UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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In the matter of: :

METROPOLITAN EDISON COMPANY : Docket No. 50-289 (Restart)

(Three Mile Island Unit 1) :

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25 North Court Street,
Harrisburg, Pennsylvania

Wednesday, January 21, 1981

Evidentiary hearing in the above-entitled matter was resumed, pursuant to adjournment, at 9:06 a.m.

BEFORE:

IVAN W. SMITH, Esq., Chairman,
Atomic Safety and Licensing Board

DR. WALTER H. JORDAN, Member

DR. LINDA W. LITTLE, Member

8101260431

APPEARANCES:

On behalf of the Licensee, Metropolitan Edison Company:

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THOMAS A. BAXTER
Shaw, Pittman, Potts and Trowbridge,
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On behalf of the Commonwealth of Pennsylvania:

ROBERT ADLER, Esq.
Assistant Attorney General,
505 Executive House,
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Nuclear Engineer

JAMES M. CUTCHIN, IV, Esq.
JAMES TOURTELLOTTE, Esq.
Office of Executive Legal Director,
United States Nuclear Regulatory Commission,
Washington, D. C.

C O N T E N T S

P R O C E E D I N G S

CHAIRMAN SMITH: Is there any preliminary business?

(No response.)

CHAIRMAN SMITH: Mr. Cutchin, should we have a cross-examination plan from the staff?

MR. CUTCHIN: You should, Mr. Chairman, but I am only going to have three or four questions. And to be candid with you, my witness only arrived about 10:00 o'clock last night, and I have not had an opportunity to reduce it to writing.

CHAIRMAN SMITH: Okay.

MR. BAXTER: Licensee is recalling to the stand Robert W. Keaten, Michael J. Ross, and Robert C. Jones, Jr., all of whom have previously testified.

Whereupon,

ROBERT W. KEATEN

MICHAEL J. ROSS

ROBERT C. JONES, JR.

recalled as a witnesses by counsel for the Licensee, Metropolitan Edison Company, having first been previously duly sworn by the Chairman, were examined and testified further as follows:

FURTHER DIRECT EXAMINATION:

BY MR. BAXTER:

Q Gentlemen, I call your attention to a document which bears the caption of this proceeding. It is dated September 15, 1980. It is entitled "Licensee's Testimony of Robert W. Keaten, Michael J. Ross, and Robert C. Jones, Jr., in Response to UCS Contention Number 7, ANGRY Contention Number 5B, and Sholly Contention Number 6B (Detection of Inadequate Core Cooling)."

Is the material associated with your name in this document, including attached statement of qualifications, testimony which you have prepared or had prepared under your direct supervision for presentation at this hearing, Mr. Jones?

A (WITNESS JONES) Yes.

Q Mr. Keaten?

A (WITNESS KEATEN) Yes.

Q Mr. Ross?

A (WITNESS ROSS) Yes, it is.

Q Do you have any changes or corrections to make to your testimony, Mr. Jones?

A (WITNESS JONES) No.

Q Mr. Keaten?

A (WITNESS KEATEN) No.

Q Mr. Ross?

A (WITNESS ROSS) No.

Q Is the testimony true and accurate to the best of your knowledge and belief?

A (WITNESS JONES) Yes.

A (WITNESS KEATEN) Yes.

A (WITNESS ROSS) It is.

MR. BAXTER: Mr. Chairman, I move the receipt into evidence of the testimony and ask that it be physically incorporated into the transcript as if read.

CHAIRMAN SMITH: With no objections, the testimony is received.

(The complete testimony follows.)

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

LICENSEE'S TESTIMONY OF

ROBERT W. KEATEN, MICHAEL J. ROSS AND ROBERT C. JONES, JR.

