermally stratified. This stratification is due to differences in temperature and component density and causes a variety of heat transfer mechanisms. Efficient heat removal capability in the steam generators is concentrated in a relatively shallow region in the vicinity of the primary and secondary side water levels. As these levels move up or down with gas contraction or expansion, heat transfer can be initiated or terminated very abruptly.

Until all gas is moved out of the upper hot leg region, primary system flow within the steam generator always remains close to zero, and whatever little flow that occurs is caused by changes in makeup pump flows and relief block valve position.

Four major primary-to-secondary heat transfer mechanisms of a large variety of mechanisms are illustrated for discussion in Figure TH17. These four mechanisms are chosen because they demonstrate the dominant heat transfer characteristics of the gas-blocked steam generators.

(1) Heat transfer is essentially blocked when the primary system is highly pressurized, compressing the primary gas space and drawing up subcooled to cold liquid from the lower steam generator region. Changes in secondary side level do not have a great effect on heat transfer capability because of the exposure of cooler liquid on either side of the tube wall. If the secondary side level is raised to the interface level, heat transfer may be reinitiated.

(2) When primary system pressures are such that hot gas space can expand sufficiently down into the tube region such that the level of liquid on either side of the tube wall are in proximity, mechanism 2 is observed. (This may also occur when primary side pressure is high but the gas space is still somewhat into the tube region.) The major heat transfer occurs between the hot primary side gas and the secondary steam resulting in superheat of the secondary steam. If the primary side interface level is thin and primary level high there is the potential for condensation of secondary steam when primary liquid temperatures are low. This mechanism in conjunction with mechanism 1 are exhibited in steam generator pressure and level response during the period between 7:00 and 10:00.

(3) When primary system pressure is low, the gas spaces expand and primary level drops, exposing hot primary gases and liquid to saturated liquid on the secondary side. This results in rapid secondary level boil-off, and secondary side pressure rises. This is augmented by additional gas-to-gas heat transfer. If the steam generator is fed during this period, heat transfer is continued until the gas spaces cool and contract. This

mechanism is exhibited in the loop A steam generator resonse between 4:30 and 5:30, and between 10:30 and 13:30 as well as during many other periods.

(4) The most efficient heat transfer is accomplished by emergency feedwater which is injected right into the upper tube region, cooling and contracting the primary gas space within the steam generator. This feedwater also helps to raise the secondary level into proximity of the hot primary interface level to induce mechanism 3. The behavior of the loop A steam generator between 2:00 and 4:30 exhibits this heat transfer mechanism.

The specific events which relate primary system pressure and level to secondary side behavior at specific times are discussed in subsections 3.4.2, 3.5.2, and 3.5.3. Each relies on the ideas presented here regarding heat transfer capability versus differential primary to secondary level. It should be remembered that as long as the gas blocks exhist, steam generator response exhibits local heat transfer effects only and does not imply heat removal from the reactor core region.

3.5 Phase 5 - Extended Depressurization

The fifth phase of the accident is defined as the time period from when the relief block valve is opened at 7:38 until the relief block valve is closed again at 11:08. The length of this phase is three hours and thirty minutes.

3.5.1 Summary. This phase is initiated when the relief block valve is opened at 7:38 and is left open with the intention to depressurize the primary system to allow core flood injection. During most of this period, reactor coolant system pressures are below 700 psig. Indications of core flood dump are indeed seen, but the amount of fluid injected is small relative to concurrent reactor coolant makeup pump flows. The primary system is essentially prevented from further depressurization due to the existing large volume of noncondensible gas, the continuing gas generation within the reactor core, and the saturated condition of the liquid. Without continuous makeup or high pressure injection with flow through the reactor core, via the loop A hot leg, the pressurizer, and out through the relief block valve, a repeat of the 2:30 core dryout becomes a distinct possibility. At 10:00, reactor coolant hot leg temperature in loop A began to decrease below 620°F for the first time in nearly eight hours.

At 9:49, a 28 psi overpressure occurs in the reactor building indicating a rapid burn of hydrogen discharged from the primary system. The loop B secondary side is isolated, and the loop A secondary side is depressurizing, with no condenser vacuum. Neither loop is acting as a significant heat sink during this period. Sometime prior to 10:20 (operator interviews indicate about 8:30) the atmospheric dump valves were closed, thus removing the capability of a loop A secondary heat sink until restoration of condenser vacuum and condenser dump are initiated at 13:30. Cooling of the core was maintained through injection of coolant, which flowed from the core via the pressurizer and out through the relief block valve. This phase ends at 11:08 with the closing of the relief block valve.

3.5.2 Chronological Narrative of Events. During approximately two hours prior to the start of depressurization, the reactor coolant system was held at a high pressure within 1900 and 2150 psig, controlled through the opening and closing of the relief block valve. A varying mixture of two-phase two-component water/steam/noncondensible gas was intermittently discharged during the periods when the valve was open. Primary system inventory was maintained during this period by continuous makeup injection via pumps MU-P-1B and MU-P-1C. Prior to the start of depressurization the pressurizer liquid was highly subcooled, indicating that adequate, subcooled liquid flow into the core and out via the pressurizer and relief block valve had been maintained through the intermittent cycling of the valve. At 7:13 the utility printer indicated a pressurizer temperature of 340°F with primary system pressure above 2000 psig. Neither steam generator was absorbing any heat from the primary system.

When the relief block valve was opened at 7:38 and the pressurizer heaters turned off at 7:40, the primary system pressure began a continuous, rapid decrease over the next hour. Close to 7:50 into the accident, as the pressure drops close to the previous low of 1200 psig, the gas spaces again expand to near the size they had at 5:30. This expanded gas must then begin to be ejected through the surge line and out via the pressurizer, as the reactor coolant system pressure continues to fall. Since the pressurizer surge line is directly coupled with the hot leg in loop A, the release of gas from this loop occurs when the gas expands down the hot leg from the steam generator towards the vessel. The gas in the loop B hot leg, however, has to expand downward through the loop B hot leg piping and then migrate through the reactor vessel upper plenum and into the loop A hot leg and the pressurizer surge line. This process of expansion and ejection continues throughout the depressurization phase. The gas ejection appears in the decrease to on-scale readings (at or below 400 inches) and oscillations of pressurizer level near 7:50. The coolant in the pressurizer remains well subcooled until nearly 10:30.

Pressurizer temperatures indicate the ejection of mainly noncondensibles rather than a mixture with steam and thus provide additional indication that the major gas space components are noncondensibles. Pressurizer temperature changes from 342°F to 351°F, between 7:13 and 8:30, indicate only a small pressurizer heatup through gas bubble to liquid heat transfer. If the gas were composed of significant quantities of steam, this temperature increase would be expected to be larger due to the lengthy migration paths (except in the A loop). Pressurizer temperatures thus provide information on heating of the makeup fluid from core decay heat and the presence of noncondensibles.

Near 8:31, the primary system pressure reaches the core flood setpoint of 600 psig. It is expected that coolant from the core flood tanks could have been injected into the primary system. However, this coolant addition of approximately 100 ft^3 over the period from 8:31 to 9:10 would be very small compared to makeup flow.

As determined through reactor building pressure, the relief block valve was closed at approximately 9:10, halting the depressurization of the reactor primary coolant at approximately 435 psig. This is nearly concurrent with the shutdown of makeup pump MU-P-1C, leaving only makeup pump MU-P-1B in service. Based on the relief valve discharge temperatures, per the alarm printer, it is possible that the relief block valve was opened between 9:20 and 9:30. The lack of significant reactor coolant system response and reactor building pressure response, however, discount its importance in subsequent events.

The response of the loop A cold leg to the depressurization is shown by a sharp decrease in temperature near 8:45. Depressurization causes the trapped gas in the upper sections of the steam generator to expand, forcing the primary liquid in the steam generator tubes to move down towards and into the cold leg. The liquid in the hot leg is forced downwards towards the pressurizer surge line connection. Over the next one and a half hours, the loop A cold leg response varies greatly with changes as described in Figure TH18.

The loop B cold leg, isolated by the continuous injection via MU-P-1B, or at times MU-P-1A, does not show a similar response. As discussed later, the change in the rate of temperature decrease in the loop B cold leg may be an indication of lack of usual high pressure injection pump reconfiguration close to 9:55.

At 9:49, the hydrogen created by the zirconium/water reaction in the core, together with the oxygen in the air in the reactor building, reached a combustible concentration, which resulted in a 28 psig pressure spike measured in the reactor building. The time when this overpressure occurred can also be observed in nearly all plant pressure readings which measure a differential pressure relative to reactor building pressure. The time when this burn occurred indicates that hydrogen was released from the expanded gas regions in the core upper plenum and upper hot legs in the course of primary system depressurization after 7:50. The high building pressure resulting from the hydrogen burn caused the Engineered Safety Feature System to activate the high pressure injection, the reactor building isolation spray, and the decay heat removal system which currently is not configured for participation above 400 psig.

The bypassing of the Engineered Safety Features System by the operator resulted in only approximately 30 seconds of high pressure injection. Makeup pump MU-P-1C stopped 15 seconds later, at 9:50, presumably leaving makeup pump MU-P-1B in service.

The relief block valve was closed after the hydrogen burn, and primary system pressure rose. Again, at 10:00 the relief block valve was reopened, apparently to return the primary system to core flood capability. The resulting depressurization continued to 10:30, when a contraction of high temperature gas, or a partial steam quench due to makeup pump MU-P-1C being put into service at 10:31, brought the primary system pressure down to 408 psig. This additional depressurization below the previous level of approximately 435 psig would have allowed an additional small amount of core flood, approximately 22 ft^3, to be injected over a period of approximately two minutes around 10:35. This combined mass injection was sufficient to bring primary system pressure up to 425 psig, with MU-P-1C tripped at 10:36. Until the closure of the relief block valve at 11:08, primary system pressure was maintained near 420 psig, with makeup pump MU-P-1B which was assumed to be in service. Further primary system depressurization was not effective.

3.5.3 Component Behavior.

3.5.3.1 Primary Coolant System. The substantial changes in pressurizer level and hot and cold leg coolant temperatures during this period are due to four physical conditions. All of these result from the gas/liquid configurations established during the first three hours of the accident and from a continuation of the generally inadequate cooling. Sufficient cooling was provided only during in the high pressure injection mode with the relief block valve open. The four physical conditions are: (1) the large existing void fraction in the reactor coolant system, (2) the continuous hydrogen generation in the reactor core, (3) the large amount of remaining liquid in the pressurizer and reactor vessel, at or near saturated conditions, and (4) the high degree of thermal stratification in the gas-blocked steam generators.

The reasons for the primary system pressure not dropping below 400 psig with the relief block valve open are: (1) the steam/hydrogen generation rate in the core, and (2) the large amounts of system liquid at or near the saturated state in the pressurizer and core. The only currently available effective heat removal mechanism is provided by the makeup pumps, in normal or high pressure injection mode, and the flow through the core, the pressurizer, and the open relief block valve. The steam generators are gas-blocked on their primary side. The depressurization and simultaneous boil-off heat removal is limited by the volumetric flow rate of gases and/or two-phase mixtures through the open valve.

From about 10:00 the pressurizer begins to dominate the events until the repressurization phase which begins at 13:30. From before 10:00 until 10:28 the pressurizer has been subcooled and heating up at approximately 3°F/min. This heatup rate is due to the pressurizer heaters and the continued flow of hot liquid, steam, and gas through the core and the pressurizer out through the open relief block valve. This temperature increase ends when the pressurizer reaches saturation temperature at approximately 10:28, from which time it remains at or above primary system saturation pressure until 11:25. While the pressurizer temperature is at saturation there is a tremendous potential for additional steam formation in the pressurizer whenever the system pressure decreases. Aided by the elevation head of the pressurizer above the rest of the primary system, this steam formation in the pressurizer will serve to force fluid from the lower pressurizer region into the hot leg, because discharge is limited by the critical flow conditions in the relief block valve. The relief block valve is open between 10:00 and 11:08. When primary system pressure drops slightly below saturation, steam generation takes place in the pressurizer which, together with gas and/or steam entering the pressurizer in the surge line, cannot be discharged fast enough through the open relief block valve. This combination of events prevents further substantial system depressurization. A balanced situation occurs, where discharge flow determines the rate of depressurization. The short periods of level increases in the pressurizer are due to the entrance of slightly colder fluid, which inhibits local surface boiling and/or affects the temperature compensation of the pressurizer level instrument. These periods are short, and the return to the general trend of boil-off is rapid. When the relief block valve is closed, the boil-off capability still remains. With the primary system at a pressure less than saturation pressure, liquid is boiled off and the generated steam flows into the pressurizer forcing fluid out of the pressurizer back into the hot leg, until the energy content in the form of voids balances the system pressure at saturation conditions. For example, the primary system pressure is held up due to the fact that the pressurizer is at saturated conditions. Reactor coolant system pressure in turn will be balanced by the amount of gas space contraction with the incoming pressurizer liquid volume.

The pressurizer spray line valve is open between 10:00 and 11:15 and is believed to be active. However, if there is gas in the loop A cold leg, the spray line will only facilitate the transfer of this gas to the top of the pressurizer. If there is liquid in the cold leg, the small pressure difference between the cold leg and the hot leg due to flow will allow liquid level in the spray line to rise only until it reaches equilibrium with the pressurizer coolant level. Its impact on the reactor coolant system response is believed to be small, and it is not fundamental to the understanding of system behavior during this period. Indeed, when the spray line valve is closed, there is no perceptible impact on pressurizer coolant level. However, during the period when the spray line valve is open and the pressurizer is controlling the reactor coolant system pressure, gas is forced back into the cold leg to allow maintenance or renewal of a gas pocket in the loop A cold leg.

These four conditions prevent further depressurization of the primary system via the relief block valve. Due to this pressure holdup, the continuous injection of core flood liquid for cooling and high pressure injection, with flow through the core and out the open relief block valve, is very limited.

Rapid variations in primary system hot leg temperatures for both loops A and B were observed for this and the following phase. These variations can be grouped in two categories: first, the difference in response between loops A and B, and second, the response of the thermally stratified gas in each upper hot leg and upper portion of the steam generator tube bundle.

The first category was reviewed previously in discussing the loop A cold leg temperature response in subsection 3.4.2. The difference in response occurs primarily because the pressurizer is connected to loop A rather than to loop B. The net result is a faster response of liquid levels in the loop A hot leg to reactor coolant system pressure changes when the relief block valve is open. The coupling of the pressurizer to loop A is also believed to allow the gas space in this loop to be smaller than that in loop B throughout nearly the entire course of the accident. The earlier hot leg temperature drop in loop A at 10:30, and maintenance below 620°F after 10:40 while the loop B hot leg coolant temperature remains above 620°F, is due primarily to gas maintained in the loop A hot leg and steam generator due to prior steaming.

