PDR-016

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 13 1985

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IN RESPONSE REFER
TO FOIA-85-240

Dear Mr. Freire:

This is in partial response to your undated letter in which you requested, pursuant to the Freedom of Information Act (FOIA), records related to fuel loading and/or low power testing of the Diablo Canyon nuclear power plant; a copy of the damage claim against NRC filed by GPU on December 8, 1980; and a copy of the September 1980 report prepared by the NRC Office for the Analysis and Evaluation of Operational Data (AEOD) regarding interim equipment and procedures at the Browns Ferry nuclear power plant to detect water in discharge.

We are enclosing copies of the two documents listed on the enclosed Appendix A which respond to your request.

The NRC has not completed its search for and review of any additional documents which may be subject to your request. We will communicate with you again when search and review are completed.

Sincerely,

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Enclosures: As stated

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Re: FOIA-85-240

APPENDIX A

1. 09/80 AEOD Report: The Interim Equipment and Procedures At Browns Ferry to Detect Water In The Scram Discharge Volume (34 pages)

2. 12/08/80 GPU Damage Claim Against NRC (34 pages)

AEOD/C002

REPORT ON

THE INTERIM EQUIPMENT AND PROCEDURES AT

BROWNS FERRY TO DETECT WATER IN THE SCRAM DISCHARGE VOLUME

by the

OFFICE FOR ANALYSIS AND EVALUATON OF

OPERATIONAL DATA

September 1980

Prepared by: George Lanik

NOTE: This report documents results of studies completed to date by the Office for Analysis and Evaluation of Operational Data with regard to a particular operating event. The findings and recommendations contained in this report are provided in support of other ongoing NRC activities concerning this event. Since the studies are ongoing, the report is not necessarily final, and the findings and recommendations do not represent the position or requirements of the responsible program office of the Nuclear Regulatory Commission.

EXECUTIVE SUMMARY

On June 28, 1980, the Browns Ferry Unit 3 reactor experienced a partial failure to scram while shutting down for a scheduled outage. As reported by the Office for Analysis and Evaluation of Operational Data (AEOD) on July 30, 1980, the apparent cause of this event was found to be water accumulation in the Scram Discharge Volume (SDV) prior to the attempted scram. The AEOD study identified possible fundamental deficiencies in the SDV which cast doubt on the ability of the Scram Discharge Volume/Scram Instrument Volume (SIV) to adequately perform their intended functions. In view of these deficiencies, AEOD recommended design changes to improve the performance of the scram system for the long term.

Following the event, the Office of Inspection and Enforcement (IE) issued Bulletin 80-17 and Supplement Nos. 1, 2, and 3. Supplement 3 was issued in response to the concerns raised by the AEOD memorandum of August 18, 1980 which identified degraded air pressure in the control air system as a mechanism which could rapidly fill the SDV. The equipment and procedural changes required by Bulletin 80-17 and its Supplements are intended to provide the basis for continued operation of BWR's during the period prior to completion of design changes to the scram system.

AEOD has evaluated the procedures and equipment at the Browns Ferry Units 1, 2 and 3 to determine their adequacy with respect to providing assurance that the SDV will not fill with water and interfere with a successful scram. This evaluation applies specifically to the Browns Ferry units. However, the findings and recommendations should be considered in the review process for all applicable BWR's.

The principal findings of the study are summarized below:

- The present system, which uses recently installed ultrasonic water detection equipment and special procedures, in conjunction with previously installed instrumentation and procedures, does not restore the level of

scram protection capability thought to be assured in the original design. However, except for degraded control air pressure events, it does provide adequate assurance for the interim that, accumulation of water in the Scram Discharge Volume (from currently identified sources), which could result in a loss of scram capability, will be reliably detected and adequately responded to by the operator.

- Degraded HCU control air pressure could result in scram outlet valve leakage to the SDV which would require operator action to manually scram the reactor within a few minutes before scram capability would be completely lost. Control air related disruptions in the plant would likely also initiate a plant disturbance which would require a scram. Such an event would be accompanied by numerous control room alarms and indications which could distract the operator from a prompt manual scram actuation. The current system does not adequately assure sufficient time for operator diagnosis and actions for this event.

- Operating experience indicates that a significant number of reactor scrams attributed to loss of HCU control air pressure have occurred. These provide evidence that rapid filling of the SDV is a credible event.

The principal recommendations of the study are as follows:

- An immediate manual scram should be required based on control room indication of degraded HCU control air pressure. Review of licensee proposals should include consideration of the available pressure indications and procedures to assure that other alarms and indications do not divert operator attention from this priority action.

- Redundant HCU air header pressure instrumentation should be provided in the control room. To aid the operator in quickly focusing his attention on the need for protective action, a distinctive alarm for degraded air pressure should be provided.

- Because of the possibility that a currently unidentified water source could result in water accumulation in the SDV, it would be prudent to monitor the ultrasonic system alarm output in the control room and require an immediate verification of a sustained alarm by operator dispatch to the equipment. Operability and calibration checks of the system should be continued on a schedule of once per shift.

The conclusions of the study are summarized below:

AEOD has reviewed the interim surveillance system at Browns Ferry used to detect the presence of water in the SDV. The AEOD assessment considers the procedures and equipment changes initiated in response to IE Bulletin 80-17 with Supplements, 1, 2, and 3 to be adequate for continued interim operation of the Browns Ferry Nuclear Plant, if the recommmendations of this report relating to degraded control air pressure are implemented.

As of the date of this report, the instrumentation and procedures in place to respond to the loss of control air scenario at Browns Ferry are judged to be inadequate. For this event the operator must respond promptly to a single indistinctive alarm for loss of control air pressure during a period when numerous alarms may be occurring. Additionally, the operator must take actions outside the control room in a very limited time frame because of the absence of a pressure readout in the control room. IE is currently taking steps to upgrade the procedure for response to the degraded control air pressure event.

In the past, operator action to perform a vital safety function within less than 10 minutes has not been considered acceptable by the NRC. However, providing the operator with both a distinctive low pressure alarm and reliable air pressure instrumentation in the control room would help assure adequate operator response within the required time period. Such an arrangement should be acceptable for the interim. A dedicated operator with adequate alarms and instrumentation in the control room could provide even greater assurance of a timely manual scram. If the provisions made to accomplish a manual scram are

found to be untimely or inadequate, provisions should be made for an automatic scram on low HCU control air pressure.

For the long-term, the scram system should be upgraded according to the recommendations of the AEOD report of July 30, 1980. However, the consequences of degraded air pressure in the HCU control air headers were not fully recognized at the time of that report and were not directly addressed. Although the recommended scram system modifications may be sufficient to enable the scram system to respond to rapid inflows of water from the scram outlet valves due to degraded HCU control air pressure, design review of the long-term modifications should include specific consideration of the effects of degraded air pressure.

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1. INTRODUCTION

On June 28, 1980, the Browns Ferry 3 reactor experienced a partial failure of the scram system while shutting down for a scheduled outage. The operators were able to completely insert all control rods within 14 minutes of the initial scram attempt. Because of the initial partial success of inserting rods on the first scram and because no unplanned transient requiring a scram was in progress, no immediate challenge to reactor safety and integrity developed.

As documented in the AEOD report dated July 30, 1980, [1] the cause of this event was found to be water accumulation in the East Bank Scram Discharge Volume (SDV) prior to the first attempted scram. Following the event, IE Bulletin 80-17 and Supplement Nos. 1, 2, and 3 were issued. These directed BWR licensees to begin surveillance of the SDV to detect the presence of water. A requirement for continuous monitoring of the SDV water level in the control room was stated in Supplement No. 1. Scram system problems revealed by testing subsequent to the Browns Ferry event were reported in Supplement No. 2. Supplement No. 3 was issued in response to the concerns raised by the AEOD memorandum of August 18, 1980[2]. This supplement required operator actions for a loss of control air to the Hydraulic Control Units (HCUs).

The following report is an evaluation of the current measures being taken at Browns Ferry in response to the IE bulletin and supplements to prevent events of the type that occurred on June 28, 1980. This assessment was undertaken by AEOD because of its concern about the adequacy of the interim system which will be used during the period preceding the implementation of long-term corrective measures. The scope of this report is purposely limited to: a) Browns Ferry Units 1, 2, and 3: b) the interim measures; c) selected bulletin requirements; and d) procedures and equipment in place on the date of this report.

The findings, recommendations, and conclusions are based on information gathered through informal channels between AEOD and the Tennessee Valley Authority, the General Electric Company, and the U.S. NRC headquarters and regional offices.

Section 2 of this report contains a description of the present equipment and procedures at Browns Ferry used to prevent a recurrence of the failure to scram event. Section 3 provides an AEOD evaluation of the effectiveness of the present system (equipment plus procedures) for providing a timely response to a range of postulated scenarios. Sections 4 and 5 present, respectively, the findings and recommendations. The conclusions are given in Section 6.

2. SYSTEM DESCRIPTION

To compensate for the identified deficiencies[1] associated with the protection system instrumentation installed at Browns Ferry prior to the June 28, 1980 event, additonal hardware and operating procedures have been put in place. The additional equipment installed at Browns Ferry for monitoring the SDV for the presence of water is an ultrasonic (UT) system. Ultrasonic transducers are mounted on the East and West SDV header low points. The transducer is driven by a signal generating and processing device which incorporates a cathode ray tube (CRT) display and provides an output to a strip chart recorder.

Unit 3 has eleven transducers located as shown in Figure 1. Units 1 and 2 each have four transducers located as shown in Figures 2 and 3. Unit 3 was instrumented to a greater extent to attempt to find the cause of the June 28, 1980, partial scram event. Since completing the testing of SDV drainage, long-term monitoring has been limited to use of transducers #2 and #7 on Unit 3; transducers #12 and #13 on Unit 2; and transducers #14 and #15 on Unit 1. In the case of failure of these transducers, a backup transducer is available on each header. The transducers are bonded to the headers with a high temperature adhesive.

The pulse-echo technique of depth measurement is used and is illustrated in Figure 4. The top illustration shows a cross section of the SDV pipe on which the transducers are mounted. The bottom illustration shows the CRT display arising from this situation. Since sound travels one-fourth the speed in water as in steel, the reflection from the inner tube wall is received very quickly following the initial pulse. This is shown on the left hand side of the CRT display in Figure 4. Multiple reflections are seen on the CRT because of sound reflections between the inner and outer diameter of the pipe. These show a decreasing amplitude and die out rapidly.

The sample illustration is shown containing 5.2 inches of water. A second series of echoes is received at a later time on the CRT indicating the

reflection from the water-air surface at a distance of 5.2 inches. The instrument has been previously calibrated on a pipe with a known water level. The numbers shown on the horizontal axis of the CRT display correspond to the depth in inches of water in the SDV above the transducer location.

A continuous recorder is provided. By use of gating devices, it is possible to pick the signal of interest to look at which is the water-air interface and not the pipe inside diameter (i.d.). The gating device is set to gate signals which come in at a time later than those corresponding to one inch of water. This eliminates the reflection from the i.d. of the SDV pipe and the associated multiples. The gating device is also set to gate only those signals with an amplitude greater than approximately 20% of full-scale amplitude. The first signal associated with a given pulse to pass the gate is transmitted to the recorder. The recorder is a two channel recorder; one channel records the amplitude of the gated signal, and the other records the calibrated depth of water associated with the gated signal.

A local alarm is provided. Any echo signal which passes the gate will generate an audible and visual alarm. The alarm is generated when the water level is greater than one inch and self-clears when the level is less than one inch.

Two characteristics of the gating method used are of particular interest with respect to the recorder output: (1) only water depths greater than one inch are recorded; and (2) only the first echo received at a depth greater than one inch is recorded.

When no water is present in the header, the echo from the pipe i.d. is the only return pulse. Since this initial pulse and its multiples indicate less than one inch, nothing is gated to the recorder. The second pulse indicating water level never comes. The recorder sees this as a long delayed second pulse. Thus, the normal empty header condition reads full scale on the recorder. The recorder full scale reads ten inches. Since the full pipe condition would read only six inches, there is no confusion in the reading. When a pulse is

gated, indicating the presence of water in the header, the recorder pen is driven down toward the lower part of the scale. (See lower portion of Figure 5).

At Browns Ferry Unit 3, two separate UT devices and recorders are provided to monitor both the East and West SDV headers independently. On Units 1 and 2, however, a single UT device is used to drive and monitor two transducers at the same time, one on the East SDV header and one on the West SDV header. This provides another characteristic of the system that must be recognized by those operating the system. The gating device passes the first pulse which returns corresponding to a depth greater than one inch, and the recorder responds to this pulse. With water in both headers, the indication seen on the recorder corresponds to the first returning echo greater than one inch.

Thus, if both East and West headers have a water depth above one inch, the smaller depth indication is recorded because it is the first pulse to return. Individual measurements for either side can be made by disconnecting the cable from the transducer on the header opposite the side of interest.

The following is a brief discussion of the recorder output from a scram test at Browns Ferry 2 (Refer to Figure 5). Figure 5 shows only the calibrated water depth trace. The amplitude trace has been omitted for simplicity. Increasing time is from the bottom upward with one division equal to approximately 5 minutes. Water depth is measured in inches starting from zero on the right hand side of the trace to 10 inches on the left. Note that the pen location prior to the scram at 0202 hours is full scale left (10 inches). This is because no water is present and the second echo never returns, which the instrument interprets as maximum distance from the bottom mounted transducer. The momentary readings where the pen is driven downward (to the right) prior to the scram are due to the instrument reacting to the "walkie-talkies" used for communication. These momentary readings also activate the visual and audio alarms which clear each time the walkie-talkie transmission stops. Since Unit 2 was scrammed at 0202 hours the water level indicates 6 inches. The trace has been blacked in below the 6 inch level for emphasis. At about

0230 hours the indicated water level falls from 6 inches to 1 inch. This is due to the West header going empty. At the time when the level indicator reaches 1 inch, the pen is driven back up to the 6 inch level. This is because on Unit 2, a single UT instrument is used to monitor both East and West headers. Since the gating device passes the first returning signal above one inch, the recorder tracks the header which empties first (West side) and when the echo from the West side indicates less than one inch, the gating device begins to pass the echo from the East side header. Since the East side header has not yet emptied, the pen is driven back up to about 6 inches. Between 0234 and 0256 hours, the East side header continues to drain. When the level reaches one inch on the East side, no returning echo is gated and the pen returns to the 10 inch position. As indicated on the trace, a series of momentary indications of water are present at 0440 hours. These are due to some CRD survillance tests which were run at that time.

2.1 Calibration and Operation

All tranducers used in the system were tested prior to use to assure adequate performance. The gain of the signal generating and receiving equipment is adjusted to provide an adequate signal output from the least responsive transducer. The minimum acceptable signal for reflection from the water interface is adjusted for 80% full scale output on the CRT display. The gating device is set to pass any signal with an amplitude more than approximately 20% of full scale so as to provide an adequate margin of sensitivity.

The time scale on the CRT is adjusted to read the depth of the water in the header in terms of horizontal divisions on the CRT screen. As shown in Figure 4, a total of ten horizontal divisions are used on the CRT display. The sweep time and the horizontal centering of the CRT are adjusted so that the sixth division on the screen corresponds to a water depth of 6 inches while the first division on the screen corresponds to a water depth of one inch. Thus, the CRT displays water depth directly. If no water is present, only the echo from the i.d. of the pipe is displayed on the CRT screen at a position below the one inch mark.