IN RESPONSE TO UCS CONTENTION NO. 7,

ANGRY CONTENTION NO. V(B) AND SHOLLY CONTENTION NO. 6(b)

(DETECTION OF INADEQUATE CORE COOLING)

OUTLINE

The purposes and objectives of this testimony are to respond to UCS Contention 7 and ANGRY Contention V(B), which assert that instrumentation to directly indicate core coolant level is required. The testimony also responds to Sholly Contention 6(b) and certain NRC staff positions, which assert that additional instrumentation for the detection of inadequate core cooling should be installed. It is shown that core water level instrumentation is not required to assure adequate core cooling and that such instrumentation would not provide the basis for any incremental corrective action. Operator training, procedure revisions and instrumentation changes necessary to evaluate core cooling conditions and avoid the onset of inadequate core cooling are discussed. The testimony continues with a description of operating guidelines which have been developed and implemented to determine and respond to an inadequate core cooling situation should such occur. It is shown that unambiguous, easy-to-interpret, anticipatory indication of inadequate core cooling and the necessary instructions to take appropriate action to assure adequate core cooling conditions are provided.

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INTRODUCTION

This testimony, by Mr. Robert W. Keaten, GPU Manager of Systems Engineering, Michael J. Ross, TMI-1 Supervisor of Operations, GPU, and Robert C. Jones, Jr., Supervisory Engineer, ECCS Analysis Unit, Babcock & Wilcox Company, is addressed to the following contentions:

UCS CONTENTION NO. 7

> NRC regulations require instrumentation to monitor variables as appropriate to ensure adequate safety (GDC 13) and that the instrumentation shall directly measure the desired variable. IEEE 279, § 4.8, as incorporated in 10 CFR 50.55a(h), states that:
>
> > To the extent feasible and practical protection system inputs shall be derived from signals which are direct measures of the desired variables.
>
> TMI-1 has no capability to directly measure the water level in the fuel assemblies. The absence of such instrumentation delayed recognition of a low water level condition in the reactor for a long period of time. Nothing proposed by the staff would require a direct measure of water level or provide an equivalent level of protection. The absence of such instrumentation poses a threat to public health and safety.

ANGRY CONTENTION NO. V(B)

> The NRC Order fails to require as conditions for restart the following modifications in the design of the TMI-1 reactor without which there can be no reasonable assurance that TMI-1 can be operated without endangering the public health and safety:
>
> (B) Installation of instrumentation providing reactor operators direct information as to the level of primary coolant in the reactor core.

SHOLLY CONTENTION NO. 6(b)

It is contended that the short-term actions identified in the Commission's Order and Notice of Hearing dated 9 August 1979 are insufficient to provide the requisite reasonable assurance of operation without endangering public health and safety because they do not include the following items:

b. Completion of the installation of instrumentation for the detection of inadequate core cooling.

RESPONSE TO UCS CONTENTION NO. 7 AND ANGRY CONTENTION NO. V(B)

BY WITNESS JONES:

UCS Contention 7 asserts that since TMI-1 does not have instrumentation available to measure the water level in the fuel assemblies there is a threat to public health and safety. The lack of such instrumentation is also presented as being a violation of NRC General Design Criterion 13 and 10 CFR Part 50, Section 50.55a, Paragraph (h). The contention is not valid. (ANGRY Contention V(B) makes a similar assertion and is invalid as well.)

The goal of measuring the water level in the core, or any similar variable, would be to assure that the core is adequately cooled. To achieve this goal for power operation the safety analyses which have been performed for TMI-1 defined the parameters which must be monitored. These "desired" variables are then directly measured and input to the Reactor Protection System (RPS) and/or the Engineered Safety Features Actuation

System (ESFAS). They are reactor power, reactor coolant pressure, temperature and flow, and containment pressure.

The RPS serves, in part, to protect the reactor core by initiating a reactor trip upon the following conditions:

(1) Reactor power exceeds a maximum level.

(2) Reactor power exceeds a maximum level as determined by reactor coolant flow.

(3) Reactor coolant temperature exceeds a maximum level.

(4) Reactor coolant pressure exceeds a maximum level.

(5) Reactor coolant pressure falls below a minimum level.

(6) Reactor coolant pressure falls below a minimum level determined by reactor coolant temperature.

(7) Containment pressure exceeds a maximum level.