The second category, the response of the trapped gas volumes in the hot legs, is a result of: (1) temperature change with pressure change; and (2) movement of gas regions at different temperatures will be observed.

These mechanisms are coupled with one another, sometimes countering and at other times amplifying each other. For example, during the 7:40 depressurization the hot leg coolant temperature decreases, but hotter trapped gas is brought down as the gas expands, thus limiting the measured temperature decrease. The increases in loop B hot leg wide range temperatures between 9:00 and 9:50 are due, first to a readjustment of level with relief block valve closure, and second to the makeup pump trip, followed by a migration and rise in the loop B hot leg of increasingly hotter gases generated in the inadequately cooled core. Similarly, between 9:50 and 10:00, the combination of emergency injection, with system pressurization followed by relief block valve cycling, serves to readjust level and push new, colder gas into the proximity of the loop B hot leg resistance temperature. Various combinations of these mechanisms also correlate with loops A and B hot leg temperature decreases at 10:35. Loop A hot leg temperature behavior from 10:40 on is also coupled to the limited heat transfer capability to the secondary side.

Such behavior generally serves to underscore the high degree of thermal stratification in the gas-blocked loops and the continuing hot gas generation capability of the reactor core. However, the process most important to core integrity is still the cooling capability provided by makeup and/or high pressure injection with flow through the core and out via the open relief block valve. This cooling capability is insufficient, which is evidenced by the increase in loop B hot leg temperatures between 9:00 and 9:50, when a general temperature decrease is expected.

3.5.3.2 Core Flood Behavior. Core flood tanks A and B both appear to have been actuated from an initial charge pressure of over 600 psig and equilibrated after the injection at 410 psig. The probable sequence of operation is as follows:

(1) Reactor coolant system pressure reached 600 psig at about 8:30 and continued to decrease at a uniform rate to about 435 psig at 9:10. Approximately 100 ft^3 of water should have been injected based on expansion of the nitrogen charge over a forty minute interval.

(2) Reactor coolant system pressure increased after 9:10, remaining above 435 psig but below 525 psig until about 10:34, at which time it continued down to 410 psig at 10:35, after which it began to rise. It never again went below 410 psig. Approximately 21 ft^3 of water should have been injected during this period.

(3) Alarm printer indications of loop A level fluctuations above and below the high level alarm point for the period between 9:56 and 11:07 seem to be spurious since the level should have gone down and re-charge was unlikely to have been achieved.

Figure TH19 illustrates the assumed core flood charge versus pressure characteristics based on volume versus pressure change for the nitrogen gas volume.

The following conclusions are drawn:

(1) Core flood tank A probably never opened during the period 8:31 through 12:20, when loop A pressure was at or below 600 psig. This assumes that the alarm instrument is good.

(2) No computer logs of core flood tank B exist and its behavior remains unknown. It could have operated as designed, probably dumping coolant into the system from about 8:31 to 9:45, and it could possibly have opened again briefly between 10:33 and 10:35. This assumes that no recharge of core flood tank B was performed in the interim.

3.5.3.3 Loop A Steam Generator Behavior Relative to Atmospheric vs Condenser Dump. During the accident, the steam from the secondary side of loop A, when venting, is vented to the condenser or to the atmosphere via the atmospheric dump valves. The switching off of the condenser circulating water pumps after approximately one hour into the accident allows manual atmospheric dump valve capability, while return of two of the four circulating water pumps at 2:55 does not necessarily prevent manual control. Condenser dump is not possible when condenser vacuum is lost, which occurs at approximately 4:35 and lasts until 13:13.

The response of the loop A steam generator is in part a function of the capability to release generated steam when the level of the hot liquid/gas interface on the loop A primary side allows significant heat transfer. Changes in reactor coolant system pressure and loop A steam generator operating level and pressure agrees well with the description of heat transfer versus relative primary to secondary level, as described in subsection 3.4.3.1. The purpose of this section is to discuss the differences between the two noteworthy periods of loop A steam generator steaming from 4:30 to 7:30, and from 10:25 to 15:30.

(1) Comparing the rates of pressure rise when the relative primary to secondary level is configured for steam generation, the rates of pressure rise at 10:25 and 10:40 are between two and five times larger than those between 4:30 and 5:30. The actual temperature difference driving the steaming is unknown and currently cannot be well established because of the very high degree of thermal stratification on the primary side. However, it is thought that the driving potential should not be much different between the two periods because of the stagnation of the primary side tube region. The difference in later to earlier rate of pressure increase then implies no steam dumping in the later period.

(2) The rate of pressure decrease beginning at 11:30 is only half of that during the period between 5:30 and 6:10, again indicating that there is no steam dumping in the later period.

(3) During a region of known limited heat transfer, the period where loop A steam generator pressure levels off in the early period (6:30) is matched in the later period where pressure increased (13:30).

(4) The rate of pressure decease between 6:10 and 6:20 is reasonably matched by the rate between 13:40 and 13:50.

The above indicates that the steam generator was not dumping steam during the pressure rises after 10:20 and until 13:40. Return of condenser vacuum at 13:13 would allow pumping to the condenser and this would have been desirable. Since a loop A steam generator pressure decrease occurs at 13:15, this could also be an indication of initiation of dump. However, the lack of a significant operating level change at this time indicates otherwise. The combination of pressure decrease with level decrease occurring at 13:45, with the resumption of startup feed-water near 13:55, appears more likely.

3.6 Phase 6 - Repressurization and Recovery

This phase of the accident is defined to start with the closure of the relief block valve at 11:08, and it ends at 15:50 with the successful start of one of the reactor coolant pumps. The length of this phase is four hours and forty-two minutes.

3.6.1 Summary. This phase begins at 11:08 with the closure of the pressurizer relief block valve. The apparent intention of this action was to repressurize the system in an attempt to collapse the steam and initiate natural circulation. When, after about 4.5 hours, these attempts had not succeeded, one reactor coolant pump was operated briefly at 15:32 and then restarted at 15:50, after which it ran continuously. The continued operation of one reactor coolant pump finally established cooling via forced circulation in the reactor coolant system and heat removal via the loop A steam generator. This marks the end of the transient.

As with the depressurization phase, prior to operation of the coolant pump, this phase can be characterized as a noncirculating system with very limited heat transfer through the steam generators. Heat removal from the system occurred almost solely through the relief block valve when it was open. A second major influence on system pressure was the operation of the high pressure injection system, which caused both absorbtion of heat into the cold injection water and compression of the steam/gas space in the system.

This phase includes a long period, from 11:36 to 13:23, where there was apparently only minimal makeup flow, no heat transfer via the steam generators, and only occasional flow through the relief valve. During this period there were some indications that the system water level was reduced below the level of the cold leg nozzles. It appears that the core region was slowly boiling, and it is possible that the water level in the downcomer dropped and also that there was some uncovery of the upper region of the core.

Since there was little flow in the system during most of this phase, the individual system components are not tightly coupled. The bulk of the discussion, therefore, is centered around the behavior of the individual components.

The chronological narrative presents a background for discussion of the behavior of the individual components, and the indications of possible core uncovery in the period before 13:23.

3.6.2 Chronological Narrative of Events. The phase was initiated by closure of the relief block valve at 11:08. About 30 minutes prior to this closure, the system pressure had reached a minimum of 410 psig and had leveled out at about 420 psig. The pressurizer temperature indicates that the pressurizer liquid had returned to saturation at that time after being subcooled for more than five hours. It is most likely that this return to saturation was the principal factor in the arrest of the system pressure decrease.

Immediately after closure of the relief block valve, the pressurizer level decreased from 400 to 175 inches over a period of about ten minutes. This draining of the pressurizer transferred about 700 ft^3 of saturated liquid to the reactor vessel via the loop A hot leg. There are several factors which may have influenced this pressurizer drain. With the relief block valve closed, the boiling of the saturated pressurizer liquid due to the operation of pressurizer heaters would allow separation of the steam into the pressurizer steam space, pushing liquid out through the surge line. The high pressure injection system was operated between 11:19 and 11:38. This probably caused a minor system pressure reduction which drew the last ten to twenty inches of water from the pressurizer after the spray line valve was closed. A short period of high pressure injection occurred between 11:33 and 11:36, after which there was no sustained system operation, except for minor makeup flow, until 13:23.

During the 90 minutes following relief block valve closure, the system pressure slowly rose from 420 to 620 psig, and the pressurizer slowly refilled. As with the drain, this refill removed about 700 ft^3 of liquid from the rest of the primary system. The pressurizer refill was driven by the system pressure increase, which compressed and condensed the contents of the steam space, and was possibly aided by slow leakage of the contents of the pressurizer steam space out through the safety valves. During the refill period, the pressurizer liquid is slightly subcooled, confirming that condensation of steam from both the steam space and from inflow through the surge line could occur. Some liquid was carried into the surge line during the refill period. The level of the two-phase mixture was probably at or above the surge line during the pressurizer refill.

Shortly after the completion of pressurizer refill, the relief block valve was opened for two short periods, from 12:36 to 12:47 and from 12:52 to 13:00. In each case, the system pressure dropped while the valve was open. The second operation of the valve caused the pressurizer level to drop about 40 inches, indicating that liquid was being expelled and steam and/or hydrogen was entering the pressurizer from the hot leg.

Immediately after the pressurizer drain, which started at 11:08, the loop A hot leg temperature stayed constant, indicating no fluid motion in that region. At 12:10, the loop A cold leg temperature became constant at about the system saturation temperature, indicating that steam was in communication with the cold legs at the elevation of the nozzles. The pressurizer requires 200 ft^3 of liquid to refill after 12:10, so it is reasonable to presume that the downcomer water level dropped several feet below the cold leg nozzles as the pressurizer finished refilling. From 12:30 until initiation of high pressure injection at 13:23, the water level in the reactor vessel was probably being reduced slowly via boil-off, and uncovery of the upper core regions may have occurred for a second time.

High pressure injection flow is initiated at 13:23 and continues uninterrupted until 14:43. Upon initiation, the system pressure drops and the pressurizer level is rapidly reduced to 275 inches, dumping about 350 ft^3 of liquid into the reactor vessel. After about 8 minutes, the system pressure begins to rise significantly. At 13:50, the pressurizer level begins to rise smoothly and the pressurizer is refilled by 14:22. Pressure continues to rise more steeply after the pressurizer is refilled, and reaches almost 2300 psig at 14:43 when the high pressure injection is shut off. Clearly, after the first 8 minutes, the high pressure injection system is charging the primary system with water in an expected level control mode. However, it is also clear that about 850 ft^3 of liquid, 350 from the pressurizer and 500 from 8 minutes of high pressure injection system operation, were needed to fill the system to the level at the top of the cold leg nozzles. Before that much fluid had been injected, the injection flow was condensing steam, causing a pressure decrease. These observations support the previous speculation concerning core uncovery.

The sustained injection left the system substantially filled, with the remaining pockets of noncondensible gas compressed, but there was still no evidence of effective cooling through natural circulation in either loop. In particular, the loop A cold leg temperature increased above temperatures corresponding to either the steam pressure or temperature in the loop A steam generator. Natural circulation was still blocked, and the core was merely heating the cooler liquid which had been injected.

Since repressurization in combination with these various actions had not achieved the desired natural circulation, a forced circulation mode was decided upon. At 15:33, reactor coolant pump 1A was turned on briefly and then off again. Though pressure initially dropped quickly, it again began to rise immediately after the pump was stopped. Inlet temperature and steam generator conditions indicate that again no natural circulation was achieved. At 15:50, reactor coolant pump 1A was again started. The steam-water separation in the primary system was broken, at least in loop A, and the system began cooling down via forced convection heat transfer via the loop A steam generator.

3.6.3 Component Behavior.

3.6.3.1 Steam Generator Behavior. During the depressurization phase, there was essentially no heat transfer via either steam generator. This was true even though the loop B steam generator secondary side was substantially colder than the fluid in the loop B hot or cold legs. As has been mentioned previously, heat transfer was blocked by the effect of the hydrogen gas on the transport of steam to the steam generator tubes. As the hydrogen was vented through the pressurizer, its concentration, particularly in the loop A hot leg, was reduced. These periods of minimal heat transfer took place at 10:25 and at about 12:48. With each such period of heat transfer, the secondary side pressure rose and the level dropped. Between heat transfer periods, pressure slowly dropped and level increased as the new equilibrium conditions affected the secondary side metal. This behavior is consistent with, and is the strongest indicator of, the fact that the loop A steam generator was isolated. As discussed in subsection 3.5.3.3, the secondary pressure and level traces for the loop A steam generator indicate that bypass to condenser vacuum was reestablished at 13:45. Loop B remains essentially inert throughout most of this period. The major changes in the parameters for the loop B steam generator are due to the level increase which started at about 11:40. Later, after about 12:00, some minimal heat transfer appears to occur, since the level trace shows a very slow decrease. Periods of evident heat transfer occur at about 12:00 and at 14:00.

3.6.3.2 Cold Leg Behavior. The cold legs in loops A and B are responding only to the occasional injection of cold water via the high pressure injection system and to the occasional feeding of the steam generators in the respective loops, and do not match conditions on the steam generator secondary sides. This response pattern in the cold legs confirms that there was no circulation pattern induced through the steam generators.

The loop B cold leg temperatures trend continually downward except for the two periods of heat transfer in the loop B steam generator evident from 11:40 to 12:00, and from 14:00 to 14:30. During these heat transfer periods, a slow drift of liquid from the tube region causes slightly increased temperatures in the loop B cold leg. Between periods of heat transfer, the temperature is responding to the injection of makeup water in loop B. Some of the injected water is driven backwards into the pump casing, where it drops into the cold leg loop seal region causing the indicated temperature to decrease. The continued injection of makeup water into the loop B cold leg apparently causes a sufficient flow bias to isolate the temperature indicator from changes in the cold legs induced by thermal-hydraulic transients in the downcomer. These downcomer transients are indicated, however, as changes in the loop A cold leg temperature.

The loop A cold leg temperature behavior is quite complex, and is best discussed by reference to the sets of external phenomena which are influencing that behavior. The clearest indications result from operation of the high pressure injection system, which occurs during four periods: 10:31 to 10:36, 11:19 to 11:28, 11:33 to 11:36, and 13:23 to 14:43. During these periods, injection flow is expected in the loop A cold legs, and some backflow into the pump casings and the region of the temperature indicator is expected. In each case, a noticeable reduction of indicated loop A cold leg temperature accompanies the injection period.