Initial system calibration and later checking of the calibration is done by use of standard pipes filled with known amounts of water. Once per shift, a level two QC inspector takes a reading from a standard containing 2 inches of water and from a standard containing 6 inches of water. This is done by disconnecting the cable from the transducers on the headers and connecting it to hand-held transducer which is held against the bottom of the standard sample pipes. If the reading on the CRT and the recorder does not agree with the known depth of water in the standard pipes, the gain and amplitude of the UT instrument are adjusted to recalibrate the system. At this time, all transducers are functinally checked by examining the CRT display for indications of transducer deterioration.

The two UT instruments on Unit 3 are physically located at the ends of the rows of HCUs, one on the East side and one on the West side. On Units 1 and 2, the UT instruments are located on a mezzanine level above the HCU level approximately midway between the two sides.

Browns Ferry has an auxiliary operator on each shift who observes the UT system recorder strip chart for each unit every 30 minutes. The operator is not qualified or required to monitor the CRT output. His sole responsibility is to monitor the strip chart recorder and the alarm. The calibration and operability of each UT device and each transducer is checked once per shift by a level two QC inspector trained in the use of UT equipment. Communication with the control room concerning SDV water accumulation is by a hand held walkie-talkie.

As mentioned earlier, a separate UT transducer, CRT and recorder is provided for each side on Unit 3, while Units 1 and 2 each have a single UT, CRT and recorder to monitor both the East and West SDV header transducers. Thus for Units 1 and 2, since the recorder tracks the first returning pulse for a water level greater than one inch, it is necessary to disconnect one lead at a time to determine separate SDV water levels. If water is detected, a level two QC inspector must be called to verify the readings.

2.2 Operating Procedures

Procedures have been written for the control room operator to respond to the presence of water in the SDV as detected by the UT system. If the water level reading in the SDV is less than 1-1/2 inches, procedures call for an operator to: (1) visually verify that the SDV vent and drain valves are open; (2) check for leaks in the scram discharge valves by observing CRD temperature probe outputs and by touching the HCU discharge risers; and (3) request QA verification of the UT reading.

If the water level reading is between 1-1/2 and 2 inches, procedures call for the control room operator to: (1) immediately request QA to dispatch a level two QC inspector to verify the reading; and (2) unless the QC inspector determines that the water level is less than 1-1/2 inches, begin an orderly shutdown within one hour.

If water level exceeds 2 inches, procedures call for the control room operator to immediately begin an ord ; shutdown without verifying the UT reading.

Plant personnel estimate that a level two QC inspector can reach the area of the UT device within approximately three minutes of being notified. At least one level two QC inspector is available for this duty at Browns Ferry during each shift.

2.3 Other Leakage Detection Capabilities

Leakage of water through a scram outlet valve to the SDV is recognized as one of the ways for water to reach the SDV. This leakage may be detectable by means other than the UT system or SIV instrumentation.

Flow out of a scram outlet valve would change the flow of the CRD cooling water through the CRD seals so as to increase the water temperature at the location of the CRD temperature probe. At this time, no good data is available to correlate CRD temperature with scram outlet valve leakage. In particular,

the rate of temperature change for a given leakage is unknown. However, for the leakage rates postulated in this section, it is reasonable to assume that the temperature probe alarm set point would be reached within a relatively short time; on the order of a few minutes. Although the CRD temperature probe alarms in the control room, not all 185 CRD temperature probes are read simultaneously. A sequential scan is used and it is estimated by GE that the cycle time to read all temperature probes is approximately six minutes.

Another means of inferring scram outlet valve leakage is the observation of control rod drift. Leakage past the scram valve in excess of the CRD seal leakage would cause the associated control rod to begin to drift into the core. A rod drift alarm is available in the control room.

Another indication of scram outlet valve leakage is movement of the scram outlet valve stem sufficient to actuate the scram valve position indication switches. This requires a stem movement of approximately 1/32", out of a total valve stroke of approximately 3/16". Actuation of the stem mounted switch will light the associated scram outlet valve position indication light in the control room.

One means by which the scram outlet valves can open sufficiently to leak is degraded air pressure in the HCU air header. A low pressure alarm is provided to alert the operator at approximately 70 psia. The actual pressure reading on the HCU header is available locally in the area of the HCUs.

2.4 Procedures for Loss of Control Air

The procedures at Browns Ferry for loss of control air were modified in response to IE Bulletin 80-17, Supplement 3, to protect against scram valve leakage on gradual loss of control air. The details of the concern for loss of control air are discussed in the AEOD memorandum of August 18, 1980.[2]

At Browns Ferry, control room indication of the air pressure in the HCU air header is limited to a single alarm with a setpoint at 70 psia. A local air

pressure guage is available at the HCUs. Normally air pressure is maintained between 70 and 75 psia. Since only a slight degradation of air pressure initiates the alarm, the licensee considers it undesirable to initiate a scram based on this alarm alone. Upon receipt of the 70 psia alarm, procedures call for the control room operator to dispatch an auxiliary unit operator to read the local air pressure gauge. Plant operations personnel at Browns Ferry have told the NRC resident inspector that an operator will be dispatched to read the local pressure guage no later than 2 minutes after receipt of the 70 psia alarm. If air pressure in the HCU air header is found to be 60 psia or less, the auxiliary operator informs the control room operator. The control room operator then initiates a manual scram. Communication between the auxiliary and control room operators is maintained via walkie-talkie.

In addition to the above procedure for gradual loss of air to the HCU air header, other procedures have been implemented in response to IE Bulletin 80-17, Supplement 3. These call for manual scram initiation in the event of: (1) multiple rod drift-in alarms; or (2) a marked change in the number of control rods with high temperature probe alarms.

2.5 Procedures for Standby Liquid Control Initiation

Bulletin 80-17, Supplement 1, requested that operating procedures be revised to provide clear guidance to the control room operator regarding initiation of the standby liquid control system (SBLC) following a failure of control rods to fully insert.

At Browns Ferry, mandatory SBLC system actuation is required by operating procedure if either of the following conditions exist: (1) five or more adjacent rods are not inserted below 06 position <u>and</u> either reactor water level cannot be maintained or suppression pool water temperature limit of 110°F is reached; or (2) thirty or more rods are not inserted below 06 position <u>and</u> either reactor water level cannot be maintained or suppression pool water temperature of 110°F is reached.

3. ANALYSIS AND EVALUATION

For purposes of analysis and evaluation of the Browns Ferry failure to scram event, an effort was made by AEOD to identify water sources that could fill the SDV. The AEOD report of August 1980, identified the following sources of water: 1) water left from a previous scram; 2) purge line inflow; 3) water or steam backing up from the clean radwaste drain system; and 4) inflows through the scram discharge valves. It is recognized that this list of water sources may not be complete. However, at this time, neither operating experience nor design review has revealed any other sources.

As discussed in Section 2 of this report, the current capability at Browns Ferry to detect and respond to accumulation of water in the SDV is based on the following elements: previously installed SIV instrumentation; recently installed UT instrumentation; other previously installed instrumentation such as CRD temperature probes, CRD drift alarms, control air flow pressure alarm, etc; and operating procedures for response to the aforementioned instrumentation. The response of the interim system at Browns Ferry is dependent on both the instrument capability and the operator response. Both aspects are addressed in this analysis and evaluation.

The original design, as understood prior to analysis of the Browns Ferry event, was thought to have provided continuous, redundant, safety grade and automatic protection which was functional for all water sources. It was thought to also fail safe on loss of HCU control air pressure. However, the analysis of the Browns Ferry event showed that this system did not work for all situations of water accumulation. Accordingly, the original system was supplemented with a functional UT system. This interim system is neither continuous, redundant, safety grade, nor automatic for many cases. Furthermore, its capability may be inadequate for a loss of control air pressure. That is, the interim system does not provide the same level of protection as was perceived of the original design.

At this point, a short discussion of the capability and reliability of the UT system will be presented. As stated in Section 2, the UT system includes a CRT display of the return echo. This is shown in Figure 4. As stated earlier, an echo is received from the i.d. of the pipe and is displayed on the CRT as the left most peak. The presence of this so-called "reference pulse" is interpreted by the inspector performing the calibration as verification of the operability of the transducer which is being monitored. The calibration technique also requires that the UT system be connected to a separate transducer to detect a known depth of water in a standard pipe. These surveillance and calibration procedures are performed once each shift by a level two QC inspector who has extensive training in UT techniques. We believe that because of the presence of the "reference pulse" and the high level of training of the level two QC personnel who performed the surveillance and calibration of this instrument, any degradation of the operation of the UT system due to heat, vibration, radiation or other failure mode would be discovered during the scheduled surveillance. It is recognized that during the period of 8 hours between surveillance of the UT system, it would be possible for equipment failure to go undetected. However, because of the unlikelihood of a rapid water inflow with an accompanying need to scram occurring during the same period of the equipment failure (except for the degraded air case), this surveillance interval is judged to be adequate.

The following analysis and evaluation addresses the capability of the interim system to detect and respond to water from various postulated sources.

3.1 Water From the Previous Scram

Water left from the previous scram after scram reset will be detected by the ultrasonic system. Because of the time required between a scram and startup operations, a number of ultrasonic readings and equipment calibrations would normally take place during the shutdown. For this situation, rapid detection is not required and no immediate action is needed if water is detected. Startup would simply be delayed until it was assured that the situation was corrected and no water remained in the SDV.

3.2 Purge Line Inflow

Purge line inflows would be detected in time depending on the rate of inflow and the time of purging. Purging of the SDV is an operation which is performed when the reactor is shutdown in order to reduce accumulations of radioactivity in the SDV and its associated piping. The ultrasonic system would be used to check for the presence of water prior to startup. Enough time would be available during the plant shutdown for the operator action required to detect and remove all water from the SDV prior to startup.

An administrative or operator error, which allowed purging during normal operation, could provide a flow rate of water into the SDV which might not be detected soon enough by either the interim system or the original system. However, the likelihood of a properly informed operator performing this unprecedented action is remote.

3.3 Water or Steam From the Drain System

Water backing up from the clean radwaste (CRW) drain system would most probably be detected by the SIV instrumentation of the original design. An exception to this would be a SIV drain line blockage which could prevent flow from the CRW drain system into the SIV but would allow flow up through the SDV vent lines to fill the SDV. However, at Browns Ferry, positive vent paths to atmosphere have been provided on the SDV vents. Any water backing up in the vent line would be released via this path rather than to fill the SDV unless the backflow rate was high due to a large water release to the drain system. Operating experience at Browns Ferry to date has shown that water backing up from the CRW drain system has actuated level switches in the SIV.

Low pressure steam backing up from the CRW drain system due to flashing hot water in the drain pipe would not be detected by the SIV instrumentation. If the drain line between the SDV and SIV became plugged by the slow drain of condensed vapor mixed with rust, the steam backing up through the SDV vent lines would slowly fill the SDV with condensed vapor. The positive vent paths

to atmosphere would vent a portion of the low pressure steam but would not prevent the SDV from filling with condensate. Operating experience at Browns Ferry Unit 1 has shown that flashing hot water can appear in the drain pipe from valve leakoff connections or other sources. Therefore, the ultrasonic system must be used to monitor this situation and the surveillance interval must be short enough to assure timely discovery during a large hot water release.

Thus, with respect to the three identified sources of water listed above, we believe that the UT system provides adequate interim assurance that water can be detected and actions taken before the plant reaches a condition where the SDV is filled and a scram is required. It appears that this would be true with a surveillance interval longer than the 30 minutes currently used at Browns Ferry provided that surveillance was performed prior to any start-up. However, if protection for currently unidentified water sources and flow rates is to be provided, continuous monitoring of the UT system in the control room would be preferable to the current 30 minute surveillance interval. This would allow for a more rapid control room operator response. As an intermediate method (between continuous monitoring and lengthy surveillance intervals) for providing response to unidentified water sources and rates, an alarm output of the UT device could be provided in the control room. This would allow more timely operator response without the complexity of locating the complete UT system output in the control room. Upon receipt of the sustained alarm, an auxiliary operator could be dispatched to the UT readout located by the HCUs. We believe this approach would also provide adequate protection for unidentified water sources and flow rates.

3.4 Single Scram Outlet Valve Leakage

To evaluate the adequacy of the interim system for various leak rates from the scram outlet valves, it is necessary to identify the causes of leakage. First, it should be noted that scram valve leakage during normal operation is quite low. At Browns Ferry Unit 3, tests following the June 28, 1980 event indicated an aggregate leakage of from 0 to 3 gallons per hour.

Discussions with GE on the leakage characteristics of the scram outlet valves indicate that any leakage is likely to cause degradation of the valve seats and could lead fairly rapidly to greater leakage. Rapid deterioration of the seating surface of one valve would result in cbvious problems with the associated control rod drive but would not affect others.

One aspect of the scram outlet valve leakage problem that must be addressed is the difference in character between a leak arising from a single valve failure and that which could arise from a common mode failure leakage of many valves. With respect to a single valve failure, the maximum inflow of water into the SDV is limited by leakage past the CRD seals. GE has estimated that with the CRD seals completely destroyed, a leak rate of 10 to 12 gpm into the SDV is the maximum that could occur. This is the rate if the flow is completely unrestricted by the scram outlet valve. If the scram valve is only partially open or leaking, the flow rate would be less. If the CRD seals are intact, leakage would be expected to be in the range of 1 to 5 gpm. Thus, a single failure of a scram valve results in only a limited flow into the SDV which would drain out with no accumulation for the current SDV drainage characteristics.[1]

Indications of scram valve leakage would be available to the operator. The CRD temperature probe alarm would be actuated. If the scram outlet valve leakage is greater than the corresponding CRD seal leakage for a pressure differential across the piston of approximately 550 psig, the rod would move into the core.

When assessing the probability of an event that could cause problems for the SDV, it must be recognized that the probability of a simultaneous failure of more than one scram outlet valve at a given time is very low. Multiple valve failures would have to occur simultaneously before the drainage capabilities of the current system would be challanged. Because of (1) the low probability of this event, (2) the likelihood of early detection by rod drift alarm or CRD temperature probe alarm, and (3) because the event does not cause an accompanying plant disturbance, this postulated event is not considered to be a serious concern for the interim period. The interim precautions should be adequate to protect against failures of this type.

3.5 Multiple Scram Outlet Valve Leakage from a Common Cause

Multiple scram outlet valve leakage due to a common cause can raise serious concerns about the ability to scram the plant successfully. To date, the only plausible common cause which leads to substantial leakage of a large number of scram outlet valves is degraded air pressure in the control air header for the HCUs. Loss of air pressure in the control air header has occurred due to a variety of reasons such as failure of an air compressor, improper valve alignment, clogging of filters and dryers, and severance of an air line.

As air pressure in the header decays, the scram outlet valves, which are held closed by air pressure, begin to open. Although the exact pressure at which a given valve begins to open depends on manufacturing tolerances, the pressure for a group of valves is in the range of about 40 to 45 psia. Information from GE indicates that a leakage flow of from 1 to 5 gpm out of a scram outlet valve for a given drive could occur without producing rod motion. The actual value for a given drive would depend on the condition of the seals in that particular drive. GE has stated that for a typical reactor, if the scram discharge valve flow rate to produce rod motion for each individual CRD was averaged with the scram discharge valve flow rate to produce rod motion of all other CRDs, the average would be in the range of 2 to 3 gpm.