That is, for power operation the variables appropriate to assure adequate safety have been defined and these parameters are directly measured and input to the protection system. Water level in the core is not a part of the required instrumentation and no incremental protection system action can be identified based on such indication. There is no known sequence of events which, from a power operation condition, could result in a low water level in the reactor vessel which would not be preceeded by a reactor trip from the RPS.

Should an accident such as a loss of coolant accident (LOCA) occur, the ESFAS is designed to actuate the Emergency Core Cooling System (ECCS) upon the following conditions:

(1) Reactor coolant pressure falls below a minimum level.

(2) Containment pressure exceeds a maximum level.

The ECCS then provides sufficient inventory to assure that adequate core cooling is maintained. (Note that during the TMI-2 accident the RPS functioned as designed, tripping the reactor on high reactor coolant pressure promptly following the initiating event, loss of feedwater. The ESFAS also functioned as designed, actuating the ECCS on low reactor coolant pressure.)

Following reactor trip and engineered safeguards actuation, protection system input and actuation requirements are no longer applicable. Obviously, however, the goal of assuring adequate core cooling is ongoing and is achieved by maintaining subcooled conditions in the Reactor Coolant System (RCS) or, in the absence of such conditions, by providing sufficient reactor coolant inventory.

Reactor coolant subcooling is assessed by monitoring system temperature and pressure. The indications can be used directly in combination with steam table data to determine system status relative to saturation, or the indications can be processed by a saturation margin meter to display the same information. The temperature instrumentation utilized for this function is located in the RCS hot legs and saturated conditions will occur in the hot legs before core fluid conditions degrade below those necessary for adequate core cooling. The existence of a saturated condition is a direct indication of an abnormal condition which requires use of ECCS. Pursuant to current criteria, therefore, maximum achievable ECCS flow will be provided. No additional action based on core water level instrumentation can be identified. (At TMI-2, saturated conditions were indicated several minutes after the reactor tripped and if sufficient injection flow had been maintained the core would have been adequately cooled and not damaged.)

If an accident occurs which results in core uncovery, superheated reactor coolant conditions would be indicated by core exit thermocouples and the reactor coolant hot leg temperature instrumentation. As explained below, this indication would allow time for corrective action before the limits of 10 CFR 50.46 would be exceeded. Again, core water level instrumentation would not provide a basis for any additional action. (During the accident at TMI-2, indications of superheating in the RCS were available.)

<u>BY WITNESS KEATEN</u>:

Instrumentation for the detection of inadequate core cooling was among the subjects considered by the NRC Office of Nuclear Reactor Regulation TMI-2 Lessons Learned Task Force in its Status Report and Short-Term Recommendations (NUREG-0578). The Task Force concluded with the following positions:

1. Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, including Natural Circulation" (see Section 2.1.9 of this appendix).

   In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that it is not to be used exclusive of other related plant parameters.

2. Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Before addressing Licensee's response to these positions, it is important to recognize when inadequate core cooling occurs.

BY WITNESS JONES:

As I have already indicated, in a depressurization event the RCS must first reach saturation conditions before there is any danger of inadequate core cooling. If the RCS inventory subsequently is reduced and uncovery of the core begins, temperatures in the uncovered region will increase, causing superheating of the steam. Heretofore, the term "inadequate core cooling" has generally been applied whenever the core is not covered by either liquid coolant or a two-phase mixture, thus resulting in superheated conditions being indicated by the core exit thermocouples. However, core uncovery by itself does not mean that the core is being inadequately cooled. For example, design basis small-break LOCA analyses result in some core uncovery without any clad damage occurring. Furthermore, the criteria for adequate core cooling for LOCA's are those contained in 10 CFR 50.46. Therefore, for purposes of this testimony, inadequate core cooling is considered to exist when the fuel is uncovered to an extent and/or for a period of time such that the limits of 10 CFR 50.46 would be exceeded.

BY WITNESS KEATEN:

In order to avoid the onset of inadequate core cooling conditions, specific steps have been taken at TMI to ensure that the operators understand the requirements for adequate core cooling and are provided the necessary information to evaluate core coolant conditions.

First, as Mr. Ross discusses below, the operator accelerated retraining program has included specific training in heat transfer and fluid dynamics, plant operating characteristics, plant response to transients and guidance for operator response to loss of coolant accidents (see Section 6 of the Restart Report).