A major, sustained drop of the level in the pressurizer occurs, starting at about 11:08 (the cause is discussed in subsection 3.6.2). This drain is expected to pass saturated liquid into the loop A hot leg, and into the upper plenum of the reactor vessel. At this time, the loop A cold leg temperature rises rapidly, indicating the transport of steam and/or saturated liquid into the loop A cold leg. One possible flow path for this fluid is from the upper plenum, through the vent valves, indicating a local pressurization of the vessel upper plenum as a result of the presence of the saturated liquid. A second possibility is that the delivery of saturated water from the loop A hot leg to the upper plenum in the reactor vessel caused a slug of liquid to move downward through the core, pushing hot water from the downcomer into the loop A cold legs. The first of these options appears more reasonable, although neither is currently provable. From this time, until approximately 13:20, the loop A cold leg temperature remains constant, reflecting the static situation in the loop A cold legs when no injection is occurring. From about 13:20 to 14:30, a complex set of processes occurs. This is the long period of high pressure injection, and a temperature reduction results in the loop A cold leg. All cold leg temperatures represent subcooled liquid condition after this time, since the system has been repressurized. A second, less severe pressurizer drain occurs at the beginning of the period, but a cold leg temperature rise does not result. The loop A steam generator indicates heat transfer starting at about 13:45, and feeding occurs from about 13:55 to 14:10. At 14:10, the loop A cold leg temperature drops rapidly to 270°F, which is the temperature on the steam generator secondary side at that time. These data suggest that a slug of liquid might have been transferred from the steam generator tube region to the cold leg at 14:00, indicating a short period of liquid or two-phase natural circulation in loop A. If this circulation occurred, it was not sustained, since the cold leg temperature fails to track the steam generator temperature after 14:10. At 14:45, after termination of injection, the loop A cold leg temperature again begins to rise, indicative of hot liquid transport to the cold leg of loop A. Again, the likely path for this transport is via the vent valves.

3.6.3.3 Hot Leg Temperature. The hot leg regions of both loops contain a steam/gas mixture which is stratified in temperature throughout most of the period. The lower regions of the hot leg are at relatively low temperatures since they are close to the two-phase mixture in the vessel region. This is particularly true of the loop A hot leg, since that lower region has often been swept by flow to the pressurizer surge line. The upper bands of the hot legs contain highly superheated steam and hot hydrogen gas. The portion of the hot leg nearest the steam generator, and the upper region of the tubes contain steam and hydrogen, which are at temperatures consistent with the secondary side steam. Most of the changes in the loop A hot leg are explicable as shifts in this stratified column of steam so that the resistance temperature detector is reading the temperature of a different portion of the column.

While the relief block valve was open (prior to 11:08), the loop A hot leg temperature dropped, rose back above the recorded limit of 620°F, and then dropped again. Coincident with each drop, the steam generator pressure rises and the level drops, indicating a small amount of heat transfer. Each of these drops probably correspond to an incremental forward motion of the steam column. Between the two drops, and after the relief block valve closure, the static steam is reheated by the pipe walls in the region and a new equilibrium temperature is established.

When the relief block valve was opened later, from 12:36 to 12:47 and from 12:52 to 13:00, the system depressurized slightly. The temperature responds in the case of the first relief block valve opening as if some backward motion of the steam column occurred until a new pressure equilibrium was reached, at which point the column became static and was heated again by the walls. After the relief block valve closed, the column moved forward, reestablishing the old equilibrium and passing now cooler steam past the temperature detector. This forward motion apparently stimulated minimal heat transfer to the loop A steam generator secondary side, and a constant forward drift past the temperature detector which terminated at about 13:15. Some wall heating of the fluid occurred after that time, and the temperature rose. After 13:23, the primary system was being repressurized by the sustained operation of the high pressure injection system. This effect caused slowly rising temperatures as hot steam from the primary system was compressed into the hot legs. At about 14:38, the hot leg temperature began falling slowly. There was sufficient system overpressure to cause steam collapse before that time, but it is likely that the partial pressure of hydrogen in the hot leg was sufficient to retain saturated steam until that time. The change at 14:38 was due either to gross steam collapse, or to the appearance of liquid water in the region of the upper hot leg as the system refilling was nearing completion.

The wide range temperature data from the multipoint recorder indicate that the loop B hot leg is essentially static during this period. The temperatures show temporary responses to the relief block valve cycling prior to 13:00 and to the loop A pump start at 15:32.