With this information it can be postulated that a degraded control air pressure condition could exist for which leakage from a large number of scram outlet valves could exist without producing a scram, In fact, depending on the number of scram valves which partially open and the leakage rate of these valves, it would be possible to generate a significant flow of water into the SDV without producing significant rod motion. It is recognized that the possibility of the actual occurrence of high flow rates without rod insertion depends on three factors: (1) the control air pressure degradation pattern, (2) the range of air pressure over which the scram outlet valves open, and (3) the seal leakage rate of the CRD associated with each particular scram outlet valve. However, with the data given above, a flow rate in the range of 1 to

2 gpm per drive without significant rod insertion could be possible for certain degraded air pressure scenerios.

3.6 Degraded Control Air

Assuming an average leak rate that could be generated without significant rod motion (given a specific degraded air pressure) of 2 gpm per CRD, a total of 2 x 185 or 370 gpm flow into the SDV would occur. Although this large flow rate appears feasible within the characteristics of the system, lower rates of leakage to the SDV could also be generated by the same mechanism, and indeed are more probable. These are discussed below in a framework of average steady-state flow rates. It is recognized that an actual air system failure would likely lead to continuously changing leakage rates, but the air pressure degradation might level off and thereby stabilize the leakage rate at any point.

For purposes of evaluation, inflow rates into the SDV can be separated into those for the East SDV header and those for the West SDV header. Test data show that for Browns Ferry Unit 3 the average drain rate of the East SDV header is normally about 12 gpm with its vent and drain valves open. Thus, any steady state in-leakage of less than 12 gpm would not result in water accumulation in the East header unless the East side drain line were blocked. Similarly, test data show that the average drain rate of the West SDV header, with its vent and drain valves open is normally about 24 gpm. Thus, any steady-state in-leakage of less than 24 gpm would not result in water accumulation in the West header unless the West side drain line were blocked.

Test data also show that the average drain rate of the SIV, with the vent and drain lines and valves functioning normally, is about 35 gpm. To a first approximation, from the above test data and assumptions, the following general statements can be made:

1) For a steady-state in-leakage below approximately 12 gpm per side, no water accumulation would occur, no water measurement with UT is required, and no operator action is necessary.

2) For a steady-state in-leakage between approximately 12 and 24 gpm per side, water would accumulate on the East side. As an example, for a steady state flow rate of 24 gpm into each header, the West side would remain empty and the East side would fill within approximately 25 minutes. The current 30-minute surveillance interval at Browns Ferry using the UT system might not detect this accumulation before filling of the East side. Also, because the SIV drain rate is greater than the inflow rate from both the SDV sides to the SIV, the 50 gallons scram level switch would probably not activate. However, the 3 gallon and perhaps the 25 gallon level switches might be activated. This level of inleakage could result in a scenario similar to the Browns Ferry event where the West side rods scrammed successfully but the East side rods did not.

3) For a steady-state in-leakage above approximately 24 gpm per side, water would accumulate in both the East and West side SDVs. As an example, for a steady state flow rate of 36 gpm into each header, the East header would fill within approximately 12-1/2 minutes and the West header within approximately 25 minutes. The current 30-minute surveillance interval at Browns Ferry using the UT system might not detect this accumulation before filling both the East and West sides. For this case, the SIV 50 gallon level switches would probably activate somewhere between 12-1/2 and 25 minutes and initiate an automatic scram. However, the scram capability would be limited on both the East side and the West side due to the previous water accumulation.

4) For in-leakage at very high rates (approaching 150 gpm per side) water would accumulate in both the East and West side SDVs. Each side would fill within 3 minutes and probably before sufficient water could flow to the SIV to activate the automatic scram switches at the 50 gallon level. The current 30-minute surveillance interval at Browns Ferry using UT would not detect this accumulation before a probable loss of scram capability. Proper operator action would probably be required within less than 2 minutes following the initiation of this scram valve leakage rate to avoid reaching a point where it would become impossible to scram.

In summery, the above analysis adresses conditions of degraded pressure in the HCU control air header which can lead to aggregate leakage rates to the SDV in the range of 24 to 300 gpm. Flow rates at the high end range probably produce at least some rod motion and perhaps some rods might fully insert. However, at this time there is no assurance either by analysis or testing that a range of leakage rates does not exist which could fill the SDV quickly with insufficient indication to the operator or time for manual scram before tha ability to scram is lost.

The above discussion of the scram system behavior is for different but constant flow rates. This would probably not be the case for an actual degraded control air event. The scram valve flow rate would likely pass through the different regimes as discussed above and the characteristics of a particular flow rate would apply at that time. However, analysis or test results for a variable flow rate, which show acceptable system behavior, do not exist at this time. Thus, inadequate basis is available to justify disregarding these concerns.

A degraded air supply can also affect the performance of the SDV vent and the SIV drain valves. Tests done at Browns Ferry show that the SDV vent and the SIV drain valves begin to close at a control air system pressure of about 17 psig. Thus, the drain and vent valves will remain open during the type of degraded air condition that might lead to loss of scram capability.

It should be noted that the time available for operator action to respond to a degraded air condition can be separated conceptually into two phases: (1) time available before air pressure degrades from the normal alarm set-point of 70 psig to the pressure at which scram discharge valves leakage begins (about 45 psig) and (2) time available following the beginning of scram discharge valve leakage to the time where the SDV fills to the point where a scram is no longer possible. The analysis shows that the time available for operator action following the beginning of scram discharge valve leakage can be as little as 2 minutes. Because of the

short time available for operator action following initiation of scram discharge valve leakage, operator action should be taken prior to reaching a degraded or lost scram capability. Because HCU control air degradation preceeds opening of the scram discharge valves, added time would be available if operator action were based on air pressure indications. For a rapid air pressure degradation which stabalized at a point where large scram discharge valve leakage occurred, the benefits of operator action based on air pressure indication would be diminished. From the standpoint of improved assurance of a successful scram during a degraded HCU control air event, however it is preferrable to scram on the indications of degraded air pressure than on the UT system. This would be true even if UT readout were continuous in the control room. The UT system (on indication of water in the scram discharge volume) to initiate a manual scram for the degraded HCU control air event does not provide sufficient assurance that adequate time will be available for the required operator diagnosis and action. The same can be said for reliance on CRD temperature probes, rod drift alarms, and scram outlet valve indicator lights.

IE Supplement 3 to Bulletin 80-17 requires an immediate manual scram on low HCU control in pressure at a minimum pressure of 10 psi above the opening pressure of the scram outlet valves. This provides additional time for operator diagnosis and action prior to possible filling of the SDV following receipt of the low pressure alarm. However, because of the lack of any control room indication of HCU control air pressure (except the low pressure alarm at 70 psig) current procedures at Browns Ferry require an operator to be sent to the HCUs to read a local HCU air pressure gauge. This local operator then reports back the local reading to the control room by walkie talkie. Given the rapidity of the water inflow possible with the degraded air pressure condition, we judge this arrangement to be inadequate. We believe that the rapid operator response required by a degraded air system condition ncessitates that adequate alarms and instrumentation be available in the control room. However, because the present alarm is not safety grade and is a single channel, we

believe that reliance on the alarm alone to initiate a manual scram is not adequate. Short of installing safety-grade instrumentation for this function, we believe that adequate instrumentation could be provided by redundant pressure indication in the control room along with a distinctive alarm on degraded air pressure. Furthermore, since the instrumentation is not qualified to function during certain postulated events (e.g. earthquakes), procedures which require immediate manual scrams for such events should be considered.

It is our judgment that if the upgraded instrumentation and procedural changes discussed above are provided, then the system will be adequate to respond to degraded HCU control air for the interim period.

We believe that this analysis supports the position that a scram on degraded HCU control air is sufficient to respond to the complete range of aggregate leakage rates arising from degraded HCV control air pressure as enumerated earlier in this section.

This judgement is based on (1) the additional time available before any discharge valve leakage begins and (2) the relatively low probability of a rapid air pressure degradation which stabilizes in the range of serious scram outlet valve leakage.

3.7 Operating Experience With Degraded Control Air Supply

An effort was made to look at reactor operating experience relative to scrams caused by the consequences of a degraded control air supply. By looking through the Annual Report on Nuclear Power Plant Operating Experience[3-6] for the years 1974-78, a total of 21 events were found for BWRs where the description of the event mentioned a loss of control air as the initiating event leading to the scram. The dates of these events are listed in Table 1.

Because of the brevity of the descriptions and the lack of records available to make a more careful study of each event, it is probable that not all of the

events describe a loss of control air which would or could affect the HCUs. On the other hand, some of the events seemed to be very close descriptively to the type of rod behavior that would be expected given a loss of control air to the HCUs. For example, one event description mentioned massive rod drift.

Another event generated an automatic scram due to high level in the SIV. This event occurred at Browns Ferry Unit 1 on November 24, 1976. Because of the known drain characteristics at the Browns Ferry units, it is likely that during this event the SDV was at least partially filled. Because the SDV is designed to provide approximately 3.3 gallons per drive free volume, and a typical scram requires less than one gallon per drive, enough volume was available for a successful scram. However, there is no doubt that the volume margin was reduced. An air degradation of a slightly different character could have lead to a water filled SDV and inability to scram.

An effort was made by AEOD to find data that would indicate filling of the SDV during some of these events. From the event at Browns Ferry 3 on June 28,1980, one bit of evidence that leads to the conclusion that water was in the SDV prior to the first attempted scram was that the SIV high level scram switches were activated more quickly than expected during a manual scram (18 seconds vs. 45 seconds). This is because for a full SDV, water entering the SDV during the scram will more quickly pressurize the SDV and force water through the drain line to the SIV than if the SDV were not pressurized. For events where an event recorder output was available, no such change was noted. However, most events had no data from an event recorder available and no other way of recalling this data.

In general, this search for operating experience data was unsuccessful. On the other hand, the argument that because a plant has successfully scrammed 21 times during degraded control air events does not provide a large statistical basis on which to judge the adequacy of the scram system (machine and man) for responding to such events. From the observation of 20 successful scrams in 20 scram attempts, one can conclude that the 95% upper (one sided) confidence limit for the probability of a scram failure is approximately 3/20 = .15.

Alternatively, the 95% lower (one sided) confidence limit for the probability of successful scram is approximately 1 - 3/20 = .85. Both confidence limits are computed on the assumption of a common probability of a scram attempt failing and the assumption of statistical independence of scram attempts. These assumptions have been made for mathematical convenience; they are not necessarily plausible. In fact, there is no doubt that the list of successful scrams includes some events where the HCU control air header system was not affected. These would not be included in a list developed through a closer investigation of the event which would disclose that fact. A lower number of successful scrams, due to legitimate control air degradation, even if all are successes, only detracts from the merits of the argument which claims that since no scram failures have occurred to date, the system is adequate.

4. FINDINGS

Based on the system description and evaluation discussed in Sections 2. and 3. of this report, a number of findings have been determined. Again, it should be emphasized that these are based on Browns Ferry only.

- The present system (ultrsonic level instrumentation, existing SIV instrumentation, and special operating procedures, etc.) should be capable of providing adequate protection during the interim against filling of the SDV due to all identified water sources except for those related to scram discharge valve leakage due to degraded HCU control air pressure.

- Degraded HCU control air pressure could result in scram outlet valve leakage to the SDV which would require operator action to manually scram the plant within a few minutes before scram capability would be completely lost. This event would likely be accompanied by a plant disturbance requiring a scram due to other control air related disruptions in the plant. Such an event would be accompanied by numerous control room alarms and indications which could distract the operator from a prompt manual scram actuation.

- Operating experience indicates that a significant number of reactor scrams attributed to loss of HCU control air pressure have occurred. These provide evidence that rapid filling of the SDV is a credible event.

5. RECOMMENDATIONS

The principal recommendations of the study are as follows:

- An immediate manual scram should be required based on control room indication of degraded HCU control air pressure. Review of licensee proposals should include consideration of the available pressure indications and procedures to assure that other alarms and indications do not divert operator attention from this priority action.

- Redundant HCU air header pressure instrumentation should be provided in the control room. A distinctive alarm for degraded air pressure should be provided to aid the operator in quickly focusing his attention on the need for protective action.

- Because of the possibility that a currently unidentified water source could result in water accumulation in the SDV, it would be prudent to monitor the ultrasonic system alarm output in the control room and require an immediate verification of a sustained alarm by operator dispatch to the equipment. Operability and calibration checks of the system should be continued on a schedule of once per shift.

6. CONCLUSIONS

AEOD has reviewed the interim surveillance system at Browns Ferry used to detect the presence of water in the SDV. The AEOD assessment considers the procedures and equipment changes initiated in response to IE Bulletin 80-17 with Supplements, 1, 2, and 3 to be adequate for continued interim operation of the Browns Ferry Nuclear Plant, if the recommendations of this report for response to degraded control air pressure are implemented.

As of the date of this report, the instrumentation and procedures in place to respond to the loss of control air scenario at Browns Ferry are judged to be inadequate. For this event the operator must respond promptly to a single in-distinctive alarm for loss of control air pressure during a period when numerous alarms may be occurring. Additionally, the operator must take actions outside the control room in a very limited time frame because he lacks a pressure readout in the control room. IE is currently taking steps to upgrade the procedure for response to the degraded control air pressure event.

In the past, operator action to perform a vital safety function within less than 10 minutes has not been considered acceptable by the NRC. However, providing the operator with both a distinctive low pressure alarm and reliable air pressure instrumentation in the control room, would help assure adequate operator response within the required time period. Such an arrangement should be acceptable for the interim. A dedicated operator with adequate alarms and instrumentation in the control room could provide even greater assurance of a timely manual scram. If the provisions made to accomplish a manual scram are found to be untimely or inadeqiate, provisions should be made for an automatic scram on low HCU control air pressure.

For the long-term, the scram system should be upgraded according to the recommendations of the AEOD report of July 30, 1980. However, the consequences of degraded air pressure in the HCU air headers were not fully recognized at the time of that report and were not directly addressed. Although the recommended scram system modifications may be sufficient to enable the scram system to

respond to rapid inflows of water from the scram outlet valves due to degraded HCU air header pressure, design review of the long-term modifications should include specific consideration of the effects of degraded air pressure.