Second, plant procedures have been revised to emphasize the importance of maintaining an adequate saturation margin in the reactor coolant system and to provide guidance for steps to be taken if the saturation margin is less than the required value. The procedures specify the conditions under which high pressure injection flow may be reduced and specify the conditions which require restoration of appropriate HPI flow.

Third, as recommended by the staff's TMI-2 Lessons Learned Task Force, a new meter will be installed in the control room, prior to restart, which directly indicates the margin to saturation conditions in the reactor coolant system (see Restart Report section 2.1.1.6), i.e., the margin between the actual primary system temperature and the saturation temperature for the existing primary system pressure. The temperature margin will be displayed in the control room, and an alarm will be initiated if the margin falls below a pre-set value. Redundancy will be provided by computing the saturation temperature margin independently for each reactor coolant loop. The plant computer, using the same parameters, can also indicate the saturation pressure and temperature, and saturation pressure and temperature margins, for logging and alarm. This new instrumentation will aid the operator in taking action to maintain or re-establish the subcooling margin, and would also assist in the detection of the approach to inadequate core cooling.

Fourth, all 52 of the core exit thermocouples have been connected to read-out in the control room (see Restart Report section 2.1.1.6).

Fifth, an expanded range (120°F-920°F) will be provided for the RCS hot leg temperature measurement prior to restart so that the saturation meter can be used to detect the approach to inadequate core cooling outside the normal operating temperature range (see Restart Report Section 2.1.1.6). As Mr. Jones has stated, this instrumentation, along with the core exit thermocouples, will indicate superheated reactor coolant conditions and provide direct indication of core uncovery.

Finally, as discussed more fully below, a new emergency procedure has been written to define the use of the information available from the core exit thermocouples, RCS temperatures and the new saturation indicator in identifying when inadequate core cooling is approaching and to specify the operator action required to promptly enhance core cooling.

The training, procedures and instrumentation described assure that the operators take the following key actions during any approach to an inadequate core cooling condition:

1. Initiate high pressure injection;
2. Maintain steam generator level;
3. Trip the reactor coolant pumps if the engineered safety features actuation signal is initiated by low reactor coolant system pressure; and,

4. Monitor core exit thermocouple temperatures to assure that adequate core cooling exists.

No further action is required for design basis events.

BY WITNESS JONES:

For postulated events beyond the design basis, "inadequate core cooling" guidelines have been developed which define appropriate actions to prevent significant cladding damage and/or hydrogen generation. These guidelines are based on recognition of core uncovery and provide guidance to aid in prevention of a situation deteriorating to an inadequate core cooling condition. To develop these guidelines, a series of calculations was performed to develop a correlation between core exit thermocouple temperatures and peak cladding temperature. Using this correlation, two levels of operator actions were identified (see Figure 1).

For the initial level of elevated temperature conditions (Curve 1, Figure 1), the operator is instructed to take the following steps:

1. Start one Reactor Coolant Pump (RCP) per loop.
2. Depressurize operative Once Through Steam Generator(s) (OTSG(s)) to 400 psig as rapidly as possible.
3. Open the Pressurizer Power Operated Relief Valve (PORV), as necessary to maintain RCS pressure within 50 psi of OTSG pressure.

4. Continue cooldown by maintaining 100°F/hr decrease in secondary saturation temperature to achieve 150 psig RCS pressure.

If the thermocouple temperatures continue to rise above the higher predetermined temperatures, specified in the procedure (Curve 2, Figure 1), which indicate a further increase in fuel clad temperature, the operator is instructed to take the following additional actions:

1. Start all RCP's.
2. Depressurize OTSG(s) to atmospheric pressure.
3. Open the PORV to depressurize the RCS and allow Low Pressure Injection to restore core cooling.

This procedure is based upon a recognition that recovery at the higher pressure is unlikely, and that while depressurization will cause more immediate core voiding, in the longer term it will result in improved core cooling by increasing reactor coolant inventory.