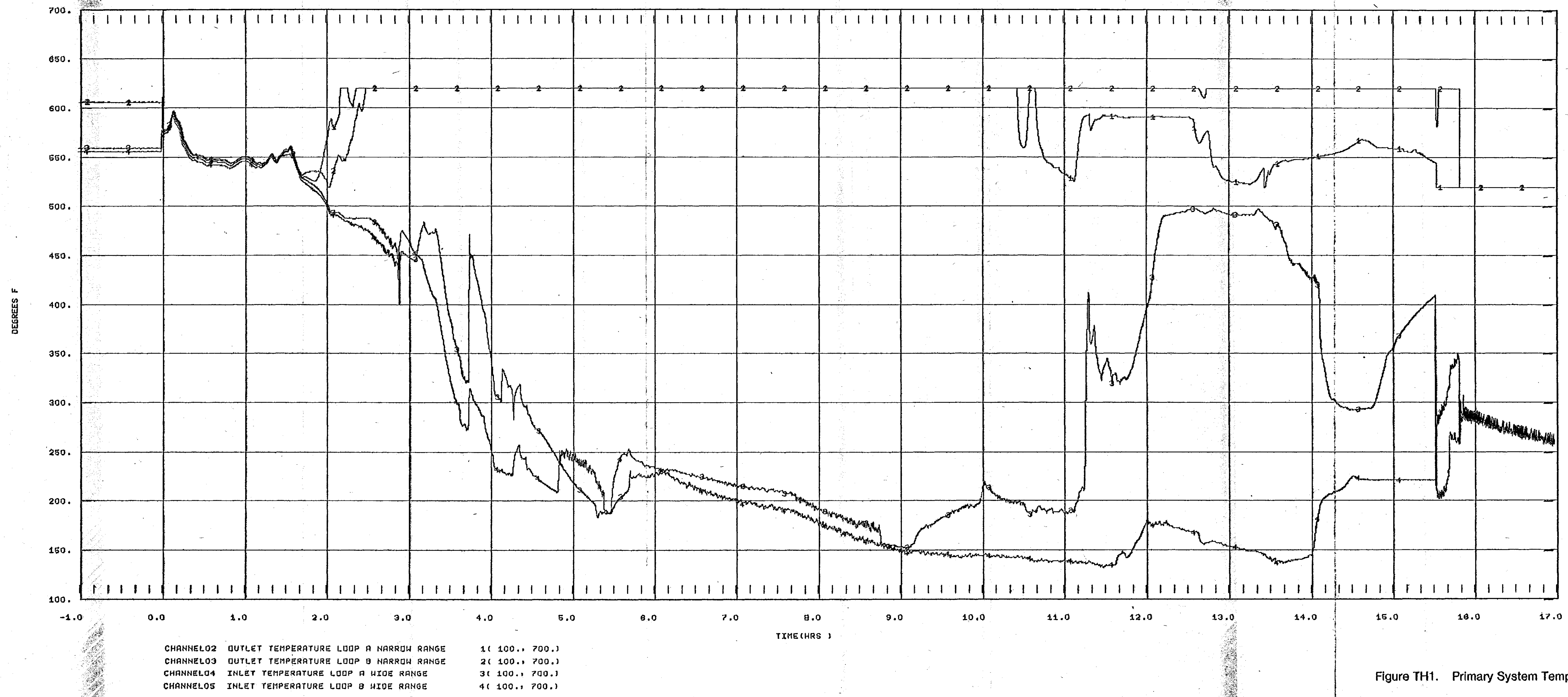


Figure TH1. Primary System Temperatures

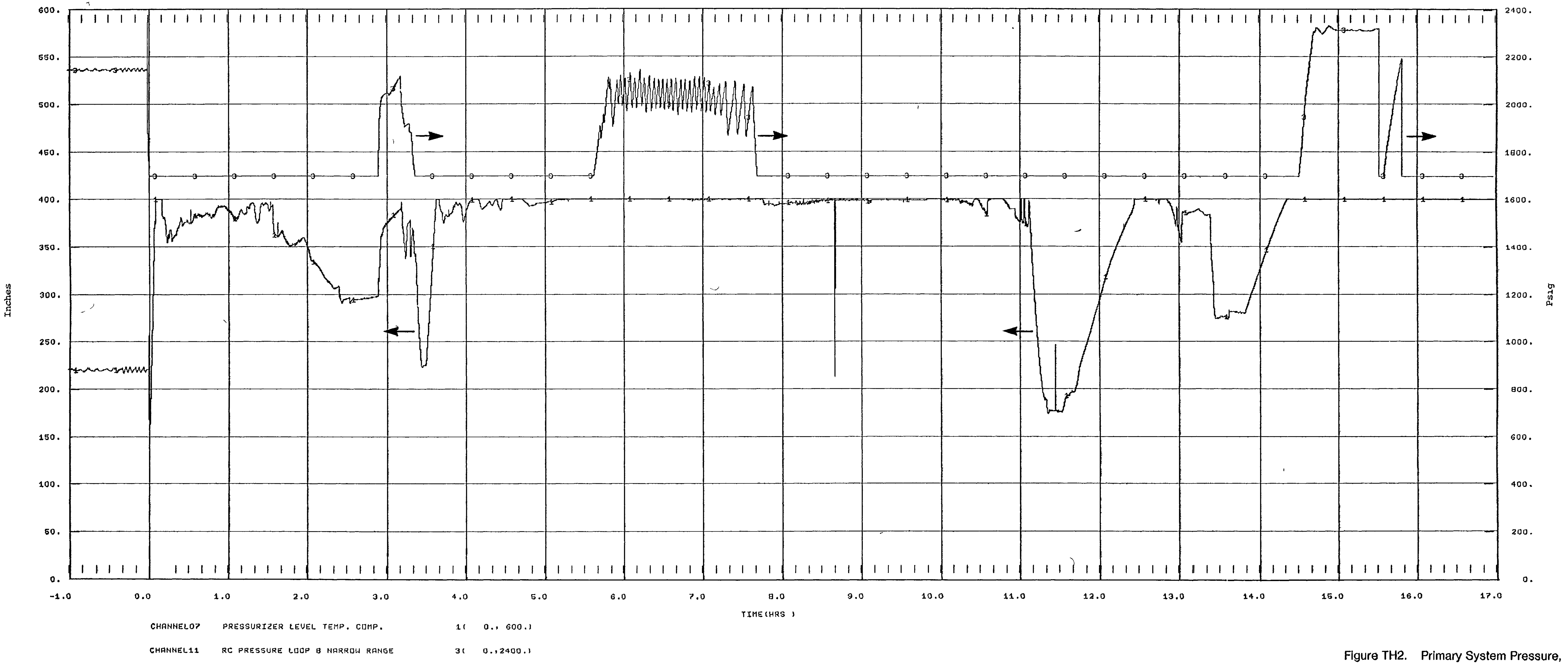


Figure TH2. Primary System Pressure, Pressurizer Level

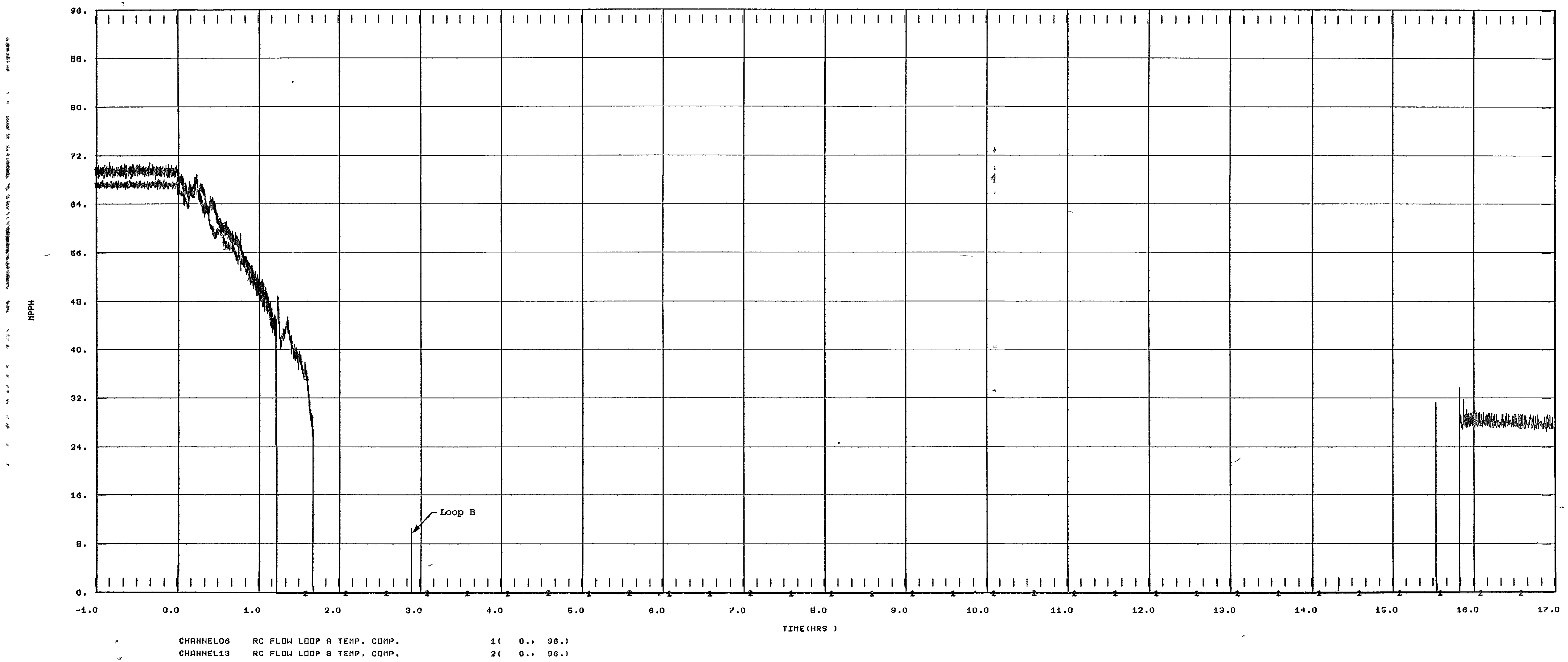


Figure TH3. Primary System Flow

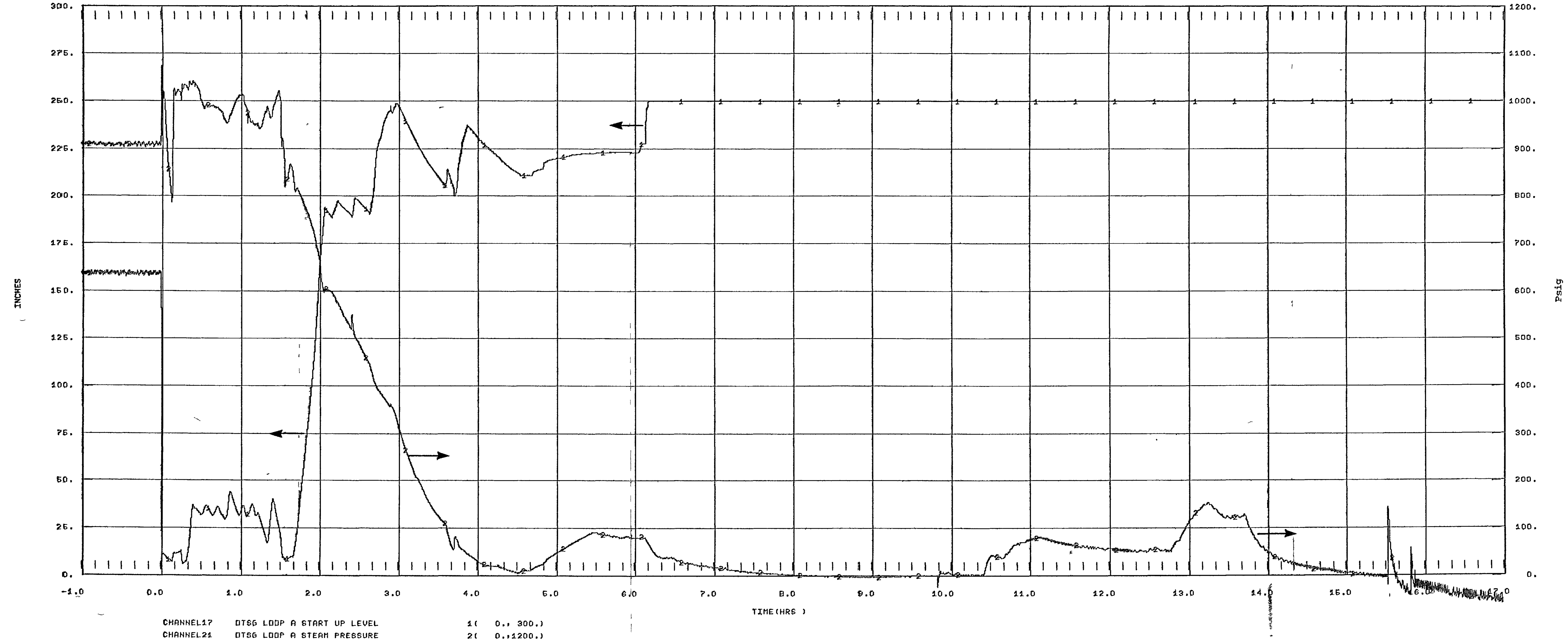


Figure TH4. Loop A Steam Pressure, Startup Level

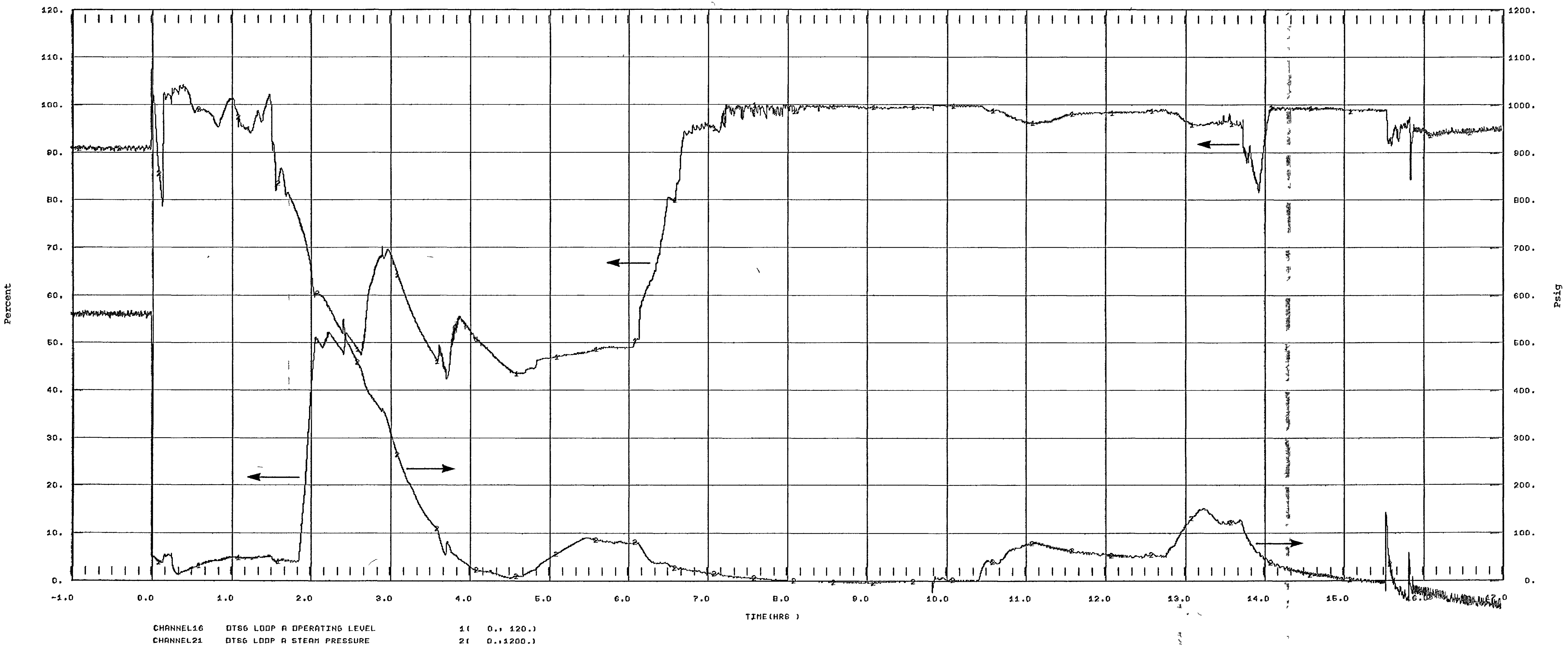


Figure TH5. Loop A Steam Pressure, Operating Level