REFERENCES

1. AEOD MEMO (Michelson) to NRR (Denton) dated August 1, 1980 with enclosures.

2. AEOD MEMO (Michelson) to NRR (Denton) dated August 18, 1980.

3. USNRC NUREG-0227 dated April 1977.

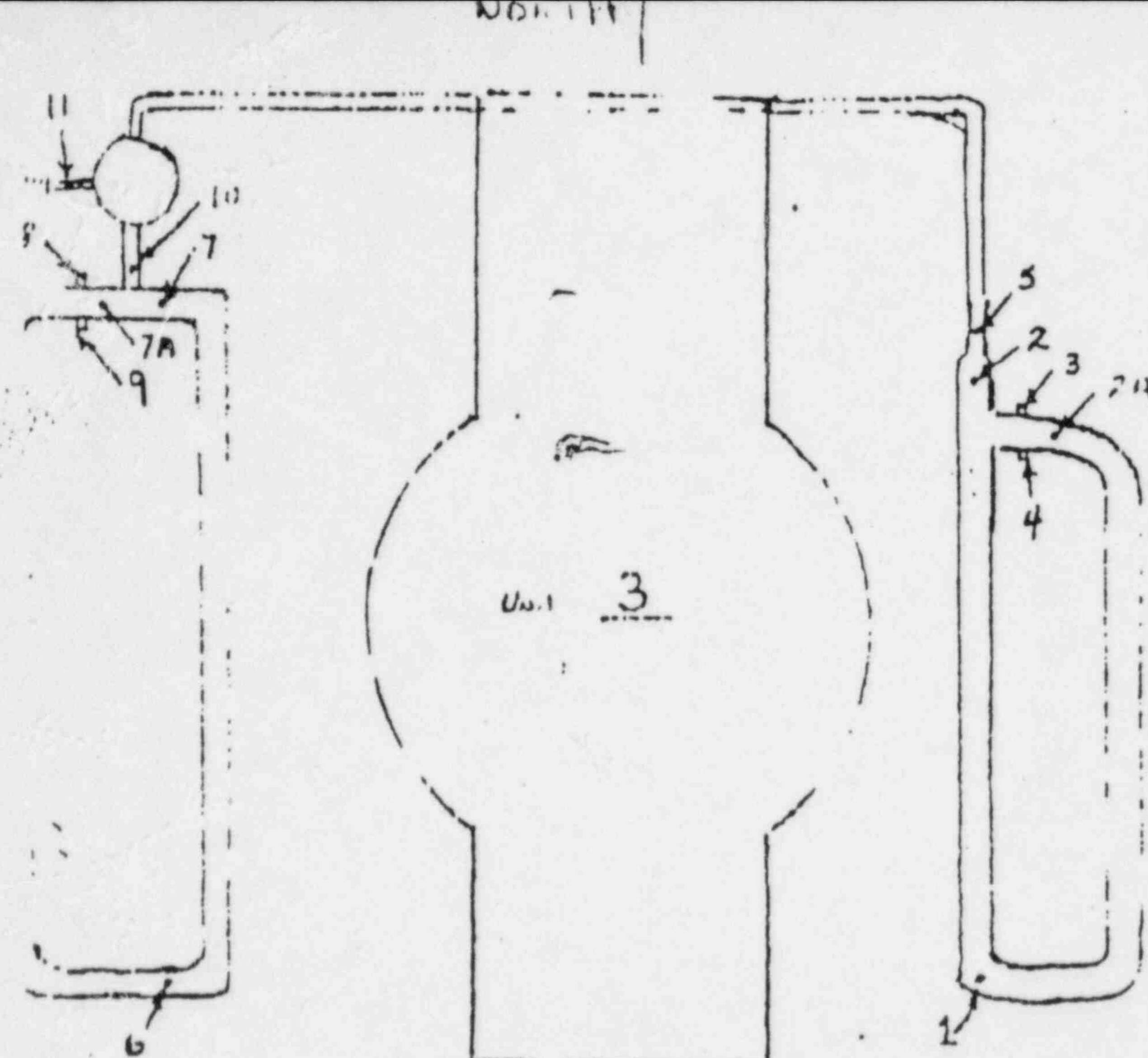
4. USNRC NUREG-0366 dated December 1977.

5. USNRC NUREG-0483 dated February 1979.

6. USNRC NUREG-0618 dated December 1979.

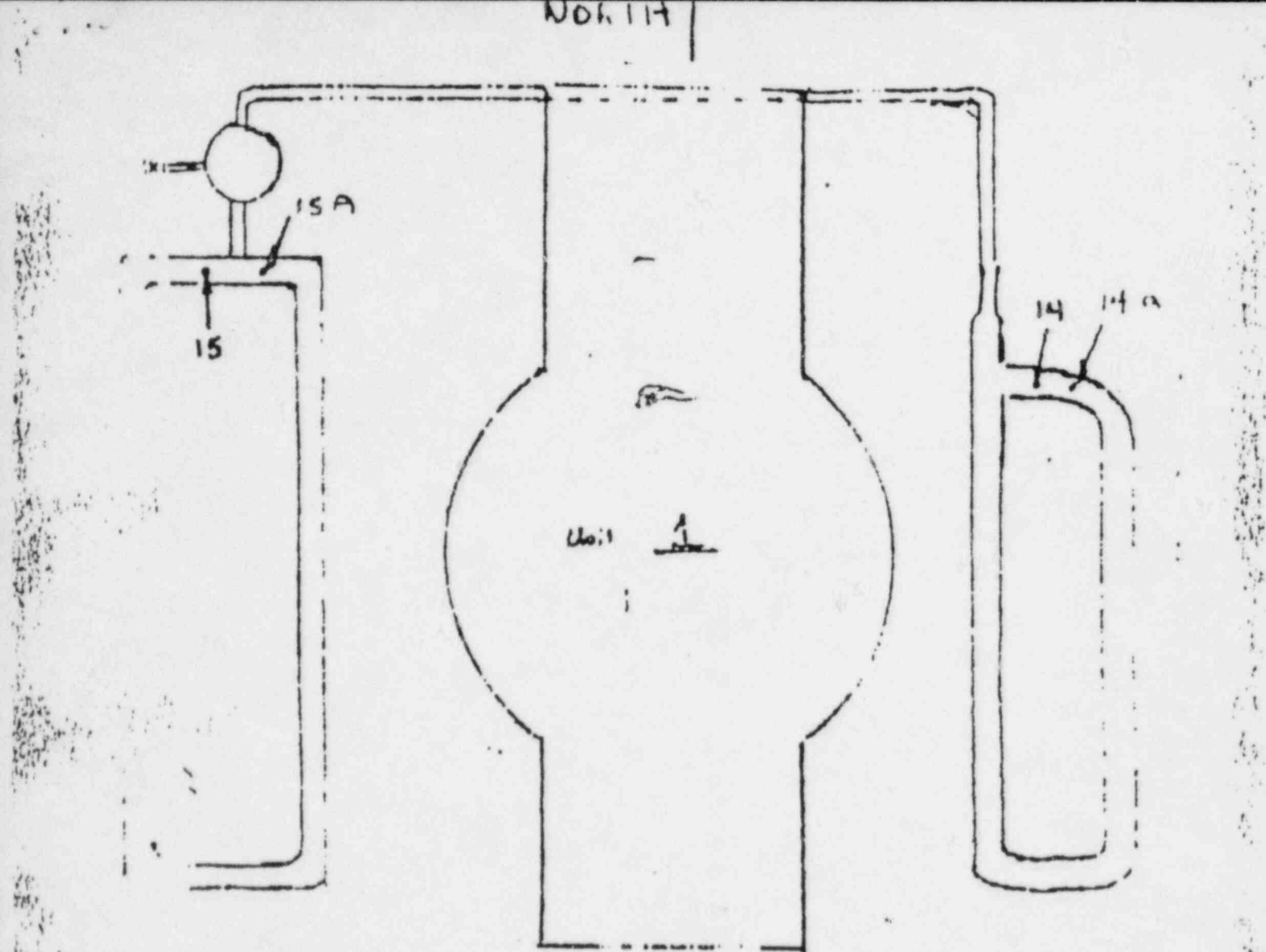
Table 1 Scrams Attributed to Loss of Air (1974-1978)

Browns Ferry 1:	8/1/74, 10/19/76, 11/24/76, 8/15/78, 8/18/78
Browns Ferry 2:	8/18/78
Brunswick 2:	4/5/77
Dresden 2:	9/7/77, 7/28/78
Dresden 3:	8/15/74
Duane Arnold:	1/9/78
Hatch 1:	3/4/76
Millstone 1:	8/6/77. 5/29/78
Nine Mile Point 1:	12/21/74
Pilgrim:	1/19/76
Quad Cities 1:	1/3/77, 4/30/78
Quad Cities 2:	7/1/74, 8/31/74, 10/25/77



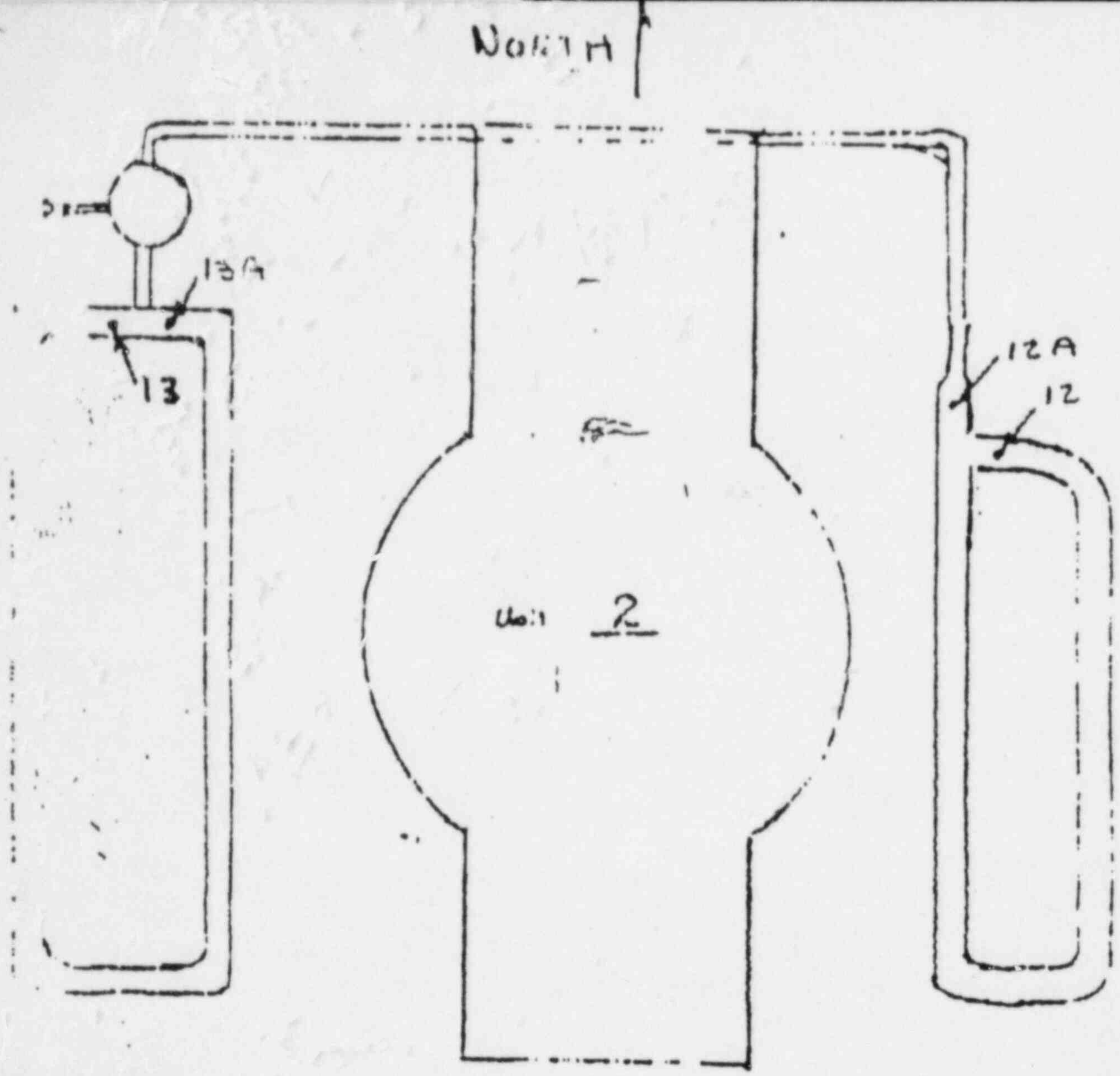
TRANSDUCER #	LOCATION
1	VERTICAL RUN OF 1" VENT PIPE
2	6" HEADER (CONTINUOUS MONITORING)
2A	6" HEADER SPACE (NO CABLE RUN)
3 & 4	6" HEADER THRU-TRANSMISSION-NORTH...
5	2" DRAIN
6	HORIZONTAL RUN OF 1" VENT PIPE
7	6" HEADER (CONTINUOUS MONITORING)
7A	6" HEADER SPACE (NO CABLE RUN)
8 & 9	6" HEADER THRU-TRANSMISSION-NORTH...
10	2" DRAIN
11	2" VOLUME TANK DRAIN

Figure 1 Unit 3 Transducer Location



TRANSDUCER #	LOCATION
14	6" HEADER (CONTINUOUS MONITORIN...
14A	6" HEADER SPARE (NO CABLE RUN...
15	6" HEADER (CONTINUOUS MONITORI...
15A	6" HEADER SPARE (NO CABLE R...