BY WITNESSES KEATEN, ROSS AND JONES:

The instrumentation and procedures described above, as well as the training which Mr. Ross describes below, assure that, as recommended in Position 2 of the NRC Staff's TMI-2 Lessons Learned Task Force (quoted above), operators have unambiguous, easy-to-interpret indication of the approach to inadequate core cooling and the necessary guidance to take appropriate action to enhance adequate core cooling. Consequently, there is no need for additional instrumentation such as reactor vessel level indication -- for which no incremental operator or automatic action has been identified beyond those specified in response to presently monitored parameters. None of the current emergency procedures at TMI-1 require operator knowledge of reactor vessel water level.

BY WITNESS KEATEN:

The NRC staff to date has not recommended or required the installation of instrumentation to measure directly reactor vessel water level at TMI-1. The staff has explained this position as follows:

> The inclusion of instrumentation to measure directly the water level in the pressure vessel, i.e., the "water level in the fuel assemblies," was not conclusively known to be feasible or practical. In addition, other considerations which entered into the decision not to make direct measurement level instrumentation a requirement were:
>
> a) Other methods available or in use may be as good, if not better, and more reliable.
>
> b) There are uncertainties in the accuracy of the responses of level instrumentation under conditions where two phase fluids might be present in the vessel.
>
> c) The applicant and reactor vendor are in a better position to assess the instrumentation best suited to determine water level within the vessel for their plant.

(NRC Staff Response to Union of Concerned Scientists First Set of Interrogatories, March 7, 1980, response to Interrogatory 67.) The staff has also stated: "It is our opinion that the existing plant instrumentation at TMI-1 provides sufficient information to the operator to indicate reduced reactor vessel coolant level, core voiding, and deteriorated core thermal conditions." (NRC Staff Response to Union of Concerned Scientists Interrogatories (Second Set), response to Question 202.) Nevertheless, the NRC staff went on to say: "Although these instruments can provide sufficient information to detect adverse core conditions, the NRC Lessons Learned Task Force concluded that a more direct indication of inadequate core cooling could be provided to the operator." The staff now appears to take the position that additional instrumentation should be installed. (Staff Restart Safety Evaluation, NUREG-0680, at C8-21.) The nature of the additional instrumentation and the basis for any such requirement is not clear.

BY WITNESSES KEATEN AND JONES:

It has been suggested that indication of loss of hot leg subcooling does not provide advance warning of inadequate core cooling because it could also be symptomatic of a severe overcooling transient. The required action for both situations, however, is the same -- initiation of high pressure injection flow -- because both situations involve a reduction in reactor coolant volume. It has also been asserted that the measurement of superheated steam temperatures by the core exit thermocouples indicates inadequate core cooling imminent or already present. This indication, however, is anticipatory of inadequate core cooling, as explained earlier. It is also unambiguous and will not erroneously indicate inadequate core cooling.

We disagree, then, with the staff that additional instrumentation is required to detect inadequate core cooling. We also disagree with UCS and [redacted] that vessel water level indication should be installed at TMI-1.

BY WITNESS ROSS:

The NRC staff has stated that ". . . the detection of reduced coolant level or the existence of core voiding at TMI-1 can be readily determined with the saturation meter and other pre-existing [sic] instrumentation" and that "[t]he operator must be made aware of the available information and how to interpret it correctly." (Staff Restart Safety Evaluation, NUREG-0680, at C8-21.) The training provided to TMI-1 operators assures that the operators are aware of available information on the status of core cooling and how to interpret it correctly.

The operations personnel who will be on duty during TMI-1 power operation and would respond to any approaching inadequate core cooling condition include two licensed reactor operators (Control Room Operators), and two licensed senior reactor operators (one Shift Supervisor and one Shift Foreman). All of the licensed TMI-1 operators have completed the Operator Accelerated Retraining Program (OARP) described in Section 6 of the Restart Report. This training, along with the ongoing requalification training program, assure that operators will recognize and respond to reactor coolant conditions approaching and following saturation, using the instrumentation described above by Mr. Keaten. In addition, each shift will have immediately available a Shift Technical Advisor, who holds an engineering degree.