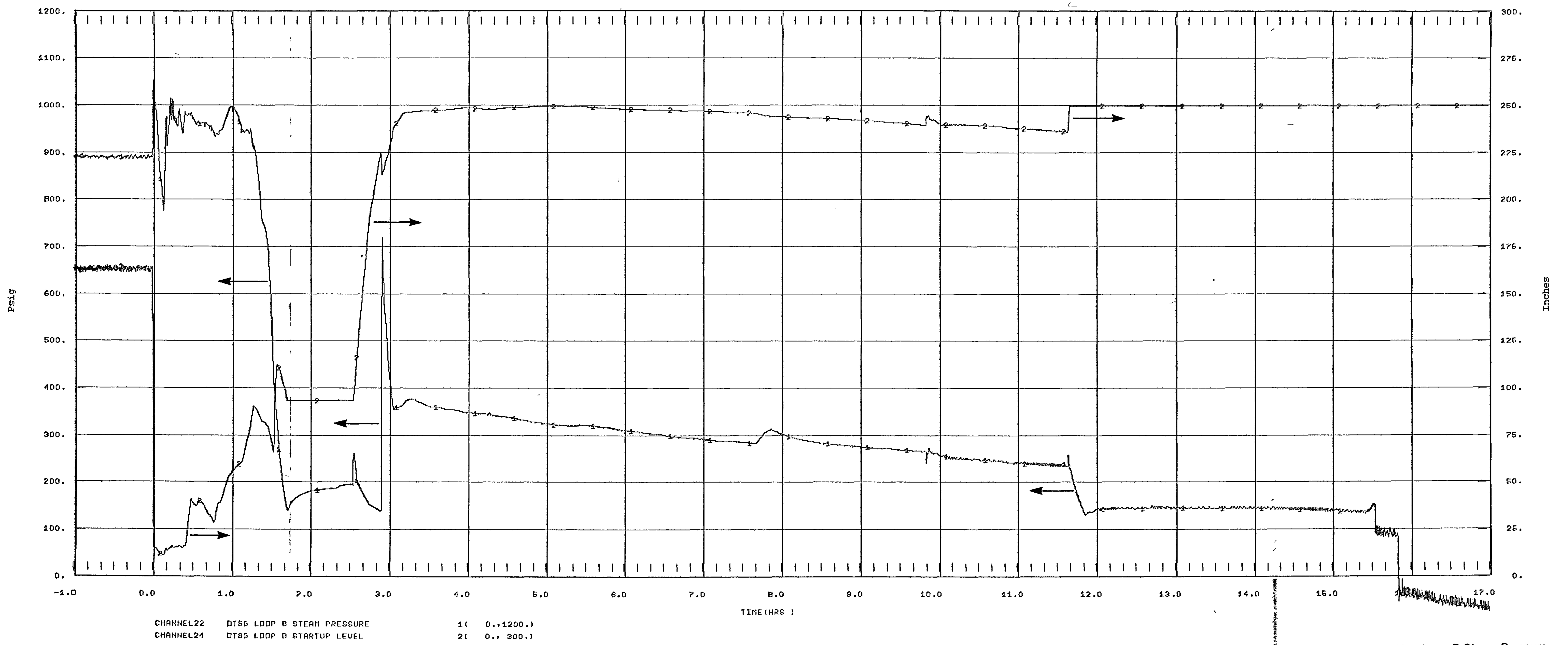


Figure TH6. Loop B Steam Pressure, Startup Level

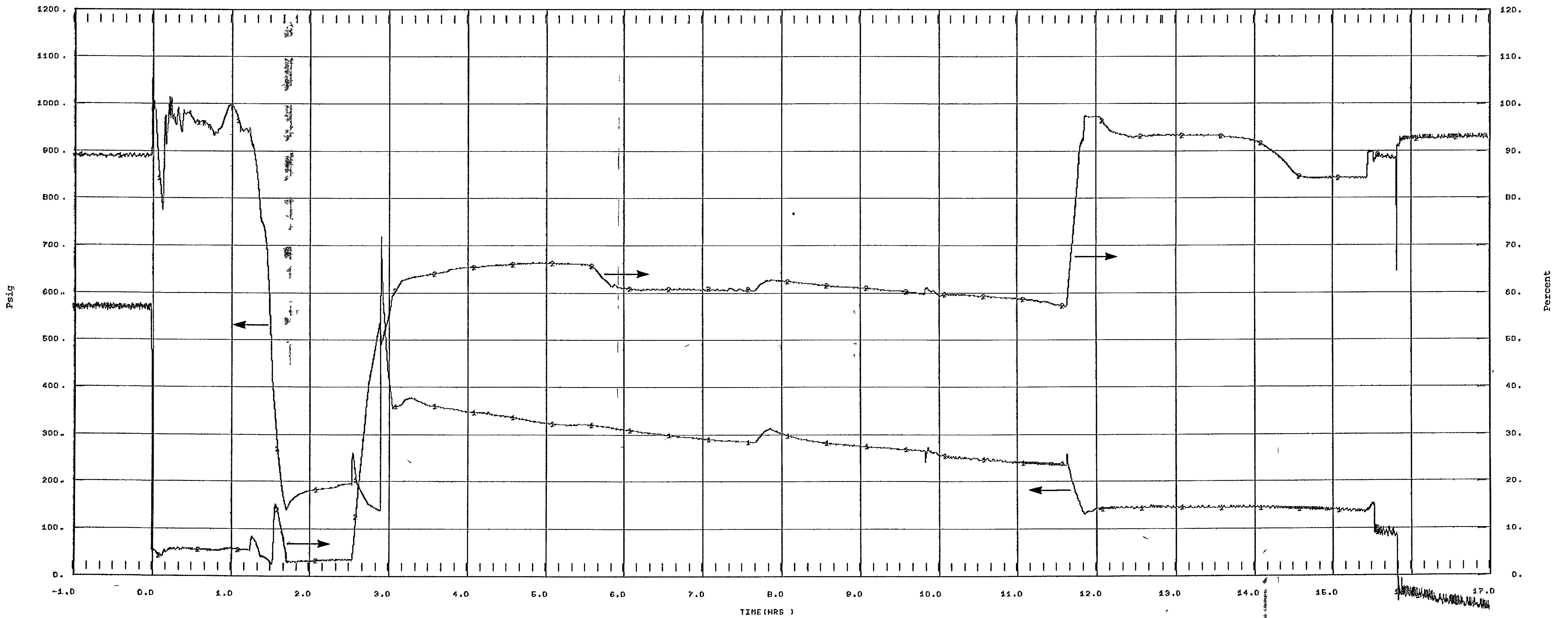


Figure TH7. Loop B Steam Pressure, Operating Level

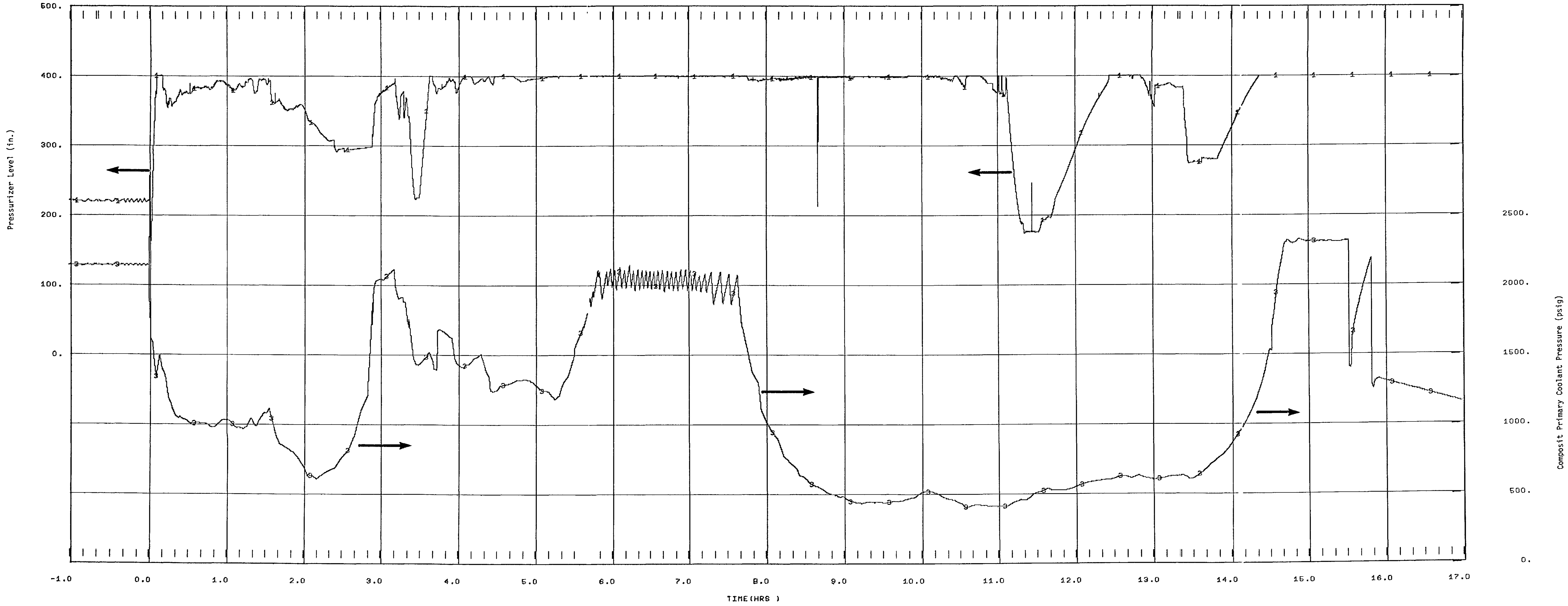


Figure TH8. Composite Primary Coolant Pressure and Pressurizer Level

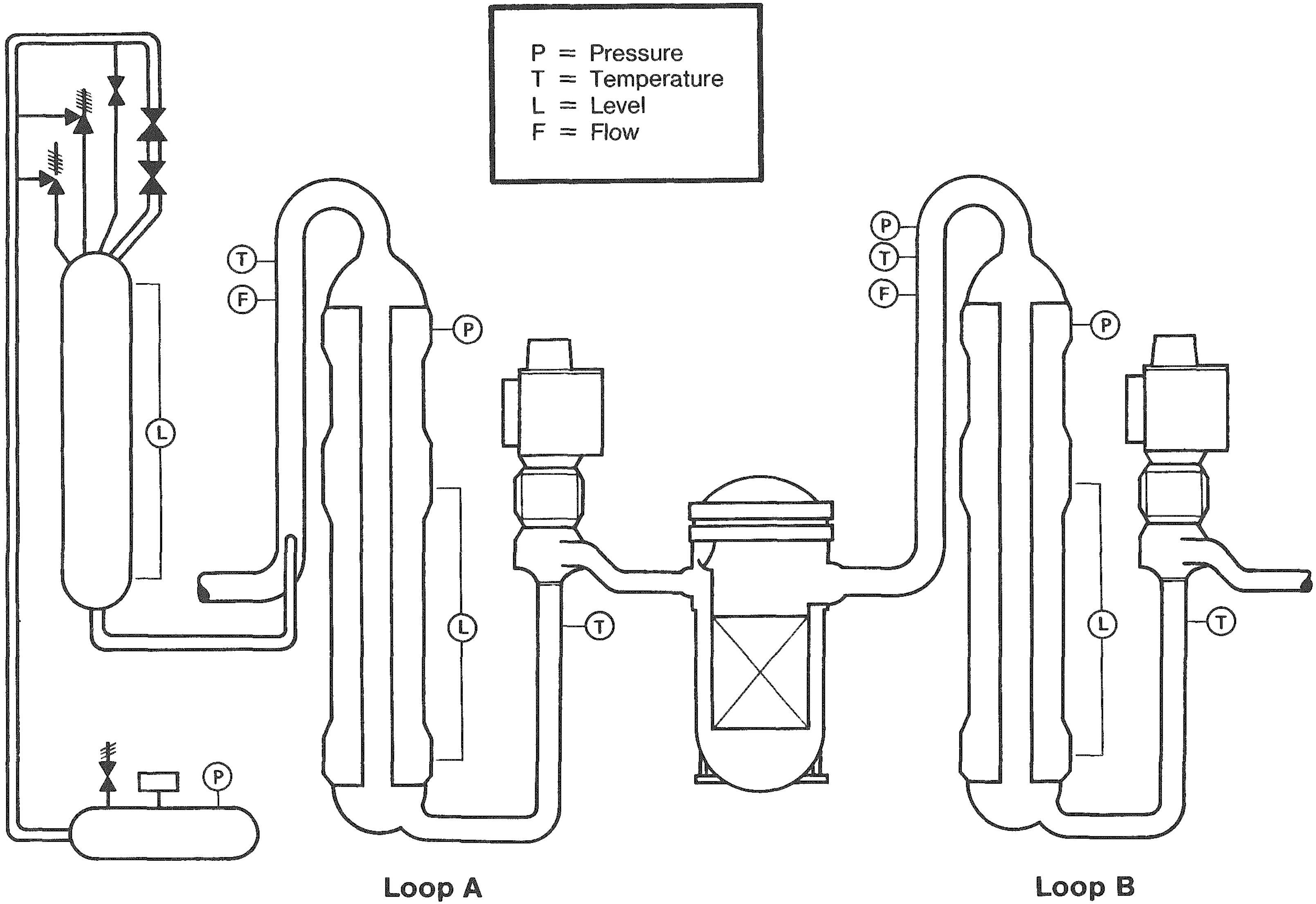


Figure TH9. Primary System Reactimeter Measurement

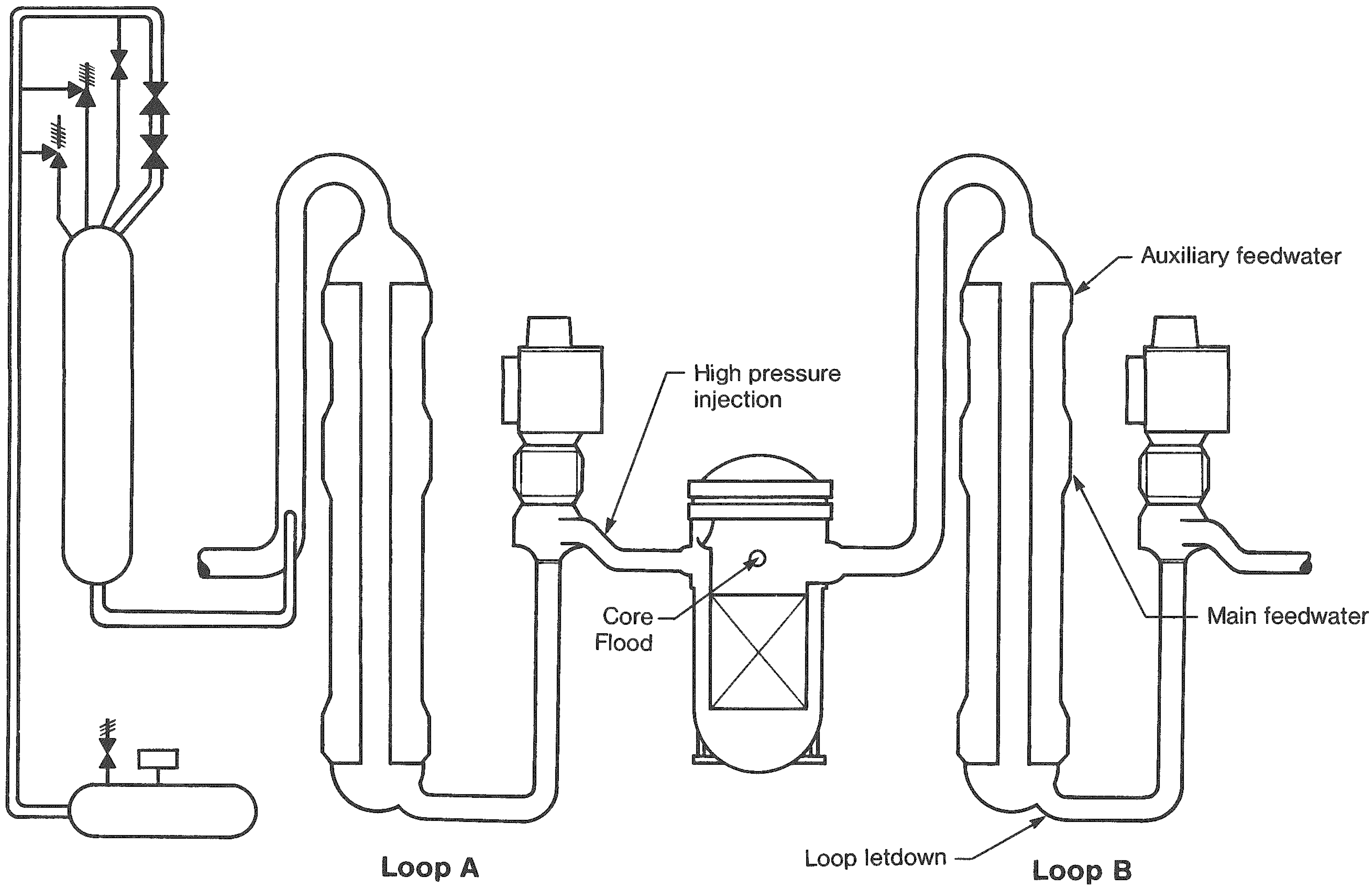


Figure TH10. Nozzle Locations