Figure 2 Unit 1 Transducer Location



TRANSDUCER #	LOCATION
12	6" HEADER (CONTINUOUS
12A	6" HEADER SPARE (N
13	6" HEADER (CONTIN
13A	6" HEADER SPARE (

Figure 3 Unit 2 Transducer Location

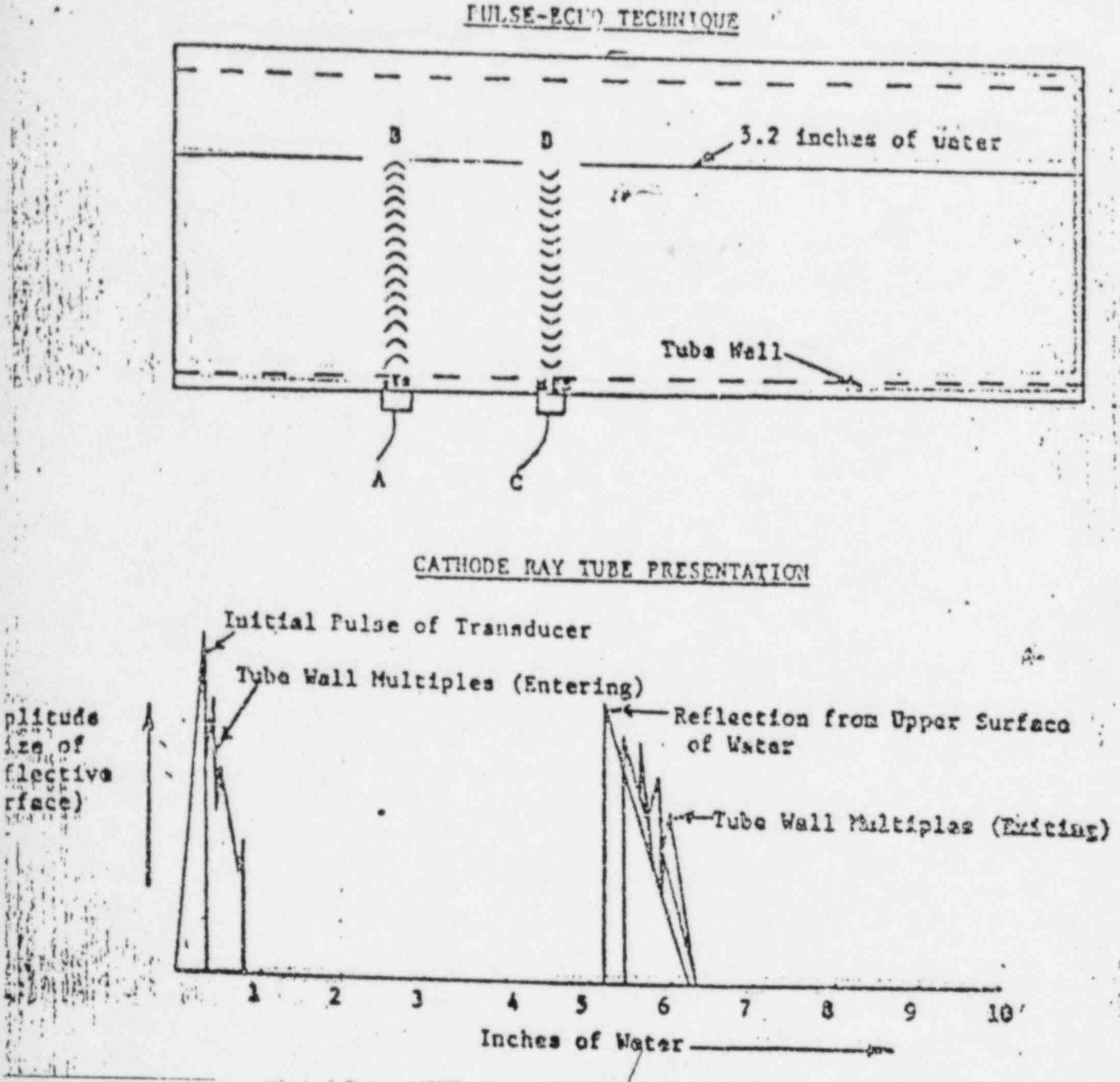


Figure 4 Pulse Echo CRT Display

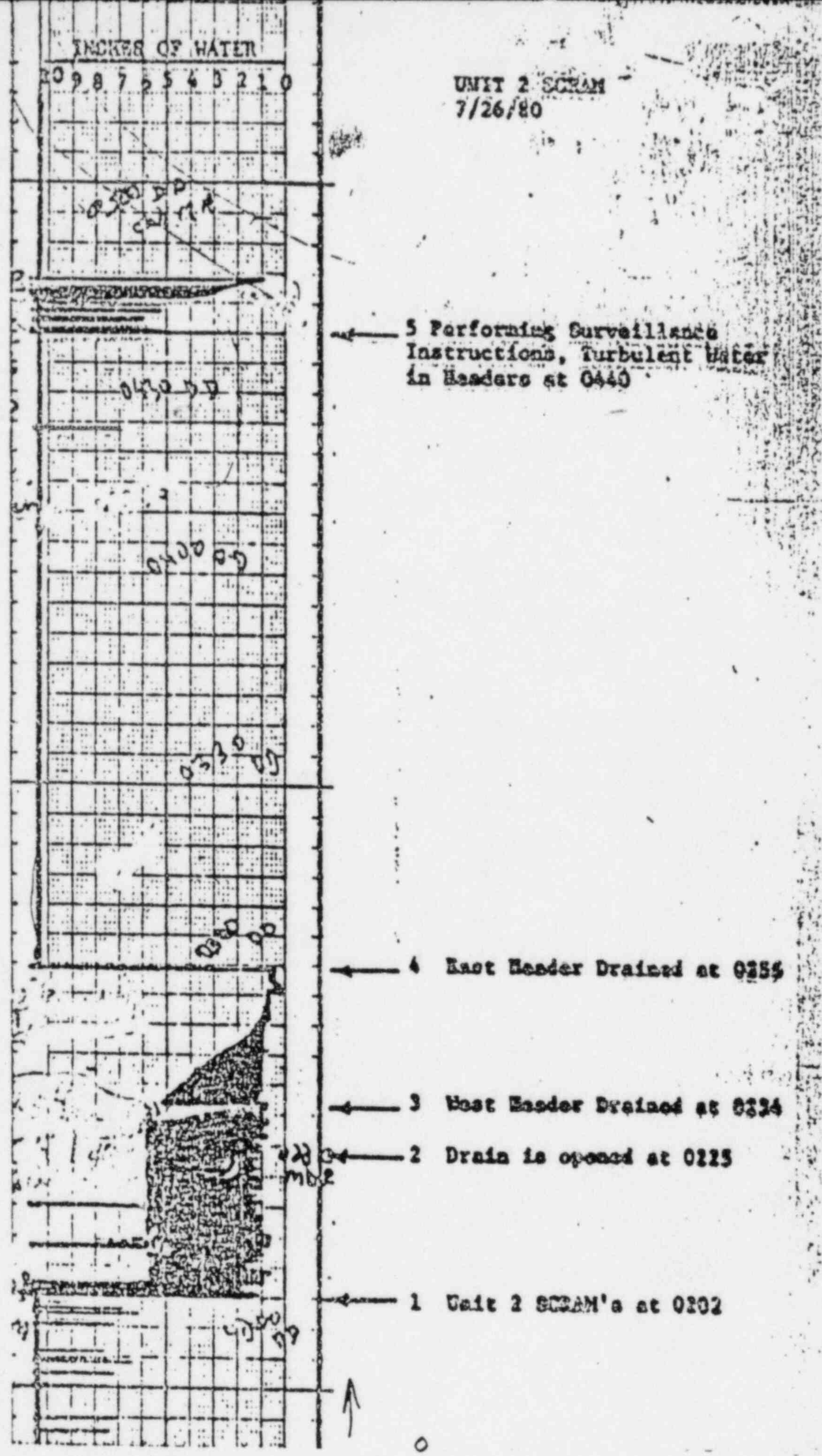


Figure 5 Strip Chart Recorder Output

CLAIM FOR DAMAGE, INJURY, OR DEATH

INSTRUCTIONS: Prepare in ink or typewriter. Please read carefully the instructions on the reverse side and supply information requested on both sides of this form. Use additional sheet(s) if necessary.

OMB NO 43-R0357

1. SUBMIT TO
General Counsel
Nuclear Regulatory Commission
Washington, D.C. 20555

2. NAME AND ADDRESS OF CLAIMANT (Number, street, city, State and Zip Code)
General Public Utilities Corp.
100 Interpace Parkway
Parsippany, N.J. 07054
et. al (see attachment)

3. TYPE OF EMPLOYMENT (MILITARY / CIVILIAN)	4. AGE	5. MARITAL STATUS	6. NAME AND ADDRESS OF SPOUSE, IF ANY (Number, street, city, State, and Zip Code)
NA	NA	NA	NA

7. PLACE OF ACCIDENT (Give city or town and State; if outside city limits, indicate mileage or distance to nearest city or town)	8. DATE AND DAY OF ACCIDENT	9. TIME (A.M OR P.M.)
Three Mile Island Unit No. 2 Londonderry Township, Pennsylvania	March 28, 1979 Wednesday	Beginning at 4:00

10. AMOUNT OF CLAIM (in dollars)

A. PROPERTY DAMAGE	B. PERSONAL INJURY	C. WRONGFUL DEATH	D. TOTAL
$4,010,000,000	NA	NA	$4,010,000,000

11. DESCRIPTION OF ACCIDENT (State below, in detail, all known facts and circumstances attending the damage, injury, or death, identifying persons and property involved and the cause thereof)

See Attachment

FOIA-85-240
A2

12. PROPERTY DAMAGE

NAME AND ADDRESS OF OWNER, IF OTHER THAN CLAIMANT (Number, street, city, State, and Zip Code)

same

BRIEFLY DESCRIBE KIND AND LOCATION OF PROPERTY AND NATURE AND EXTENT OF DAMAGE (See instructions on reverse side for method of substantiating claim)

See Attachment

13. PERSONAL INJURY

STATE NATURE AND EXTENT OF INJURY WHICH FORMS THE BASIS OF THIS CLAIM

NA

14. WITNESSES

NAME	ADDRESS (Number, street, city, State, and Zip Code)

Due to the nature of this claim based on the March 28, 1979 accident at TMI-2, there are hundreds of persons, including employees of the NRC, of the claimants and of third parties, including but not limited to the Babcock & Wilcox Co. and Toledo Edison Co. who are witnesses possessing relevant information. The NRC is already advised of the identity of many, if not all, of these witnesses as a result of its Special Inquiry "Three Mile Island -- A Report to the Commissioners and to the Public" (1980)

I CERTIFY THAT THE AMOUNT OF CLAIM COVERS ONLY DAMAGES AND INJURIES CAUSED BY THE ACCIDENT ABOVE AND AGREE TO ACCEPT SAID AMOUNT IN FULL SATISFACTION AND FINAL SETTLEMENT OF THIS CLAIM

15. SIGNATURE OF CLAIMANT (This signature should be used in all future correspondence)	16. DATE OF CLAIM
	December 8, 1980

CIVIL PENALTY FOR PRESENTING FRAUDULENT CLAIM	CRIMINAL PENALTY FOR PRESENTING FRAUDULENT CLAIM OR MAKING FALSE STATEMENTS
The claimant shall forfeit and pay to the United States the sum of $2,000 plus double the amount of damages sustained by the United States. (See R.S. §3490, 5438; 31 U.S.C. 231.)	Fine of not more than $10,000 or imprisonment for not more than 5 years or both. (See 62 Stat. 698, 749; 18 U.S.C. 287, 1001.)

STANDARD FORM 95
PRESCRIBED BY DEPT. OF JUSTICE
28 CFR 14.2

PRIVACY ACT NOTICE

This Notice is provided in accordance with the Privacy Act, 5 U.S.C. 552a(e)(3), and concerns the information requested in the letter to which this Notice is attached.

A. *Authority:* The requested information is solicited pursuant to one or more of the following: 5 U.S.C. 301, 28 U.S.C. 501 *et seq.*, 28 U.S.C. 2671 *et seq.*, 28 C.F.R. 14.3.

B. *Principal Purpose:* The information requested is to be used in evaluating claims.

C. *Routine Use:* See the Notices of Systems of Records for the agency to whom you are submitting this form for this information.

D. *Effect of Failure to Respond:* Disclosure is voluntary. However, failure to supply the requested information or to execute the form may render your claim "invalid".

INSTRUCTIONS

Complete all items—Insert the word NONE where applicable

Claims for damage to or for loss or destruction of property, or for personal injury, must be signed by the owner of the property damaged or lost or the injured person. If, by reason of death, other disability or for reasons deemed satisfactory by the Government, the foregoing requirement cannot be fulfilled, the claim may be filed by a duly authorized agent or other legal representative, provided evidence satisfactory to the Government is submitted with said claim establishing authority to act.

If claimant intends to file claim for both personal injury and property damage, claim for both must be shown in item 10 of this form. Separate claims for personal injury and property damage are not acceptable.

The amount claimed should be substantiated by competent evidence as follows:

(a) In support of claim for personal injury or death, the claimant should submit a written report by the attending physician, showing the nature and extent of injury, the nature and extent of treatment, the degree of permanent disability, if any, the prognosis, and the period of hospitalization, or incapacitation, attaching itemized bills for medical, hospital, or burial expenses actually incurred.

(b) In support of claims for damage to property which has been or can be economically repaired, the claimant should submit at least two itemized signed statements or estimates by reliable, disinterested concerns, or, if payment has been made, the itemized signed receipts evidencing payment.

(c) In support of claims for damage to property which is not economically reparable, or if the property is lost or destroyed, the claimant should submit statements as to the original cost of the property, the date of purchase, and the value of the property, both before and after the accident. Such statements should be by disinterested competent persons, preferably reputable dealers or officials familiar with the type of property damaged, or by two or more competitive bidders, and should be certified as being just and correct.

Any further instructions or information necessary in the preparation of your claim will be furnished, upon request, by the office indicated in item #1 on the reverse side.

(d) Failure to completely execute this form or to supply the requested material within two years from the date the allegations accrued may render your claim "invalid".

INSURANCE COVERAGE

In order that subrogation claims may be adjudicated, it is essential that the claimant provide the following information regarding the insurance coverage of his vehicle or property.

17. DO YOU CARRY ACCIDENT INSURANCE? ☒ YES, IF YES, GIVE NAME AND ADDRESS OF INSURANCE COMPANY (*Number, street, city, State, and Zip Code*) AND POLICY NUMBER. ☐ NO

(1) American Nuclear Insurers, 270 Farmington Ave., Farmington Ct. Policy No. 1353

(2) Kemper Insur. Co. Long Grove, Illinois Policy No. TA1084

18. HAVE YOU FILED CLAIM ON YOUR INSURANCE CARRIER IN THIS INSTANCE, AND IF SO, IS IT FULL COVERAGE OR DEDUCTIBLE?

Deductible

19. IF DEDUCTIBLE, STATE AMOUNT

ANI - $100,000.00
Kemper - $100,000.00

20. IF CLAIM HAS BEEN FILED WITH YOUR CARRIER, WHAT ACTION HAS YOUR INSURER TAKEN OR PROPOSES TO TAKE WITH REFERENCE TO YOUR CLAIM? (*It is necessary that you ascertain these facts*)

Payments to date (as of 12/2/80)

Met Ed	$96,256,228.00
JCP&L	$48,128,114.01
Penelec	$48,128,113.99

21. DO YOU CARRY PUBLIC LIABILITY AND PROPERTY DAMAGE INSURANCE? ☒ YES, IF YES, GIVE NAME AND ADDRESS OF INSURANCE CARRIER (*Number, street, city, State, and Zip Code*) ☐ NO Property damage - as above No. 20

Public Liability Insurance:
(1) Mutual Atomic Energy Liability Underwriters, 919 N. Michigan Ave., Chicago, Ill
(2) Nuclear Energy Liability Insur. Assoc. 270 Farmington Ave., Farmington, CT

BEFORE THE NUCLEAR REGULATORY COMMISSION
OF THE UNITED STATES

General Public Utilities Corporation, Jersey Central Power & Light Company, Metropolitan Edison Company and Pennsylvania Electric Company	:	CLAIM
v.	:	Docket No.
Nuclear Regulatory Commission.	:	

General Public Utilities Corporation ("GPU") and its operating subsidiaries, Jersey Central Power & Light Company ("JCP&L"), Metropolitan Edison Company ("Met-Ed") and Pennsylvania Electric Company ("Penelec"), bring this claim against the Nuclear Regulatory Commission ("NRC") alleging as follows:

1. This is a claim arising out of the March 28, 1979 accident at the nuclear electric generating facility known as Three Mile Island Unit No. 2 ("TMI-2"). Claimants seek damages against the NRC under the Federal Tort Claims Act, 28 U.S.C. § 2671 et seq.

Jurisdiction

2. Jurisdiction is based on the Federal Tort Claims Act, 28 U.S.C. § 1346(b), and regulation 10 C.F.R. § 14.1 (1975). This is a claim against the United States for money damages for injury to and loss of property caused by the negligent and wrongful acts and omissions of employees of the NRC while acting within the scope of their employment.

This claim is filed before the NRC for disposition in accordance with the provisions of 28 U.S.C. § 2675 and 10 C.F.R. § 14.1 et seq.

The Parties

3. Claimant GPU is incorporated in Pennsylvania and

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~~8012160604~~

has its principal place of business in New Jersey. GPU is an investor-owned public utility holding company, operating pursuant to the Public Utility Holding Company Act of 1935, 15 U.S.C. § 79 <u>et seq</u>., and owning all of the common stock of three operating electric company subsidiaries, claimants JCP&L, Met-Ed and Penelec. As used herein, GPU refers to GPU and all of its operating subsidiaries.