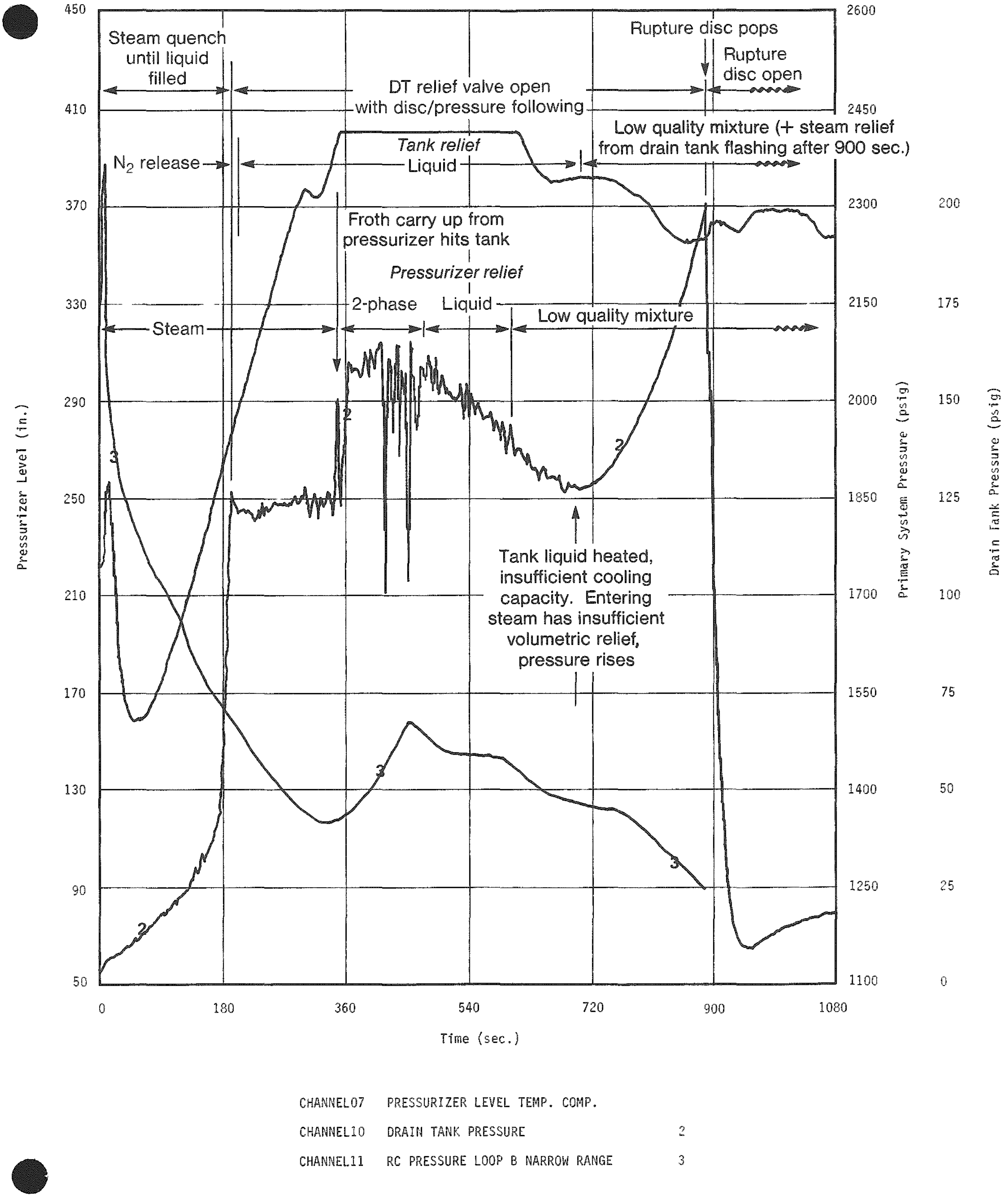


Figure TH11. Drain Tank Behavior

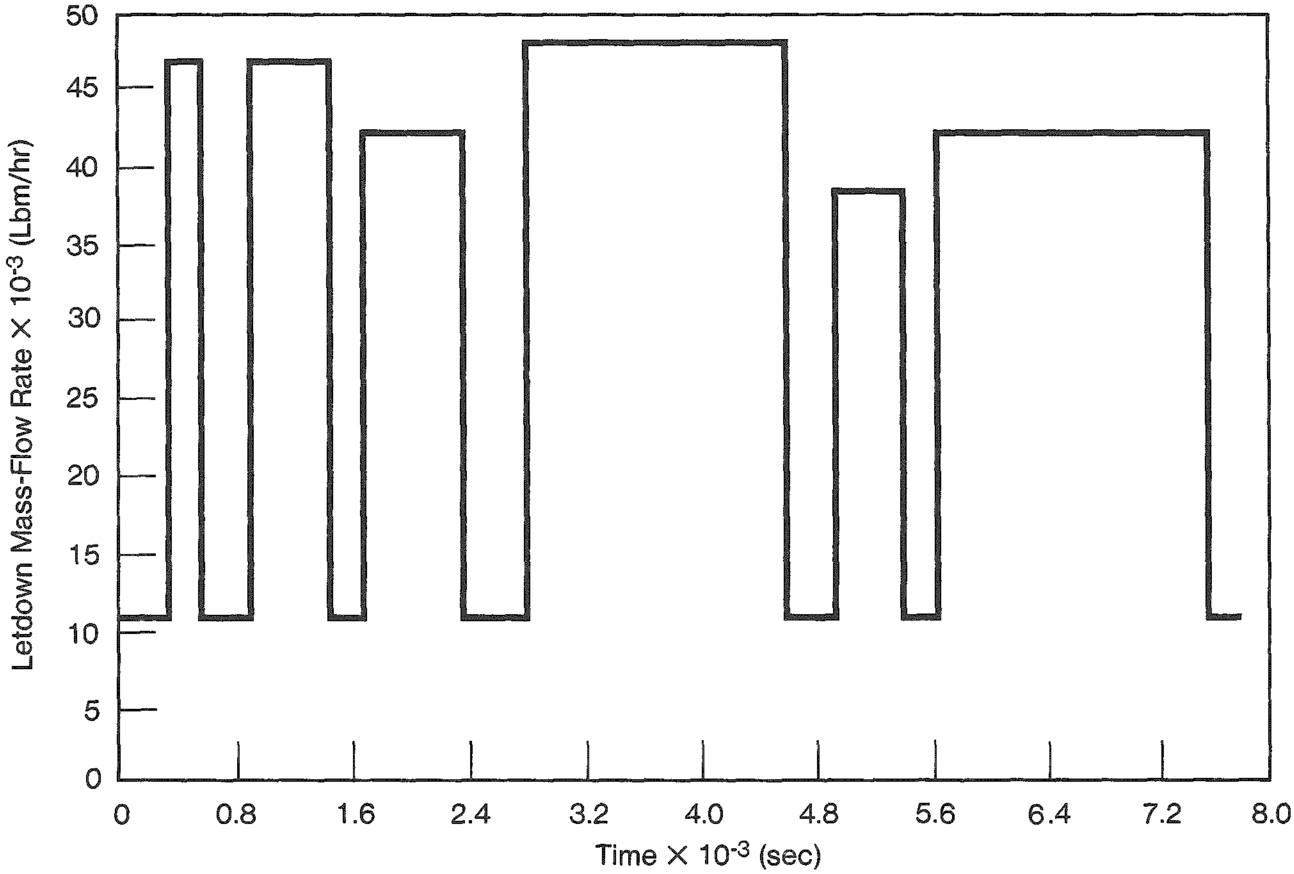


Figure TH12. Time History of Letdown Flow Through a Single Letdown Cooler

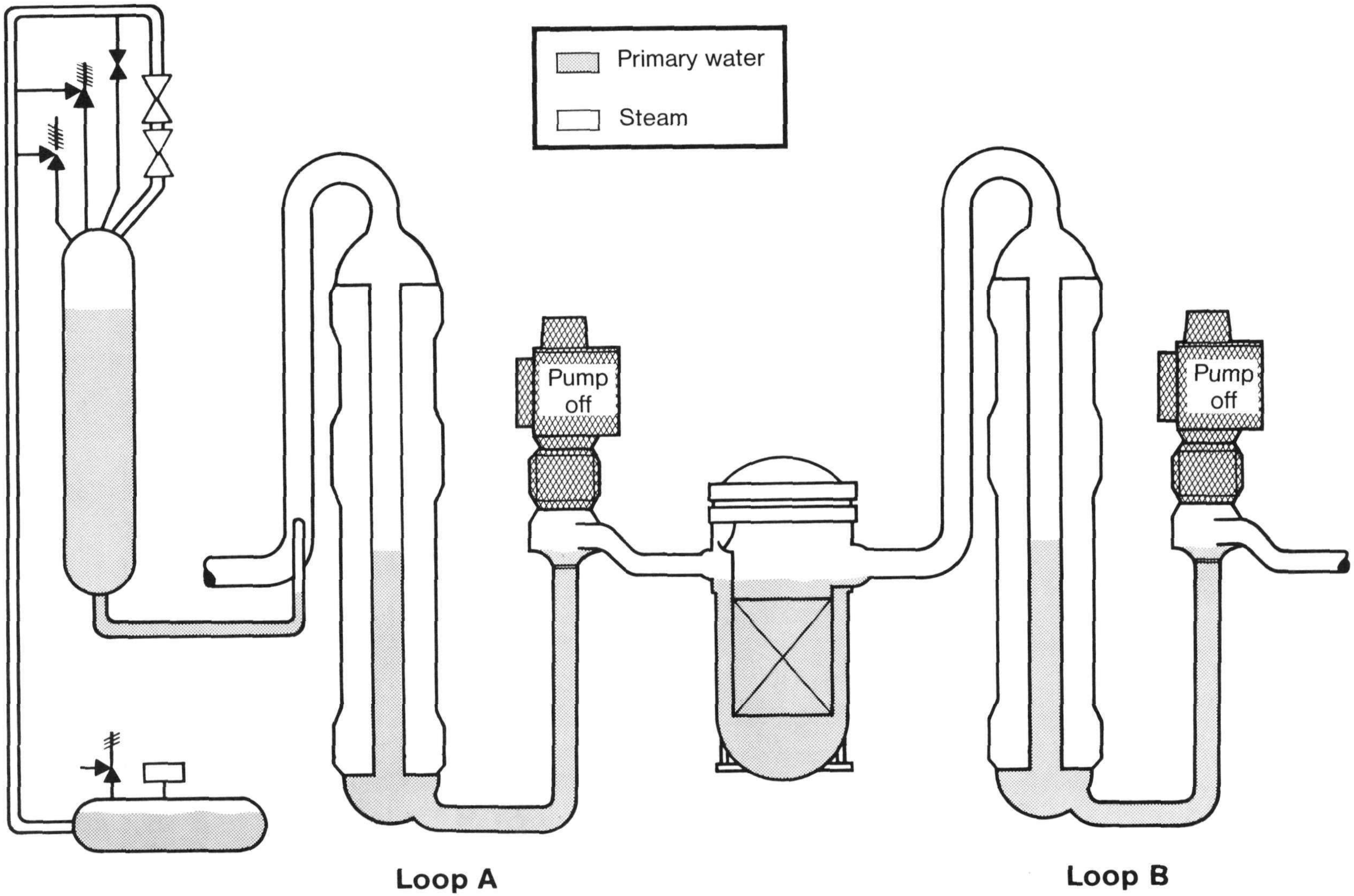


Figure TH13. Primary Inventory Following Loop A Pump Shutdown – Maximum (100 min.)

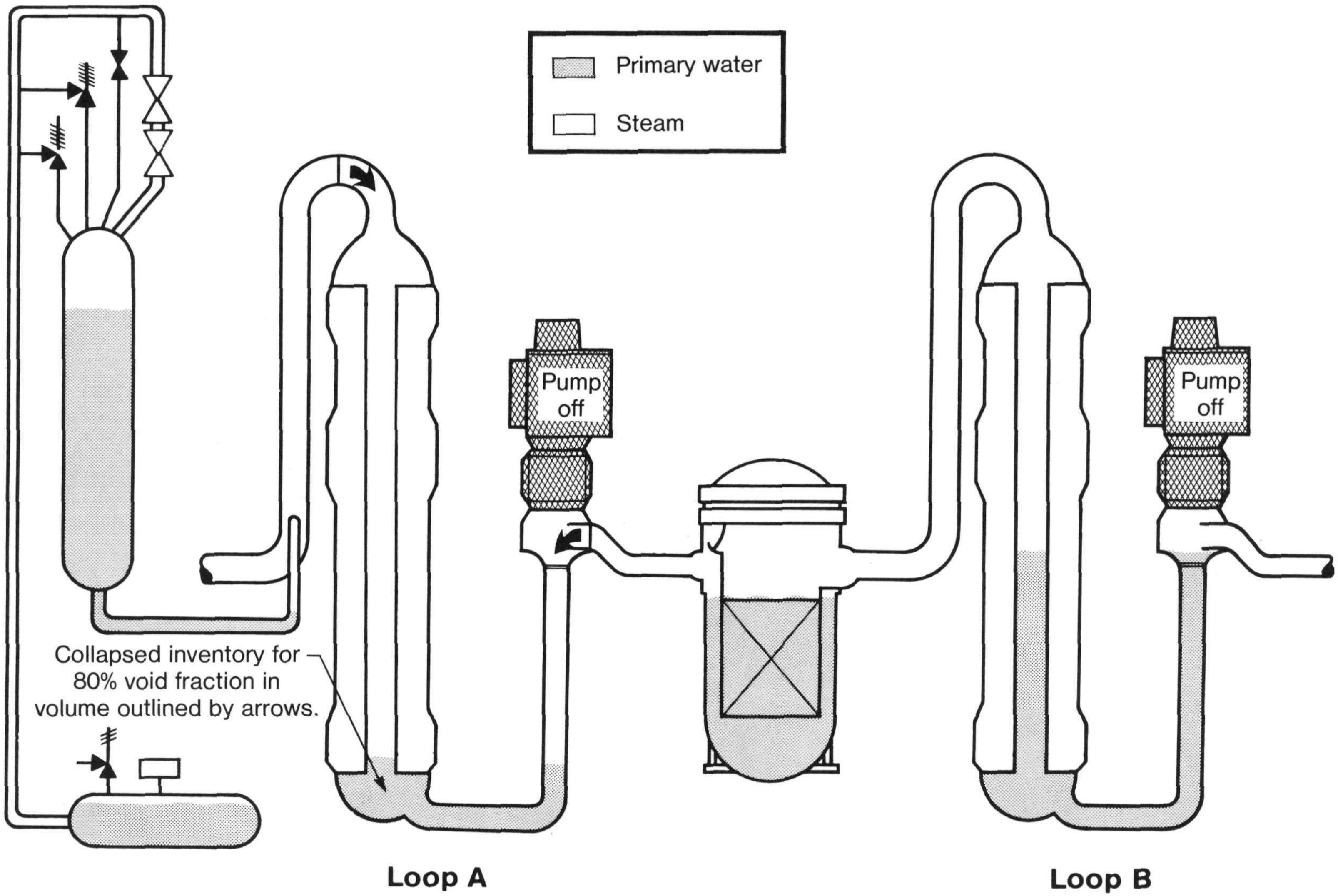


Figure TH14. Primary Inventory Following Loop A Pump Shutdown – Minimum (100 min.)

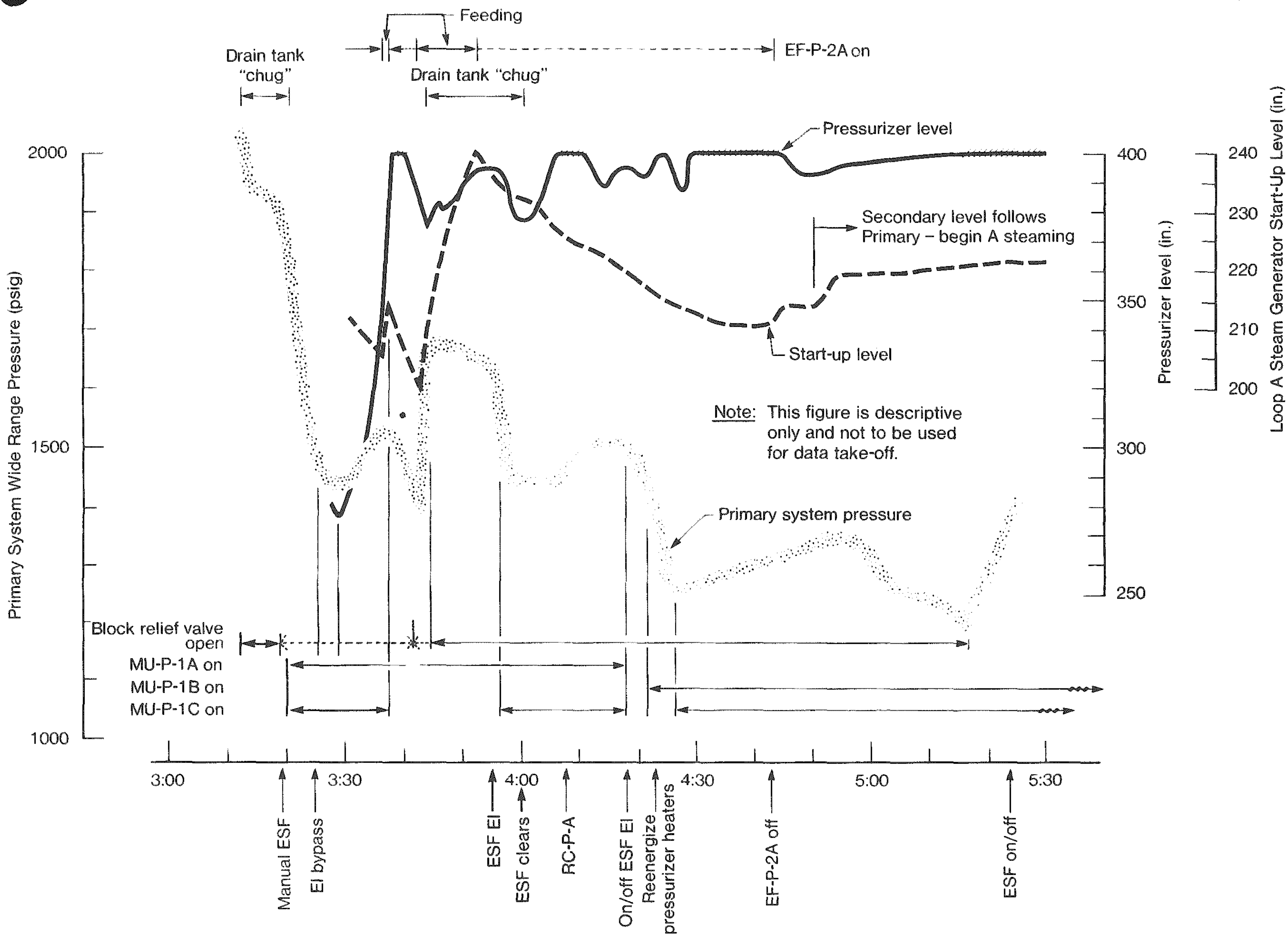


Figure TH15. Correlation of Events 3:30 to 5:30

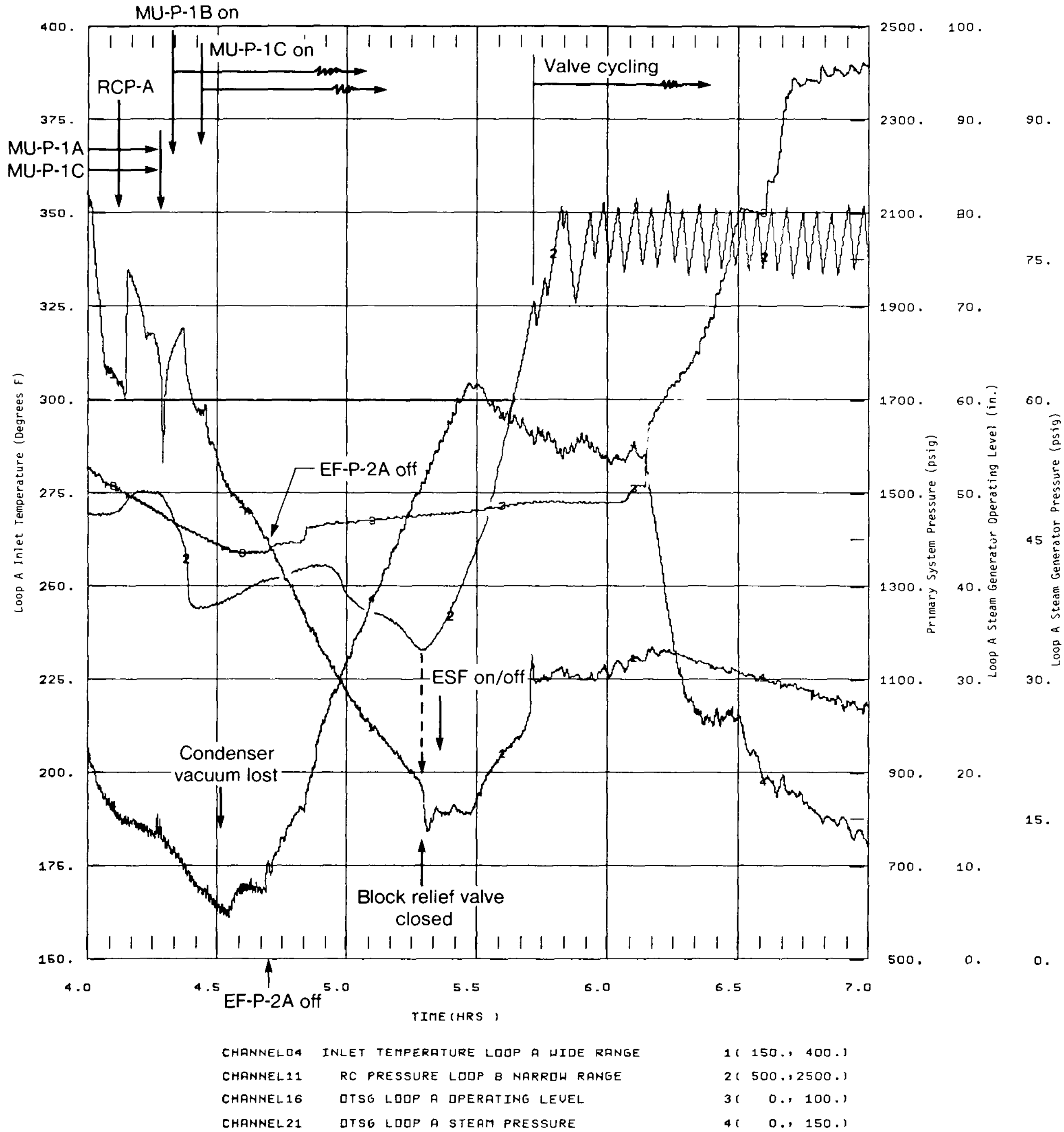


Figure TH16. Correlation of Events 4:00 to 7:00

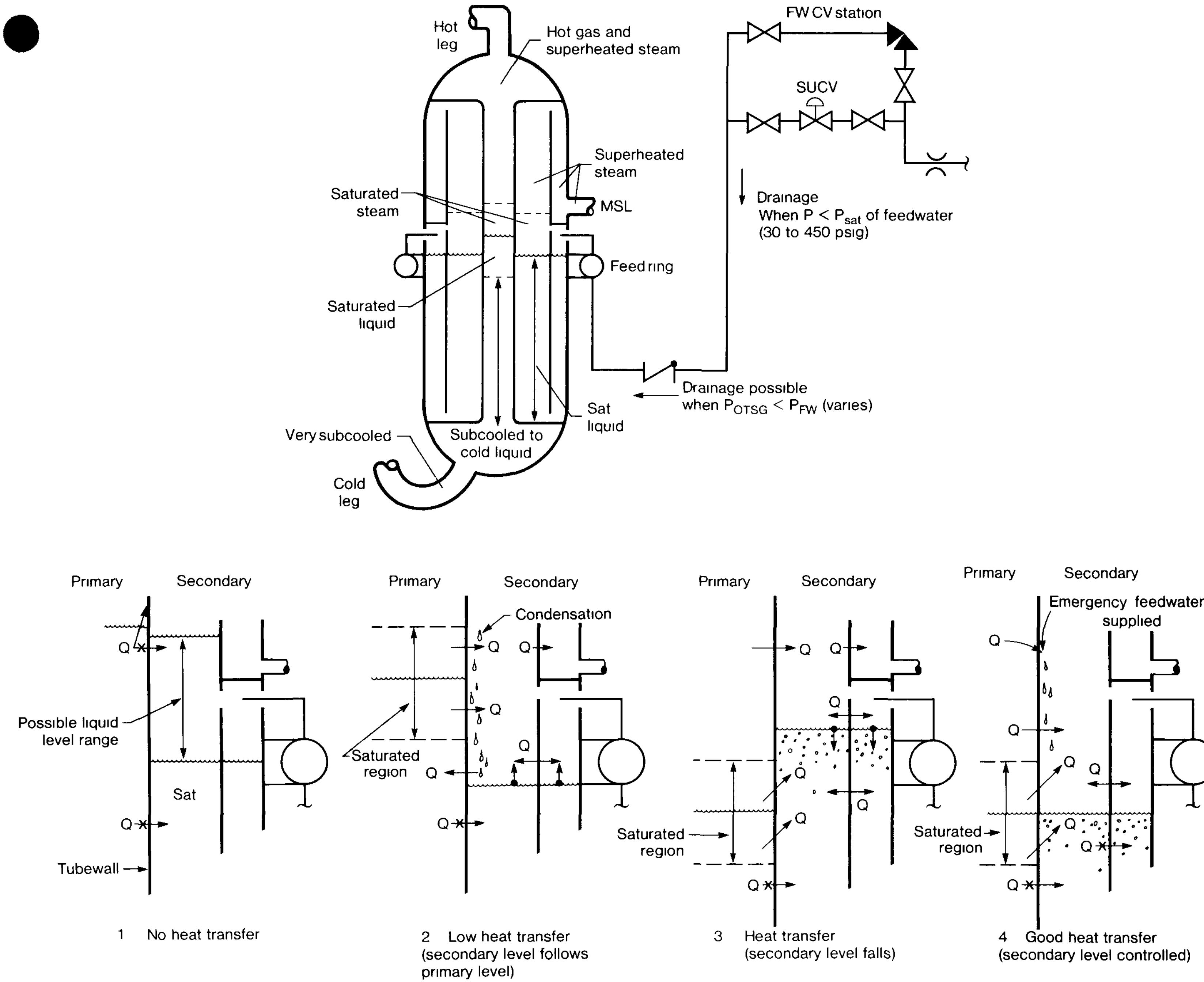
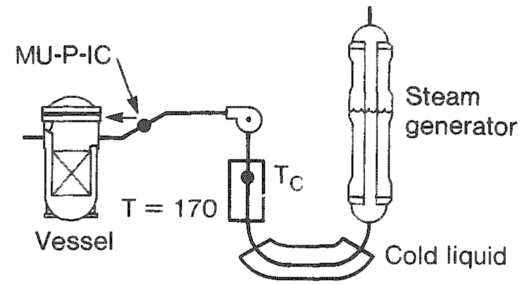
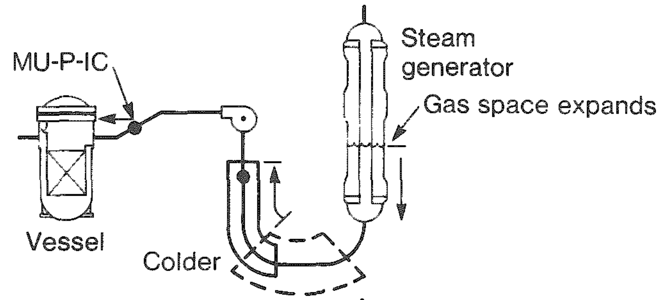


Figure TH17. Steam Generator Response With Primary Side Gas Block

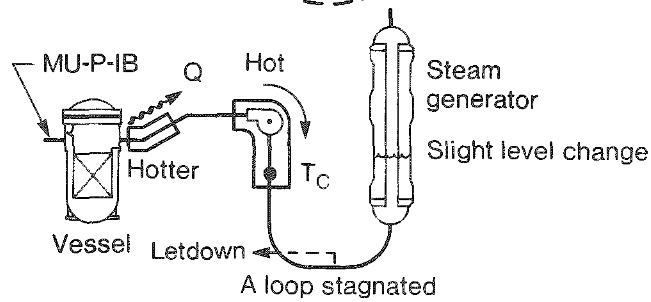
1. Initially at high pressure.
   T_C decreases slowly.



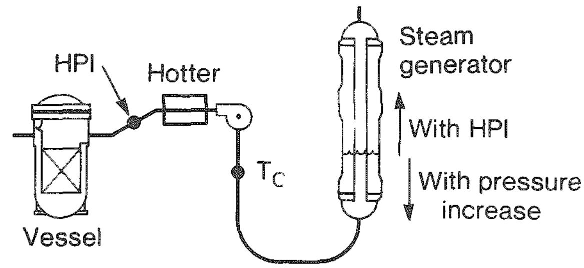
2. Depressurize.
   T_C decreases sharply and stabilizes.



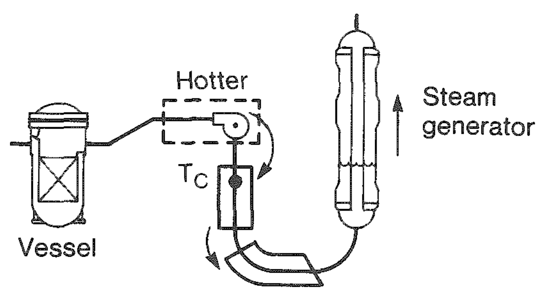
3. Trip MU-P-1C and close block relief valve (9:10).
   T_C increases with hotter liquid from near core.