4. Claimant JCP&L is incorporated in New Jersey and has its principal place of business in New Jersey. JCP&L sells electrical energy to retail customers in north-central, east-central, northwestern and western New Jersey and to other electric companies and entities for resale.

5. Claimant Met-Ed is incorporated in Pennsylvania and has its principal place of business in Pennsylvania. Met-Ed sells electrical energy to retail customers in eastern, east-central and southeastern Pennsylvania and to other electric companies and entities for resale.

6. Claimant Penelec is incorporated in Pennsylvania and has its principal place of business in Pennsylvania. Penelec sells electrical energy to retail customers located in western, northern and south-central Pennsylvania and to other electric companies and entities for resale.

7. JCP&L, Met-Ed and Penelec are co-owners of the nuclear electric generating facility known as Three Mile Island Unit No. 2 ("TMI-2"), which is located in Londonderry Township, Pennsylvania. Met-Ed owns an undivided 50% interest in TMI-2, and JCP&L and Penelec each own an undivided 25% interest in TMI-2. Met-Ed is the operator of TMI-2. The Atomic Energy Commission issued an Operating License, DPR-73, to Met-Ed for TMI-2 on February 8, 1978. The nuclear steam supply system in TMI-2, including the nuclear reactor and substantially all of

the engineered safety systems that control the nuclear reactor, were supplied by The Babcock & Wilcox Company ("B&W").

8. All of the major electric generation, transmission and distribution facilities of the claimants are physically interconnected. The operations of these electric facilities are centrally coordinated within GPU to function as a single, integrated electric utility system known as the GPU System. The energy generated by TMI-2, when operating, is commingled with the energy generated throughout the GPU System and is transmitted throughout the GPU System and distributed to retail customers or sold to other electric companies and entities for resale.

9. The NRC is a federal executive agency, established by the Energy Reorganization Act of 1974, PL 93-438, 88 Stat. 1233, 42 U.S.C. § 5814 et seq., as a successor agency to the Atomic Energy Commission. As used herein, NRC refers to the present agency, its predecessor, the Atomic Energy Commission, and all present and former divisions, offices, employees and agents of the NRC. By statute, the NRC is charged with the establishment of "standards and instructions to govern the possession and use of special nuclear material, source material, and byproduct material as the Commission may deem necessary or desirable to promote the common defense and security or to protect health or to minimize danger to life or property." 42 U.S.C. § 2201(b).

10. The NRC has the authority and duty to regulate the design and operation of commercial nuclear power plants within the United States. In the 1974 Energy Reorganization Act, supra, Congress authorized the NRC to "prescribe such regulations or orders as it may deem necessary . . . to govern any activity authorized pursuant to this chapter, including standards and restrictions governing the design, location, and operation of

facilities used in the conduct of such activity. . . ." 42 U.S.C. § 2201(i). The NRC has promulgated regulations setting forth mandatory agency operating procedures which are set forth in the Code of Federal Regulations, the NRC Regulatory Guide, the NRC Manual, the Office of Inspection and Enforcement Manual, the Standard Review Plan and other guides, manuals and publications which, in relevant part, are described below.

11. In the operational exercise of its statutory and regulatory duties, the NRC induced GPU and Met-Ed to rely and GPU and Met-Ed did rely upon the NRC to warn of defects in equipment, analyses, procedures and training affecting the operation of TMI-2 of which the NRC was or should have been aware. The NRC, in the operational exercise of its statutory and regulatory duties, induced GPU and Met-Ed to rely and GPU and Met-Ed did rely upon the NRC to review with due care the equipment, analyses, procedures and training for nuclear plant operation submitted to the NRC by nuclear equipment vendors and nuclear plant licensees.

The March 28, 1979 Accident at TMI-2

12. On March 28, 1979, beginning at 4:00 A.M., while TMI-2 was operating at about 97% of full power, the turbine generator shut down or "tripped" due to sudden loss of feedwater. Under NRC regulations, such an unscheduled turbine generator trip is an "anticipated operational occurrence" which is required to be planned for in the design of a nuclear plant. As with any such shutdown, the removal of heat from the primary loop by the secondary loop was reduced substantially. Within seconds, the continuing buildup of heat in the primary loop raised the pressure in the reactor coolant system. In turn, this caused a relief valve on the pressurizer (the "pilot-operated relief valve") to open, as it was designed to do, in

at what moment

by what authority

order to relieve the excess pressure.

13. Several seconds after the pilot-operated relief valve had opened, the reactor shut down or "scrammed," causing the pressure in the reactor coolant system to drop to within its normal range. At that point, the pilot-operated relief valve should have closed.

14. In fact, the pilot-operated relief valve improperly failed to close and, because of a lack of instrumentation to indicate clearly either the open position of the valve or the existence of flow through the valve, the operators at TMI-2 were unaware that the valve had failed to close. Thereafter, significant quantities of coolant water and steam escaped through the stuck-open valve, and a "loss-of-coolant accident" was in progress.

there was prior leakage not attended to by TMI - see NRC comments add'l invest & doc. nec.

15. As more coolant water and steam escaped, the pressure in the reactor coolant system continued to drop. Within approximately two minutes, the pressure fell to a level at which an engineered safety system began providing high-pressure injection of water into the reactor coolant system to replace the lost coolant and ensure that the nuclear core was covered and protected by coolant.

16. Approximately five minutes after the 4:00 A.M. turbine generator trip, the TMI-2 operators substantially reduced the high-pressure injection of replacement coolant into the reactor, in accordance with B&W-supplied limits and precautions, procedures and training, which the NRC had reviewed as described herein.

add'l invest & ... see NRC comments

17. As a result of the loss of coolant through the stuck-open pilot-operated relief valve and the lack of replacement coolant, the nuclear fuel core overheated, severely damaging the protective cladding on the nuclear fuel and substan-

add'l invest re why operators failed to recognize PORV open - see NRC comments

tially destroying portions of the nuclear fuel core. Radioactive material from the ruptured nuclear fuel core spread throughout the surrounding reactor equipment, further damaging and contaminating large portions of the nuclear steam supply system and other equipment and structures in the TMI-2 containment building and in the adjacent fuel auxiliary and intermediate buildings.

THE NRC'S NEGLIGENT PERFORMANCE AND OMISSIONS OF ITS OPERATIONAL FUNCTIONS

add'l insert. reqd. - see NRC comments

18. Prior to the March 28, 1979 accident at TMI-2, the NRC both had reason to know and actually knew that there were defects in the equipment, analyses, procedures and training supplied by B&W for TMI-2. Notwithstanding the statutory and regulatory duties of the NRC to warn nuclear plant licensees of such defects, and notwithstanding the reliance by GPU and Met-Ed on the NRC for the dissemination of such warnings, the NRC negligently failed to warn GPU or Met-Ed of such defects in TMI-2. That failure to warn by the NRC was a proximate cause of the March 28, 1979 accident.

1. generally [illegible]

19-23 add'l insert reqd - see NRC comments

19. Pursuant to NRC regulations, the NRC Office of Inspection and Enforcement is specifically required to inspect nuclear plant licensees to "ascertain the status of compliance with NRC requirements including rules, regulations, orders and license provisions," and to "[i]nvestigate incidents, accidents, allegations, and other unusual circumstances involving matters in the nuclear industry which may be subject to NRC jurisdiction to ascertain the facts and to take or recommend appropriate actions." NRC Manual Ch. 0127 (1978).

20. The NRC regulations mandate that the NRC disseminate among licensees of nuclear power plants information derived from operating experience at all nuclear plants in the United States, including data on component failures and procedure

changes. 10 C.F.R. § 1.64 (1977); NRC Manual Ch. 0127 (1978). The NRC requires licensees to report unscheduled incidents or events which involve variations from regulations, technical specifications or license conditions. 10 C.F.R. § 21 (1977); NRC Regulatory Guide 1.16 (1975). The reports, called Licensee Event Reports, are submitted to the NRC Office of Inspection and Enforcement to aid the NRC in obtaining corrective action at the reporting plant and in preventing a similar occurrence at other nuclear plants. The director of each division within the Office of Inspection and Enforcement is required to "[e]valuate licensee event reports and Regional reports to identify generic problems and to determine the significance of individual incidents. . . ." NRC Manual Ch. 0127 (1978).

3. add'l invest

21. The NRC Office of Inspection and Enforcement is responsible for evaluating licensee and NRC responses to incidents or accidents to "assure adequacy of the overall response to the incident or accident." The Regional Director must review significant events, allegations and investigatory findings for matters having generic applicability. Regional Inspection and Enforcement Directors are required to review all reports mandated by NRC regulations, including all Licensee Event Reports. NRC Manual Ch. 0127 (1978); Inspection and Enforcement Manual Ch. 1110-051 (1978).

3. addit' invest

22. As the primary recipient of plant operating data, the NRC Office of Inspection and Enforcement is required by regulations to analyze and disseminate important safety information to other NRC offices and to all nuclear plant licensees. NRC Manual Ch. 0127 (1978). One of the principal methods used by the NRC Office of Inspection and Enforcement for advising licensees of important safety matters is through the issuance of Bulletins and Circulars. Because of the importance of these

notices for warning licensees of possible defects and safety problems, NRC offices other than Inspection and Enforcement, such as Nuclear Reactor Regulation and Nuclear Materials Safety and Safeguards, also recommend the issuance of Bulletins and Circulars on particular subjects. Inspection and Enforcement Manual Ch. 1125-052 (1978). NRC regulations direct all NRC staff to be alert to any information which has potential safety significance. The regulations require every member of the NRC staff "to be alert to the emergence of information -- from outside sources or within the staff -- which is new, potentially important, and potentially relevant to one or more pending proceedings." Inspection and Enforcement Manual Ch. 1530 (1978).

23. NRC regulations impose a duty on the NRC Office of Inspection and Enforcement to issue Bulletins regarding matters of "safety, safeguards and environmental significance" for nuclear plants and to require that licensees take specific actions as a result of safety-related design inadequacies, equipment defects, operating inadequacies, malfunctions, or any other failures of a generic nature that have occurred at a similar facility or operation. A Bulletin requires licensees to inspect for and correct the inadequacies described in the Bulletin. The Inspection and Enforcement Manual requires the issuance of Bulletins when an event or condition is generic and important to safety. Inspection and Enforcement Manual Ch. 1125-031, 1125-041 (1978).

24. GPU and Met-Ed relied on the NRC to comply with the comprehensive requirements of data collection, analysis and dissemination set forth in statutes and regulations. GPU and Met-Ed relied on the NRC to issue warnings as required by NRC regulations. Met-Ed maintained a formal program for the review of communications from the Office of Inspection and Enforcement to determine whether any adverse condition reported by the NRC

required corrective action at TMI-2. Met-Ed, the operator of TMI-2, promptly disseminated information from NRC Bulletins within Met-Ed and required prompt replies and appropriate action.

THE NRC's NEGLIGENT FAILURE TO GIVE WARNING BASED ON THE DAVIS-BESSE INCIDENT

true except very low power at D-B timing ...

25. In September 1977, a loss-of-coolant accident occurred at the B&W-supplied Davis-Besse I nuclear power plant of the Toledo Edison Company. That accident closely paralleled the events which occurred 18 months later at TMI-2.

further invest — did MET-ED have any knowledge — pursued it — see I&E comment on 31

26. Following the September 1977 incident at Davis-Besse, the NRC negligently failed to perform its duty (a) to adequately to investigate and ascertain the facts, (b) to take and recommend appropriate action and (c) to warn Met-Ed and other licensees of B&W-supplied nuclear plants of defects in equipment, analyses, procedures and training which the NRC had discovered or should have discovered as a result of the Davis-Besse incident. These negligent failures contravened NRC duties imposed by statute and regulations and were inconsistent with duties previously undertaken by the NRC. The NRC thus negligently performed and negligently omitted to perform operational functions mandated by statute, NRC regulations and past agency practice. If a proper warning had been given by the NRC, the TMI-2 accident on March 28, 1979 would have been avoided.

what invest nailed — see I&E comment

27. On September 24, 1977, while the Davis-Besse plant was operating at 9% of full power, a sudden loss of feedwater caused a turbine generator trip. When the pilot-operated relief valve subsequently opened and failed to close, the Davis-Besse plant experienced a loss-of-coolant accident. As reactor coolant pressure dropped, the high-pressure injection of replacement coolant activated automatically. The Davis-Besse

nuclear steam supply system design did not have a direct indicator of whether the pilot-operated relief valve was open or closed or whether there was a flow of coolant through the relief valve. Just as happened later at TMI-2, the water level in the pressurizer began to rise, misleading the operators into concluding that there was no loss-of-coolant accident in progress. Acting pursuant to NRC-reviewed limits and precautions, procedures and training, the operators at Davis-Besse then terminated the high-pressure injection of replacement coolant into the reactor coolant system.

what happened after then?

2. addit invest

28. Immediately following the September 1977 Davis-Besse incident, the NRC began an investigation, as required by 10 C.F.R. § 1.64 (1977), which included operator interviews and reviews of plant operating data, equipment and operator action.

29. The NRC conducted another investigation of the September 1977 Davis-Besse incident during 1978, which resulted in the NRC implementing a revised Operating Procedure for Davis-Besse, described more fully below at paragraph 34.

3.

30. As a result of the investigation and analyses of facts, which regulations required the NRC to perform following the Davis-Besse incident, the NRC knew or should have known the following:

(a) There were defects in equipment application and instrumentation of the Davis-Besse plant, including excessive reliance on the B&W-supplied pilot-operated relief valve to open and close and a lack of instrumentation to indicate the valve position;

addit invest needed

(b) There were defects in the transient analyses previously supplied by B&W and reviewed by the NRC, including a failure to analyze adequately potential breaks in the coolant system as small as a stuck-open pilot-operated relief valve and

a failure to analyze adequately potential breaks located at the top of the pressurizer;

(c) There were defects in the limits and precautions, procedures and training reviewed by the NRC which improperly directed plant operators not to permit the pressurizer to become filled with water or "go solid;"

(d) There were defects in the operating and emergency procedures, including procedures which improperly permitted premature termination of high-pressure injection before the operators had identified and arrested a loss-of-coolant accident;

(e) Unanticipated boiling of the water in the reactor coolant system at Davis-Besse had caused a rise in pressurizer water level which misled plant operators into concluding that there was no loss of water from the reactor coolant system.

31. Each of the defects and operating problems set forth in paragraph 30 were generic problems affecting TMI-2 and other B&W-supplied nuclear plants because those plants contained similar equipment and instrumentation and relied upon similar procedures and analyses. NRC regulations required the Commission to "[notify] licensees regarding generic problems so as to achieve appropriate precautionary or corrective action." 10 C.F.R. § 1.64 (1977). Nevertheless, the NRC negligently failed to notify licensees, including Met-Ed, of these "generic problems," which it knew about or should have known about as a result of the Davis-Besse incident. That failure was a proximate cause of the accident at TMI-2 on March 28, 1979.

32. The NRC negligently disseminated to nuclear plant licensees, including Met-Ed, summaries of Licensee Event Reports regarding the Davis-Besse incident which failed to warn that the

true?

operators at Davis-Besse had prematurely terminated high-pressure injection before determining whether a loss-of-coolant accident was in progress. Toledo Edison Corp., "Licensee Event Report: NP-32-77-16," Docket 50-346, October 1977.