4. HPI with primary system pressure increase (9:49).
   Counteracting effects: T_C levels out.



5. HPI ended. Primary system pressure increasing (9:53).
   Faster increase in T_C.



6. Primary system depressurization. Block relief
   valve open (10:00). T_C decreases.

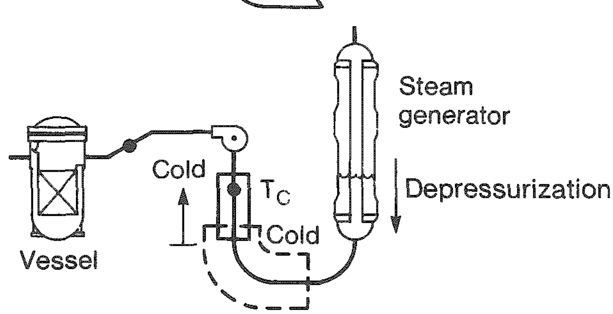


Figure TH18. Loop A Cold Leg Response to Depressurization

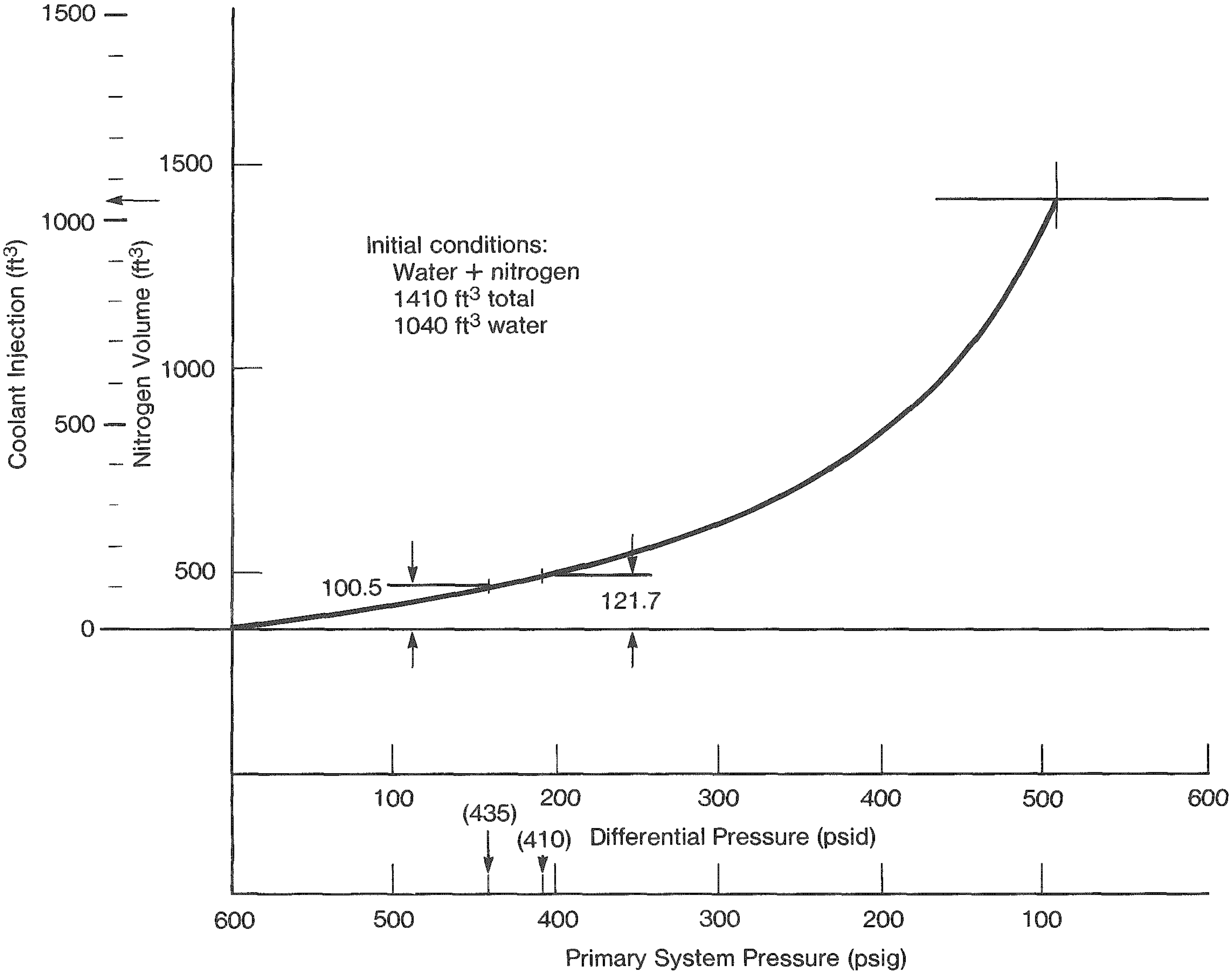


Figure TH19. Core Flood Capacity Versus System Capacity

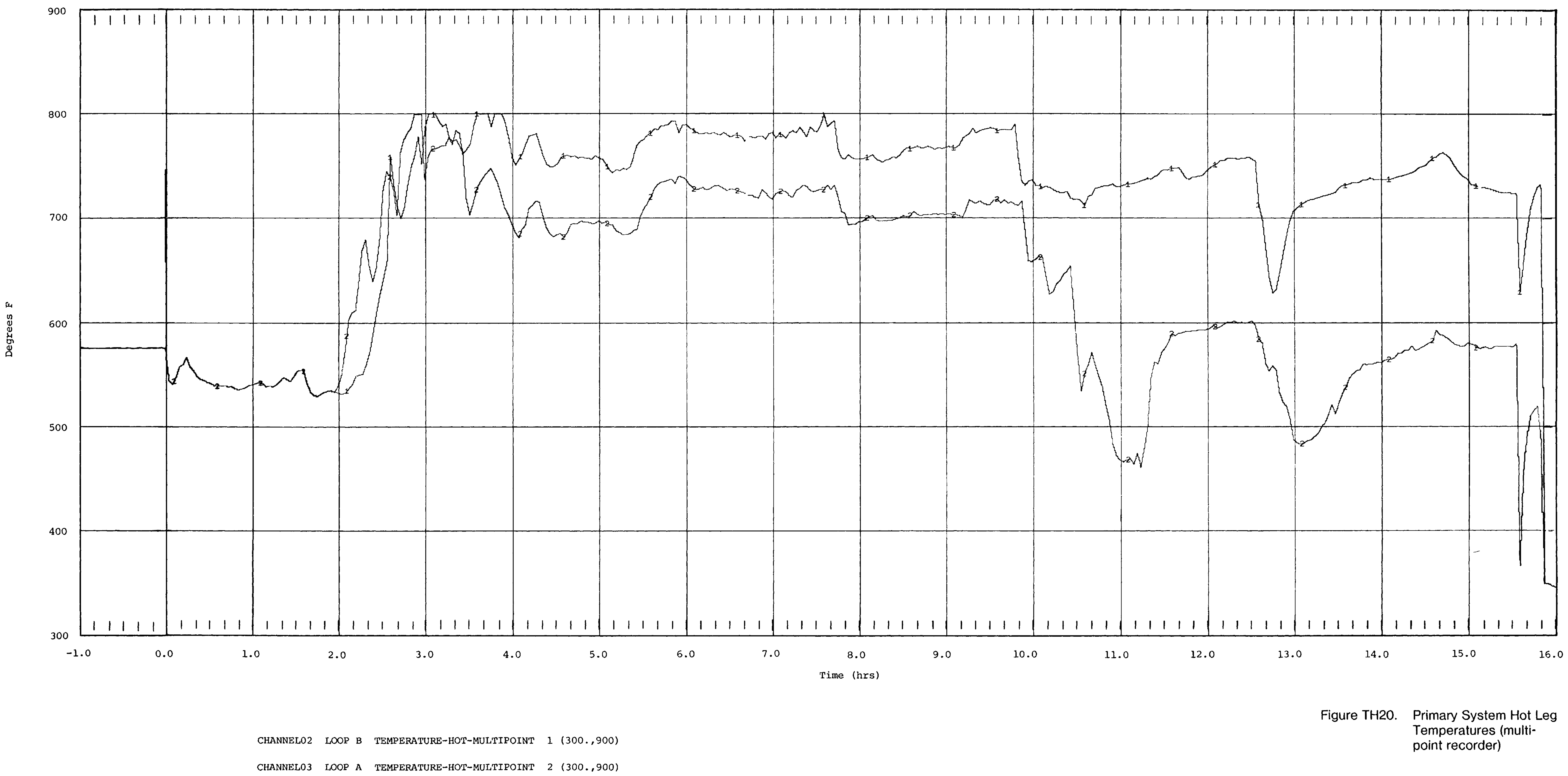


Figure TH20. Primary System Hot Leg Temperatures (multi-point recorder)

NSAC - SOE

STANDARD REFERENCE LIST

1. Reactimeter Data

a. Reactor Power - Power Range Level
b. Hot Leg Temperature Loop A - Narrow Range
c. Hot Leg Temperature Loop B - Narrow Range
d. Cold Leg Temperature Loop A - Wide Range
e. Cold Leg Temperature Loop B - Wide Range
f. Reactor Coolant Flow Loop A - Temperature Compensated
g. Pressurizer Level - Temperature Compensated
h. Makeup Tank Level
i. Pressurizer Spray Valve Position
j. Drain Tank Pressure
k. Reactor Coolant Pressure Loop B - Narrow Range
l. Reactor Trip
m. Reactor Coolant Flow Loop B - Temperature Compensated
n. Feedwater Temperature
o. Turbine Header Pressure Loop A
p. Steam Generator A Operating Level - Temperature Compensated
q. Steam Generator A Start-up Level
r. Feedwater Flow Loop A
s. Feedwater Flow Loop B
t. Turbine Trip
u. Steam Generator A Steam Pressure
v. Steam Generator B Steam Pressure
w. Steam Generator B Operating Level - Temperature Compensated
x. Steam Generator B Start-up Level

2. Operator Inverviews

a. Forum conducted by G. P. Miller, Station Manager, April 14, 1979

b. GPU Interview of Dick Dubiel, Station Chemistry/HP Supervisor; Gary Miller, Station Manager; and Jim Seelinger, TMI-l Superintendent on April 12, 1979

c. GPU Interview of Bill Zewe, Station Shift Supervisor, on March 30, 1979

d. GPU Interview of Craig Gaust, Control Room Operator, on March 30, 1979

e. GPU Interview of Ed Frederick, Control Room Operator on March 30, 1979

f. GPU Interview of Bill Zewe, Station Shift Supervisor, by O'Conner and team

h. GPU Interview of Ed Frederick and Craig Faust on March 29, 1979

i. GPU Interview of John Flint, B & W Representative on April 20, 1979

j. GPU Interview of Steve Mull, Auxiliary Operator, on March 30, 1979

k. GPU Interview of Don Miller, Auxiliary Operator, on March 30, 1979

l. GPU Interview of Dale Laudermilch, Auxiliary Operator, on March 30, 1979

m. GPU Interview of Fred Scheimann, Shift Foreman, on March 30, 1979

n. GPU Interview of Ken Bryan, Shift Supervisor, on April 26, 1979

o. GPU Interview of Craig Faust, Control Room Operator, on April 6, 1979

p. GPU Interview of Ed Frederick, Control Room Operator, on April 6, 1979

q. GPU Interview of Hugh McGovern, Control Operator, on May 4, 1979

r. GPU Interview of Brian Mehler, Shift Supervisor, on April 25, 1979

s. Events recorded by D. A. Berry, TMI Staff, From 1315 on March 28 to 0320 on March 29, 1979

3. Plant Computer Output

a.	Alarm Printer 0000 through 0500 on March 28, 1979	SOE-PC-0001
b.	Alarm Printer 0751 through 1848 on March 28, 1979	SOE-PC-0006
c.	Utility Printer 0324 through 0430 on March 28, 1979	SOE-PC-0002
d.	Utility Printer - Memory Trip Review 0345 through 0400 on March 28, 1979	SOE-PC-0003
e.	Utility Printer - Alarm Function 0502 through 0513 on March 28, 1979	SOE-PC-0004
f.	Utility Printer - 0444 through 0614 on March 28, 1979	SOE-PC-0002
g.	Utility Printer - Alarm Function 0648 through 0751 on March 28, 1979	SOE-PC-0005
h.	Utility Printer 0756 through 2008 on March 28, 1979	SOE-PC-0007

4. Control Room Stripcharts

a.	Reactor Building Pressure	SOE-SC-0001
b.	Intermediate Range Power Level	SOE-SC-0001
c.	Reactor Coolant T-ave	SOE-SC-0001
d.	Reactor Coolant Wide Range Pressure	SOE-SC-0001
e.	Turbine Throttle Pressure	SOE-SC-0001
f.	Source and Intermediate Range Power Level	SOE-SC-0001
g.	Multipoint Data on Self Powered Neutron Detectors - April 11, 1979	SOE-SC-0005
h.	Reactor Building Temperature Multi-Point Recorder (Microfilm print-out)	SOE-SC-0006
i.	Once Through Steam Generators and Primary System Temperatures Multi-point Recorder	SOE-SC-0007

5. <u>TMI-2 Plant Procedures</u>

a.	Operating Procedure	2101-1.1	Nuclear Plant Limits and Precautions	PRO-00-0002
b.	Operating Procedure	2101-2.1	Nuclear Plant Setpoints	PRO-00-0003
c.	Operating Procedure	2102-2.1	Power Operations	PRO-00-0004
d.	Operating Procedure	2102-3.1	Unit Shutdown	PRO-00-0005
e.	Operating Procedure	2102-3.2	Unit Cooldown	PRO-00-0006
f.	Operating Procedure	2102-3.3	Decay Heat Removal VIA-OTSG	PRO-00-0007
g.	Operating Procedure	2103-1.3	Pressurizer Operation	PRO-00-0008
h.	Operating Procedure	2104-1.1	Core Flooding System	PRO-00-0009
i.	Operating Procedure	2104-1.2	Makeup and Purification System	PRO-00-0010
j.	Operating Procedure	2104-1.3	Decay Heat Removal System	PRO-00-0011
k.	Operating Procedure	2104-4.6	Reactor Coolant Leakage Recovery System	PRO-00-0012
l.	Operating Procedure	2104-6.3	Emergency Feedwater	PRO-00-0013
m.	Operating Procedure	2105-1.1	Nuclear Instrumentation	PRO-00-0014
n.	Operating Procedure	2105-1.2	Reactor Protection System	PRO-00-0015
o.	Operating Procedure	2105-1.3	Safety Features Activation System	PRO-00-0016
p.	Operating Procedure	2105-1.4	Integrated Control System	PRO-00-0017
q.	Operating Procedure	2105-1.5	Incore Monitoring System	PRO-00-0018
r.	Operating Procedure	2106-2.4	Feedwater	PRO-00-0019

s.	Emergency Procedure	2202-1.1	Reactor Trip	PRO-00-0020
t.	Emergency Procedure	2202-1.3	Loss of Rx Coolant Coolant Sys. Press.	PRO-00-0021
u.	Emergency Procedure	2202-1.4	Loss of RC Flow/ RC Pump Trip	PRO-00-0022
v.	Emergency Procedure	2202-1.5	Pressurizer System Failure	PRO-00-0023
w.	Emergency Procedure	2202-2.2	Loss of Steam Generator Feed	PRO-00-0024
x.	Emergency Procedure	2202-2.6	OTSG Tube Rupture	PRO-00-0025
y.	Operating Procedure	2105-1.6	RCS-NNI Operation	PRO-00-0026

6. Core Damage Assessments

a.	Bounding Estimates of the Damage to Zircalloy Fuel in the TMI-2 Core at Three Hours After the Start of the Accident (NRC)	ANA-CO-0001
b.	Fuel Experts Meeting on Condition of the TMI Core (NRC)	ANA-CO-0002
c.	Maximum Steady-State Clad Temperatures Reached in TMI Voiding (NRC)	ANA-CO-0003
d.	TMI-2 Incore Thermocouple Indications of Flow Blockage (PNL)	ANA-CO-0004
e.	TMI-2 Core Integrity Assessment (Industry Advisor Group)	ANA-CO-0005
f.	TMI-2 Reactor Core Status Instrumentation Data Package (IAG)	ANA-CO-0006
g.	ANL Core Damage Assessment Memo 5-2-79 (Persiani)	ANA-CO-0007
h.	Core Blockage Study (Meyer)	ANA-CO-0008
i.	Bettis Core Damage Assessment from Radiochemistry Results	ANA-CO-0009
j.	Degradation Core Melt in TMI-2 and Potential Consequences (NRC)	ANA-CO-0010
k.	Notes from the TMI-2 Core Damage Assessment Meeting (IAG 4-27-79)	ANA-CO-0011
l.	Summary of Tests Performed, Results, Conclusions and Opinions on SPNDs and Core Thermocouples	ANA-CO-0012
m.	Core Damage Assessment for TMI-2 (Meyer - NRC)	ANA-CO-0013

7. <u>TMI-2 Operator Logs</u>

a.	Shift Foreman Log 2300, March 27, 1979 to 0900, April 1, 1979	SOE-OL-0002
b.	Operator Log March 25 through April 2, 1979	SOE-OL-0001

8. Core Thermocouple Data

a.	Core Thermocouple Maps, March 28 - April 1, 1979	SOE-MD-0002
b.	Thermocouple Readings 0800 - 0900 on March 28, 1979	SOE-MD-0001

9. TMI-2 Drawings

(Identify drawing numbers and titles using the Burns & Roe and the Vendors' Drawing Lists.)

10. TMI-2 Final Safety Analysis Report

(Identify Volume and Section)

11. <u>TMI-2 Technical Specifications</u> LIC-00-0001

(Identify specification number and title)

12. TMI-2 Radiation Release Data

a. Preliminary study of radiation monitor stripcharts performed at Three Mile Island. The data has not been independently verified by a NUSAD study.

ABBREVIATIONS AND ACRONYMS*

AMS	- Automatic Mode Selector Switch in Feedwater System
AUX BLDG	- Auxiliary Building
B & R	- Burnes and Roe
BWST	- Borated Water Storage Tank
CF	- Core Flood System
C/FDW	- Condensate and Feedwater System
CI	- Core Instrumentation
COMP	- Computer. Part of Plant Data System
ERV	- Electromatic Relief Valve
ES	- Engineered Safeguard
ESF	- Engineered Safety Feature
FSAR	- Final Safety Analysis Report
H & V	- Heating and Ventilating
HPI	- High Pressure Injection
ICS	- Integrated Control System
MCC	- Motor Control Center
NPSH	- Net Positive Suction Head
NSAC	- Nuclear Safety Analysis Center of EPRI
OTSG	- Once Through Steam Generator
PDS	- Plant Data System
RB	- Reactor Building
RC	- Reactor Coolant
RCDT	- Reactor Coolant Drain Tank
RCP	- Reactor Coolant Pump
RCPCS	- Reactor Coolant Pressure Control System
RCS	- Reactor Coolant System
REACT	- Reactimeter. Part of plant data system
RM	- Radiation Monitor
ROUTES	- Routes for radiation escape from containment

*For key to alpha-numeric equipment designations, see Appendix ESF.

ABBREVIATIONS AND ACRYONMS
(continued)

SCR	-	Silicon Controlled Rectifier
SI	-	Safety Injection
SOE	-	Sequence of Events
SP	-	Reactor Building Sumps
SPND	-	Self Powered Neutron Detector
STEAM DUMP	-	Main Steam System
TMI	-	Three Mile Island Nuclear Station

GPU 2205

Analysis of Three Mile Island – Unit 2 Accident

NSAC-80-1

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