33. The NRC negligently disseminated to nuclear plant licensees, including Met-Ed, a summary of an erroneous supplemental Licensee Event Report on the Davis-Besse incident. That report erroneously concluded that "[o]perator action was timely and proper throughout the sequence of events." As a result, the NRC failed to warn Met-Ed that the Davis-Besse operator action had aggravated the loss-of-coolant accident by terminating high-pressure injection of coolant. Toledo Edison Corp., "Licensee Event Report: NP-32-77-16 Supplement," Docket 50-346, November 1977.

34. More than a year after the Davis-Besse incident, the NRC implemented new operating procedures for Davis-Besse to prevent a recurrence of the September 1977 accident. These procedures stated:

> "NOTE: Prior to securing HPI [high-pressure injection], insure that a leak does not exist in the pressurizer such as a safety valve or an electromagnetic [i.e., pilot-operated] relief valve stuck open. A minimum decay heat flow of 2800 gpm is required prior to securing high-pressure injection. If the leak has been isolated, the high-pressure injection pump can be shut down after RCS [reactor coolant system] pressure increases above the shutoff head of the pump."

Davis-Besse No. 1, Emergency Procedure EP 1202.06, "Loss of Reactor Coolant and Reactor Coolant Pressure." The NRC negligently failed to direct the implementation of this new operating procedure by licensees of other B&W-supplied nuclear plants, including Met-Ed.

35. In addition to releasing incomplete, erroneous and misleading Licensee Event Reports regarding the loss-of-

legal conclusion not based on fact

D-B accident reversal

is this NRC responsibility

D-B accident ? relevant true

& justified tudor't ans. goes to ? contents ?

coolant accident at Davis-Besse, the NRC Office of Inspection and Enforcement negligently failed to issue any Bulletin or Circular warning licensees of B&W-supplied nuclear plants, including Met-Ed, of the deficiencies in the equipment, analyses, procedures and training which the NRC had discovered or should have discovered as a result of the September 1977 Davis-Besse incident. As a result of its investigations of the Davis-Besse incident, the NRC knew that the equipment and procedural deficiencies were generic to B&W-supplied plants and that the substituted operating procedure was important to the safe operation of the plant in that it instructed operators to take steps which would avoid core uncovery. Thus, the NRC knew that a Bulletin was mandated by NRC regulations, see paragraphs 22-23, supra. Nevertheless, the NRC negligently failed to issue a Bulletin.

36. Section 208 of the Energy Reorganization Act of 1974, as amended, requires the NRC to determine which incidents and events represent Abnormal Occurrences and to report those Abnormal Occurrences to Congress. The NRC must disseminate information relating to an Abnormal Occurrence to the public within 15 days after the NRC has learned of its occurrence. Inspection and Enforcement Manual Ch. 1110. "Abnormal Occurrences" include

> "Design or Safety Analysis Deficiency, Personnel Error, or Procedural or Administrative Inadequacy:
>
> 1. Discovery of a major condition not specifically considered in the Safety Analysis Report (SAR) or technical specifications that require immediate remedial action.
>
> 2. Personnel error or procedural deficiencies which result in loss of plant capability to perform essential safety functions such that a potential release of radioactivity in excess of 10 CFR Part 100 guidelines could result from a postulated transient or accident (e.g., loss of emergency core cooling system, loss of control rod system)."

Inspection and Enforcement Manual Ch. 1110, Appendix A. The NRC knew that the procedure which misled operators at Davis-Besse prematurely to terminate high-pressure injection was a "procedural deficienc[y]," as defined by the Manual.

37. In violation of NRC regulations (Inspection and Enforcement Manual Ch. 1110, Appendix A), the NRC negligently failed to classify the September 1977 Davis-Besse incident as an Abnormal Occurrence in its subsequent quarterly or annual report to Congress, thereby failing to warn licensees of other B&W-supplied nuclear power plants, including Met-Ed, of the defects and operating problems revealed by this event which required immediate remedial action at similar plants, such as TMI-2.

warn'g this within NRC discretion

reporting & classification of Davis-Besse results & dissemination of info thereabout, must be further investigated

38. In addition to the failure of the NRC to warn licensees of B&W-supplied nuclear plants of defects and problems, of which the NRC was aware as a result of the Davis-Besse incident, the NRC negligently failed to act in other ways to investigate, discover and warn licensees of defects of which the NRC should have been aware as a result of the Davis-Besse incident. These negligent failures by the NRC have been documented by the NRC in a four volume report which the NRC approved, published and released in January 1980 to the public, entitled "Three Mile Island -- A Report to the Commissioners and to the Public" (hereinafter "Special Inquiry"). The NRC, in its Special Inquiry, admits that:

(a) NRC staff personnel incorrectly advised the NRC Advisory Committee on Reactor Safety (ACRS) that the consequences of a loss-of-coolant accident, such as had occurred at Davis-Besse, did not need to be examined for a reactor operating at full power -- as TMI-2 would be on March 28, 1979 -- because of the low probability of such an event occurring at full power. (Special Inquiry, Vol. II, Part 1 at 154)

how would NRC know this?

further comment on D-B incident

what are our normal procedures

(b) While the NRC recognized that it should examine the basis for the decision by the operators at Davis-Besse to terminate high-pressure injection, the NRC failed to direct its inspectors to resolve this issue. (Special Inquiry, Vol. II, Part 1 at 152)

39. Even though the NRC knew or should have known that there was an unreasonably high rate of failure of pilot-operated relief valves supplied by various manufacturers for nuclear plants, the NRC erroneously concluded that the failure of the pilot-operated relief valve at Davis-Besse in September 1977 had no safety implications for other nuclear plants containing pilot-operated relief valves designed by different manufacturers. Another failure of the pilot-operated relief valve occurred at Davis-Besse in October 1977, while the NRC was investigating the September 24 incident. The NRC, in its Special Inquiry, admits that the NRC knew that "similar pieces of equipment with comparable probabilities of failure and similar failure modes were installed on other B&W plants and, in some cases on all pressurized water reactors." (Special Inquiry, Vol. 2, Part 1 at 156)

40. The final report of the NRC Inspection and Enforcement inspectors in Region III, where Davis-Besse is located, failed to identify the generic implications of the Davis-Besse incident, including the misleading rise in pressurizer water level, the incorrect operator response to pressurizer level and the misleading limits and precautions, procedures and training, reviewed by the NRC, which had directed that erroneous operator response.

41. Prior to the September 1977 Davis-Besse incident, the NRC knew or with due care should have known from other reports which it had received that its previous evaluations of

further invest. needed re NRC control(?) over training

B&W equipment, analyses, procedures and training were inadequate. As the NRC Special Inquiry admits, the NRC omitted to heed these early "precursors," just as the Commission later failed to respond with due care to the Davis-Besse incident, as described at paragraphs 25-40, supra. These earlier precursors included:

(a) In 1971, the Atomic Energy Commission, the predecessor agency to the NRC, was specifically advised that a small-break loss-of-coolant accident at the top of a pressurizer -- as was to occur at TMI-2 on March 28, 1979 -- could create misleading signals, thereby interfering with high-pressure injection of coolant. Although the NRC was thus on notice that it should analyze misleading signals of water level caused by such an accident, the NRC negligently failed to perform that analysis or require suppliers of nuclear equipment, such as B&W, to perform that analysis. (Special Inquiry, Vol. II, Part 1 at 139-40)

(b) In 1975, the NRC completed a comprehensive report on nuclear reactor safety, "The Reactor Safety Study (WASH-1400),"* which concluded that small-break loss-of-coolant accidents -- such as the failure of a pilot-operated relief valve to close -- were among the highest probability risks in a nuclear plant. (Special Inquiry, Vol. II, Part 1 at 142) Yet, the NRC failed to analyze or require nuclear equipment suppliers, such as B&W, to provide adequate analyses of small breaks.

(c) In 1977, the NRC substantially ignored a report prepared by Carlyle Michelson, a consultant to its Advisory Committee on Reactor Safeguards, which put the NRC on notice that neither the small-break analyses supplied by nuclear equipment suppliers nor the computer models then used to predict reactor-coolant-system behavior were valid for analyzing small-break loss-of-coolant accidents. (Special Inquiry, Vol. II, Part 1 at 144-46)

1. probably true — to whatever extent NRC undertook the obligation — probably a great deal of discretion here — add'l argument re duty

4. conjecture

4. conjecture — legal conclusion

4. conjecture — what was NRC's "due care"

2. seems like duty + [illegible] — one [illegible] here see NRR comment

EFFECTS OF NRC'S NEGLIGENT PERFORMANCE AND OMISSIONS OF ITS OPERATIONAL FUNCTIONS

42. GPU and Met-Ed relied on the NRC to issue warnings of defects in equipment, analyses, procedures and training in accordance with the NRC's statutory and regulatory duties.

43. If the NRC had exercised due care in investigating the Davis-Besse incident and analyzing other precursors, and if the NRC had issued correct warnings of generic problems in B&W equipment, analyses, procedures and training, GPU and Met-Ed would have had the equipment, instrumentation, procedures and training reasonably needed to avoid the accident on March 28, 1979.

44. The negligent failure by the NRC to issue Bulletins, Abnormal Occurrence Reports and other warnings required by statute and NRC regulations was a proximate cause of the March 28, 1979 accident at TMI-2.

NRC'S NEGLIGENT IMPLEMENTATION OF REVIEW REQUIREMENTS

45. A proximate cause of the March 28, 1979 accident at TMI-2 was the failure of the NRC to review with due care, or in accordance with statutes and regulations, the equipment, analyses, procedures and training supplied by B&W for TMI-2.

46. Pursuant to statutory and regulatory authority, the NRC issues licenses for the construction and operation of each commercial nuclear power plant in the United States. (42 U.S.C. § 2133(b)). The NRC

> "is responsible for managing safety reviews of applications for construction permits and operating licenses for reactors and evaluations of standard plant designs; evaluates technical specifications . . . [and] performs technical reviews and analyses of mechanical, structural, and materials engineering aspects of reactor systems, core performance, auxiliary systems, control systems, mechanical components, reactor structures, and power systems."

10 C.F.R. § 1.61 (1977).

The NRC Office of Nuclear Reactor Regulation "reviews applications [for licenses] and issues licenses . . . and evaluates the health, safety, and environmental aspects" of a plant prior to the approval of a Preliminary Safety Analysis Report or a Final Safety Analysis Report, which incorporate the equipment vendors' analyses, evaluations and descriptions of operation of all components and systems. 10 C.F.R. § 1.61 (1977), § 50.34 (1978).

1.

47. The NRC Office of Nuclear Regulation is required by statute to

> "[r]eview the safety and safeguards of all such facilities, materials, and activities, and such review functions shall include, but not be limited to monitoring, testing and recommending upgrading of systems designed to prevent substantial health or safety hazards. . . ."

42 U.S.C. § 5843(b).

2. Duty

48. Applicants for nuclear plant construction permits must submit for NRC review and approval "principal design criteria" for the proposed facility. 10 C.F.R. § 50.34 (1978). These principal design criteria "establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. . . ." 10 C.F.R. § 50, Appendix A. The NRC has promulgated General Design Criteria and has a duty to review equipment and designs for conformance to the General Design Criteria which "establish minimum requirements for the principal design criteria" for all commercial nuclear power plants; id. The NRC has further promulgated and has a duty to enforce additional design requirements described throughout the appendices to 10 C.F.R. § 50.

5. Fortunately min. [illegible]
Did Met Ed not have a duty to do [illegible]?
further word necessary.

49. GPU and Met-Ed relied on the NRC review of B&W equipment, analyses, procedures and training to provide for the safe operation of TMI-2.

3. add'l invest. sec.

Negligent Review and Approval of B&W
Topical Reports and B&W Generic Designs

50. Prior to any licensing submission by Met-Ed for TMI-2, the NRC has already reviewed and negligently approved numerous topical reports and generic models prepared by B&W for nuclear plant design and operation. These topical reports described generic systems in and the operation of B&W nuclear plants and were a means by which the NRC was able to review generic features once, rather than repetitiously for each succeeding B&W plant. Licensees, such as Met-Ed, have no input in the creation of topical reports and rely on the NRC to review the reports with due care prior to approving them for use in subsequent nuclear plant licensing proceedings. In reliance upon the prior review and approval by the NRC of topical reports, prospective licensees, such as Met-Ed, incorporate such reports by reference into the Safety Analysis Report for specific nuclear plants.

2. TMI went to great lengths to convince NRC of similarity — apparently to avoid necessity for review ... — see NRR comments

51. During the licensing of TMI-2, the NRC acknowledged that:

> Many features of the design of TMI-2 are similar to those we have evaluated and approved previously for other nuclear plants now under construction or in operation. To the extent feasible and appropriate we have relied on our earlier reviews for those features which were shown to be substantially the same as those previously considered. Where this has been done, the appropriate section of this report identifies the facility involved.

NRC Safety Evaluation Report for the Operating License on TMI Unit 2 (1976).

3. addit'l invest. needed regarding NRC & small break accident

52. GPU and Met-Ed relied upon and incorporated by reference in the TMI-2 Final Safety Analysis Report, a number of B&W topical reports previously reviewed and negligently approved by the NRC. GPU and Met-Ed reasonably relied on the NRC to have reviewed these submissions with due care. These included

previously approved topical reports, relating to small-break analysis, loss-of-coolant accident analysis and emergency core cooling system performance in B&W plants of substantially the same design type as TMI-2, specifically the B&W type 177-FA "lowered-loop" nuclear plants. Prior to the licensing of TMI-2, the NRC had licensed eight B&W plants, including seven which contained a 177-FA lowered-loop design. The earliest lowered-loop plant was Oconee I, licensed in 1973.

Transient Analyses

53. The NRC negligently approved B&W transient analyses for TMI-2, including those for small-break loss-of-coolant accidents and for loss of normal feedwater, even though those analyses failed to comply with NRC regulations. The NRC knew that transient analyses in compliance with NRC regulations are necessary for proper plant design and operation. A transient is an unintended change in power level or system condition in a nuclear plant, and includes anticipated operational occurrences such as a loss-of-normal-feedwater transient, which occurred at TMI-2 on March 28, 1979.

54. As set forth in paragraphs 55 and 56, below, the NRC failed to evaluate with due care B&W transient analyses and failed to compel B&W, either during the TMI-2 plant licensing process or as part of B&W's prior submission of topical reports, to submit transient analyses which complied with NRC regulations, including the Standard Review Plan and the General Design Criteria. As a result, the NRC negligently failed to require B&W to submit the transient analyses necessary for proper design and operation of TMI-2.

55. The NRC has admitted, with respect to the transient analyses submitted by B&W prior to and in support of the licensing of TMI-2, that the NRC failed to enforce compliance

3 of 5

Core licensing! Why?

NRC analyses assumed E-FS would not be put i service & the operators would not act as they had to throttle cooling w[illegible]

the idea of "never" "restpoints" must be looked into — what was our basis

3-need addit'l investigation

with the requirements of its Standard Review Plan, Section 15. As the NRC has stated:

> "The TMI-2 accident started with a loss of feedwater transient and, because of the stuck-open power operated relief valve, a small break loss-of-coolant accident resulted. According to the Standard Review Plan, such a sequence should have been analyzed in the licensing process, but it was not."

NUREG 0560, Staff Report on the Generic Assessment of Feedwater Transients in the PWR's Designed by The Babcock & Wilcox Co. (1979) at 5-4.

5. Factually incorrect - according to NRC feedwater transient - this was not an AOO because total feedwater cut-off is never anticipated

56. The NRC has admitted in the respects described in paragraph 57, below, that it failed to comply with the requirements of its General Design Criteria. As the Commission has stated:

> "Feedwater transients are anticipated operational occurrences (AOOs), since they are expected to occur one or more times during the life of a nuclear plant. The basic requirements for AOO's are given in General Design Criteria (GDC) 10 and 15. GDC-10 requires that specified acceptable fuel design limits not be exceeded during AOOs. GDC-14 and GDC-15 require that the design of the reactor coolant pressure boundary should preclude abnormal leakage and the design conditions of the boundary should not be exceeded during AOO's. Additional requirements specified in GDC-13 are: 'Instrumentation shall be provided to monitor variables and systems over their anticipated ranges . . . for anticipated operational occurrences . . . as appropriate to assure adequate safety. . . . Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.' GDC-20 states the general requirements for protection systems, including the following: 'The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specific acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences . . . .'
>
> <u>In the light of the TMI-2 experience, it is apparent that applicable criteria were not met.</u>"

Id. at 5-5 (emphasis added).

2. Facts

57. The NRC negligently failed to comply with the regulations described in paragraphs 55 and 56, above, in that

the feedwater transient analyses and small-break loss-of-coolant analyses submitted by B&W and approved by the NRC were inadequate to provide a proper basis for plant design and for the development of operator training programs and operating procedures. Specifically:

(a) The NRC failed to require B&W to submit the necessary analysis of any break size smaller than 0.040 square feet. As a result, the NRC failed to require the necessary analyses of breaks equivalent to the size of a pilot-operated relief valve (0.007 square feet) which had failed to close.

(b) The NRC failed to require B&W to submit the necessary analysis of a small break occurring in the steam space at the top of the pressurizer, where the pilot-operated relief valve is located.

(c) The NRC failed to require B&W to submit the necessary analysis of a pilot-operated relief valve failing to close, even though such a failure should have been assumed since the valve was designated as non-safety grade equipment.

(d) The NRC failed to require B&W to submit analyses which examined more than the initial minutes of a transient, whereas such analyses should have covered the time period until a stable system had been assured.

(e) The NRC failed to require B&W to submit analyses of the sensitivity of the foregoing small-break loss-of-coolant analyses (subparagraphs a-c, above), to reactor coolant pump operation or non-operation.

4. Completion —

58. As set forth in paragraph 41(b), *supra*, the NRC knew, at least at the time that it reviewed and published in 1975 the "Reactor Safety Study (WASH-1400)," that small-break loss-of-coolant accidents were substantially more likely to occur in a nuclear plant than large-break loss-of-coolant

accidents. Yet, the NRC failed to examine the B&W design and procedures with due care to ascertain the likelihood of small-break loss-of-coolant accidents and their consequences even after the Davis-Besse incident, which was a small-break loss-of-coolant accident.

2.

59. The NRC has admitted in post-accident reports that the B&W analyses submitted to the NRC had failed to provide necessary information needed for operator action following a small break. Generic Evaluation of Small Break and Loss of Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants, NUREG 0565 (1980) at 1-1.

4. Conjecture - what about violation in FW system

60. If the NRC had reviewed B&W topical reports and license submissions with due care, and had required B&W to provide the transient analyses required by NRC regulations, the March 28, 1979 accident at TMI-2 would have been avoided.

Procedures

61. Prior to the issuance of the TMI-2 operating license, the NRC Office of Inspection and Enforcement conducted an extensive audit of the TMI-2 procedures which were drafted by B&W. The audit included a review of the procedures which were later used by the operators during the March 28, 1979 accident. The NRC negligently failed to identify deficiencies in these B&W-drafted procedures and instead found that the "technical content [of the procedures] was adequate to assure satisfactory performance of intended functions." Inspection and Enforcement Report No. 77-26, August 1977.

Find memory entire document but actually constitutes "approval"

62. The NRC negligently reviewed procedures for operating TMI-2 which incorrectly prohibited permitting the pressurizer to "go solid." The TMI-2 operating procedure 2103.1.3 (Revision 0, 1977), supplied by B&W, contained the

11

following prohibition:

> "2.1.8 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."

This procedure contained no exception for emergency conditions even if there were risks of core uncovery.

D. Besse

63. The NRC knew or should have known as a result of its investigation of the Davis-Besse incident which confirmed earlier precursors, that the failure of a pilot-operated relief valve to close would cause the water level in the pressurizer to rise even though the reactor coolant system was not "going solid," see paragraphs 25-41, *supra*. Nevertheless, in the 18 months following the September 1977 incident at Davis-Besse, the NRC negligently failed to modify or direct a modification of the procedures for TMI-2. As a result, Met-Ed and GPU continued to rely on the NRC-reviewed procedures, which incorrectly proscribed filling the pressurizer "solid" with water and risked uncovering the core during small-break loss-of-coolant accidents.

conclusion does not take into account TMI violations

64. The failure of the NRC to warn GPU and Met-Ed of defects in the TMI-2 procedures was a proximate cause of the accident on March 28, 1979. On March 28, 1979, almost immediately after a turbine generator trip occurred at TMI-2, the TMI-2 operators observed that the water level in the pressurizer was rising higher than allowed by the procedures reviewed by the NRC.

Leg.! conclusion not valid — fault ~~there~~ is this would have never resulted

65. The TMI-2 procedures, negligently reviewed by the NRC, had prescribed the wrong course of action. Although the water level in the pressurizer was high, the entire reactor coolant system was not "solid" with water. Instead, the indicated water level in the pressurizer remained high due both to increasing amounts of steam elsewhere in the reactor coolant system and to the stuck-open pilot-operated relief valve at the

top of the pressurizer through which coolant water and steam continued to escape. Rather than improving the situation, the reduction of high-pressure injection, as prescribed by the procedures negligently reviewed by the NRC, resulted in a failure to replace the coolant escaping through the stuck-open valve.

Pilot-Operated Relief Valve

Conclusion

66. The NRC failed to exercise due care in reviewing B&W equipment, analyses and procedures, including the determination of reactor trip points, and other operating procedures which placed heavy reliance on the repeated and correct operation of the pilot-operated relief valve. The NRC negligently failed to review properly B&W transient analyses, as set forth in paragraphs 53-58, above, to determine the frequency with which the pilot-operated relief valve would be required to function or to determine the probability of failure of that valve. From its investigation of the Davis-Besse incident and analysis of operational data from other plants, the NRC knew or should have known of prior failures related to pilot-operated relief valves. Therefore the NRC should not have approved B&W's equipment, analyses and procedures which relied on repeated opening and closing of that valve. Staff Reports to the President's Commission on the Accident at Three Mile Island (Kemeny Commission), Reports of the Technical Assessment Task Force, Vol. IV at 193-99.

Training and Operator Licensing

4. Duty re training & procedures

67. The NRC, in the implementation of regulations requiring it to license operators, failed to exercise due care in assuring that licensed operators were properly trained to respond to transients such as occurred at TMI-2 on March 28, 1979. NRC regulations require examinations by the NRC and

2. add'l invest needed re facts - did NRC encourage simulator use?

encourage the use of simulators by vendors. 10 C.F.R. § 55.11 (1963); 10 C.F.R. § 55.20 (1975); 10 C.F.R. § 55.22 (1975); 10 C.F.R. § 55.23 (1963); 10 C.F.R. § 55, Appendix A (1976).

68. NRC regulations require that candidates for operating licenses take an operating test which includes a reactor startup from shutdown to power. 10 C.F.R. § 55.23 (1963). In fulfillment of the requirements of 10 C.F.R. § 55.23 (1963), the NRC specifically sanctioned the use of a "cold" licensing program which included a minimum of one week of training on a nuclear plant simulator. The initial TMI-2 staff of control room operators were trained in a "cold" licensing program, utilizing B&W's simulator, which was reviewed by the NRC Operating Licensing Branch for compliance with established standards and was formally approved. The training program for TMI-2 operators included eight weeks of training on the B&W simulator. Staff Reports to the President's Commission on The Accident at Three Mile Island (Kemeny Commission), <u>Reports of The Technical Assessment Task Force</u>, Vol. III at 15-16; <u>NRC Operator Licensing Guide</u>, NUREG 0094.

1-5. [illegible] simulator review never stated - [illegible] "informal" [illegible]

69. The NRC negligently certified the B&W simulator used in training the TMI-2 operators even though the NRC knew or with due care should have known that the B&W simulator was defectively designed and programmed. The NRC negligently failed to detect or correct the fact that B&W's training of TMI-2 operators by simulator and otherwise was inadequate in the following respects, among others:

(a) The B&W simulator could not simulate the presence of a "two-phase" mixture, i.e., steam and water, in the reactor coolant system, and therefore was incapable of simulating many loss-of-coolant accidents;

(b) . The B&W simulator failed to simulate the Davis-Besse September 1977 incident, or any transient in which a

pilot-operated relief valve failed to close or in which the reactor coolant system pressure dropped as pressurizer water level rose, even though the NRC knew, based on its investigation of the Davis-Besse transient and from the Reactor Safety Study (WASH-1400), that such a transient was part of a class of likely small-break accidents, see paragraph 41(b) supra;

(c) Operators were given insufficient instruction in saturation conditions;

(d) The B&W simulator training program did not use the actual operating procedures for TMI-2 supplied by B&W.

1&3 - add'l invest. needed

70. GPU and Met-Ed relied on the proper implementation by the NRC of its regulations regarding operator training to assure that the TMI-2 operators were prepared to operate the plant safely, and they relied specifically on the fact that the TMI-2 operators had scored above the national average for all operators who had passed the NRC licensing test. If the NRC had performed its investigation of the Davis-Besse incident required by regulations with due care and had reviewed B&W topical reports and small-break loss-of-coolant accident analyses with due care, it would have known that the training programs it approved did not reflect what the NRC knew or should have known were actual operating conditions and potential safety problems of B&W plants.

EFFECTS OF NRC NEGLIGENT REVIEW AND APPROVAL

5. avoid invest, add'l - check & format w/ R comments

71. If the NRC had exercised due care in reviewing and evaluating B&W submissions and had recognized their failure to comply with the NRC General Design Criteria and other regulations, and if the NRC had required B&W to submit complete and correct analyses of transients, GPU and Met-Ed would have had the equipment, instrumentation, procedures and training reasonably needed to avoid the accident on March 28, 1979 and the accident would not have occurred.

4. cogitum — B&W submission too vague

72. If the NRC had complied with NRC regulations and exercised due care in reviewing and evaluating B&W submissions, the NRC would have anticipated the circumstances under which a small-break loss-of-coolant accident at the top of the pressurizer would result in a pressure drop in the reactor coolant system while the water level rose in the pressurizer and the NRC would have required B&W equipment, analyses, procedures and training to deal properly with such conditions.

4. "

73. The negligent review and approval by the NRC of B&W equipment, analyses, procedures and training, which it knew or should have known were deficient and not in compliance with NRC regulations, was a proximate cause of the March 28, 1979 accident at TMI-2.

DAMAGES

4. "

74. As a proximate result of the foregoing, the March 28, 1979 accident at TMI-2 occurred and caused and will continue to cause claimants to suffer damages and losses in the following respects, together with other items of damage incidental thereto.

(a) Claimants have incurred and will continue to incur expenses for decontamination and debris removal -- $1,000,000,000.

(b) Claimants have incurred and will incur expenses for repair or replacement of damaged and defective plant and equipment, refueling, upgrading of equipment and systems, retraining operators and additional expenses for personnel and consultants necessitated by the accident -- $430,000,000.

(c) In order to meet the needs of their customers for electric power, claimants have had to purchase and continue to purchase from other utilities additional capacity and energy and have had to operate and continue to operate their

less cost-efficient plants longer than they otherwise would have in order to replace the loss of capacity and energy resulting from the March 28, 1979 accident at TMI-2, which caused the shutdown of TMI-2 and prevented the restart of TMI-1 -- $1,590,000,000.

(d) Claimants have lost and will continue to lose revenues based on the removal from the rate base of the capital invested in TMI-2, which revenues they would have otherwise earned during the period for which that unit is not in the rate base -- $950,000,000.

(e) Claimants have had to and will continue to incur increased borrowing of capital and at higher rates of interest than they would have otherwise incurred were it not for the accident -- $40,000,000.

(f) In the event that claimants are not able to restore TMI-2 to operation, claimants will lose all of the capital invested in TMI-2 -- $800,000,000.

A schedule of damages is attached hereto as Appendix A.

LEGAL AUTHORITY

75. The NRC has recognized in its regulations that it has the specific duty of "notifying licensees regarding generic problems so as to achieve appropriate precautionary or corrective action." 10 C.F.R. § 1.64 (1977).

76. Where a government agency has a statutory duty to warn, or undertakes to warn and thereby induces a reliance by a private party on government action, the government is liable under the Federal Tort Claims Act if it fails to provide such warning. *Indian Towing Co., Inc. v. United States*, 350 U.S. 61 (1955), *Ingham v. Eastern Air Lines, Inc.*, 373 F.2d 227 (2d Cir.), *cert. denied*, 389 U.S. 931 (1967); *Gill v. United States*, 429 F.2d 1072 (5th Cir. 1970).

77. In licensing a reactor for operation, the NRC decides that

> "a reactor whose ECCS [emergency core cooling system] meets the criteria will control a LOCA [loss-of-coolant accident] and is, therefore, safe for operation . . . ."

Union of Concerned Scientists v. Atomic Energy Commission, 499 F.2d 1069, 1087 (D.C. Cir. 1974). A government agency such as the NRC which fails to exercise due care in its licensing and fails to comply with its regulations is liable under the Federal Tort Claims Act. Griffin v. United States, 500 F.2d 1059 (3d Cir. 1974); Ingham v. Eastern Air Lines, 373 F.2d 227 (2d Cir.), cert. denied, 389 U.S. 931 (1967); United Airlines, Inc. v. Weiner, 335 F.2d 379 (9th Cir.), cert. dismissed sub nom., United Airlines Inc. v. United States, 379 U.S. 951 (1964); Hartz v. United States, 387 F.2d 870 (5th Cir. 1960).

(continued on page 31)

WHEREFORE, claimants pray for an award in the amount of $4,010,000,000.

Dated: New York, New York
December 8, 1980

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Appendix A

SCHEDULE OF DAMAGES

Description	Loss to Date	Future Loss[1]	Total
Decontamination and debris removal.	$185,000,000	$815,000,000	$1,000,000,000
Repair or replacement of damaged and defective equipment, upgrading of equipment and systems and additional expenses for personnel and consultants necessitated by the accident, retrofitting, retraining operators.	0	430,000,000	430,000,000
Increased cost of electric power due to purchase of power from other utilities and costs of operating claimants' less cost-efficient plants.	465,000,000	1,125,000,000	1,590,000,000
Loss of revenue on capital invested in TMI-2 removed from rate base	165,000,000	785,000,000	950,000,000
Increased cost of borrowing	15,000,000	25,000,000[2]	40,000,000
Capital invested in TMI-2			800,000,000

1/ Future losses are calculated on the assumption that TMI-2 and TMI-2 will resume operation on the following dates:

TMI-1	January 1, 1982
TMI-2	January 1, 1988

2/ Assumes award and payment on December 31, 1982.