The OARP included approximately 200 hours of classroom lectures, discussions and working sessions, about 62 hours of which relate directly to the recognition of and response to approaching inadequate core cooling conditions:

- o Heat Transfer and Fluid Dynamics (16 hours) - theory of core cooling and various loss of core cooling transients, including indications, response and results.
- o Small-Break LOCAs (4 hours) - symptoms, indications and actions to be taken for inadequate core cooling and small break accidents.
- o Safety Analysis Workshop (28 hours) - interactive (students-instructors) problem analysis of system conditions to ensure recognition and response to approaching inadequate core cooling, and selection of cooling mode.

- Reactor Coolant System Elevations and Manometer Effects (4 hours) - theory of, and recognition and response to, manometric behavior during transients.
- TMI-2 Transient (4 hours) - cause of and response to gas/steam binding affecting core cooling.
- Procedure Review (4 hours) - indentification and explanation of, and responses required for, all steps in the procedures for natural circulation, forced cooling, all LOCA cases, OTSG tube rupture and loss of decay heat removal.

As Supervisor of Operations, I presented a two-hour training session which stressed the importance of using procedures and verifying key plant parameters, using specific examples of plant operational conditions.

The OARP also included control room and simulator training sessions to permit "hands on" application of the guidance and training provided to TMI-1 operators. The control room sessions included a review of the specific instrumentation and information available in the TMI-1 Control Room to build an association of the operational concepts and guidance presented in the classroom with the actual system controls.

Training on the B&W simulator was a part of the OARP (four days per shift crew) and is part of the ongoing operator requalification training program (one week per each shift crew per year). The simulator training provides the opportunity for the operators to participate in plant operations as control room operators and as supervisors of control room operators. The simulator has the capability of introducing over 60 individual casualties in reactor plant systems. The individual casualties can be combined to create multiple failure accidents or the instructor may fail equipment sequentially. Thus, the simulator gives the operator the opportunity to practice his training and diagnostic skills on complex problems.

These problem situations on the simulator include situations where core cooling either approaches or reaches saturated conditions, requiring the operators to recognize and rectify the degraded conditions and assure adequate core cooling. For example, one problem presented during the OARP included a small-break LOCA sequence in which, following HPI actuation, the throttling criterion of a 50°F subcooling margin was reached, with stable reactor conditions. The simulator instructors then introduced a larger break LOCA. The shift teams were required to diagnose this condition, and in all cases prevented an approach to inadequate core cooling by manually re-establishing full HPI flow before the primary system reached saturation conditions.

In the simulator training, the operator is required to demonstrate satisfactorily his ability to: (1) use and understand applicable emergency procedures; (2) properly manipulate the controls to place and maintain the plant in a safe configuration; (3) use available alarms and indications to evaluate and control the transient; (4) explain plant response;

and (5) explain plant conditions and recommend subsequent actions to his supervisor.

In addition to weekly quizzes, the OARP included a written and oral evaluation of the trainees, administered by an independent consultant, which was equivalent to an NRC initial licensing examination. TMI-1 licensed operators who have successfully completed the OARP will also be required to pass an NRC-administered oral and written license examination.

Following the OARP, the senior reactor operators and other plant management personnel participated in a five-day decision analysis training program. The program utilized a workshop technique in which scenarios were presented, based on actual and postulated plant responses, and personnel were called upon to diagnose plant symptoms and to identify appropriate operator responses. Follow-up discussions were then conducted to provide individual-to-individual team member feedback and to assure understanding of the problem exercise by all personnel.

Licensee's ongoing operator requalification training program requires every licensed operator to devote one week out of every six to training. This program, like the other training I have described, includes many elements which relate directly to the operators' ability to recognize and respond appropriately to an approaching inadequate core cooling condition.

All of this training emphasizes that the operators must maintain adequate reactor coolant saturation margin. The main points which are stressed repeatedly include:

- o Utilize procedures.
- o Verify critical parameters.
- o Proper use, interpretation of and response to saturation margin meter, core exit thermocouple, and hot leg and cold leg temperature indications.
- o Criteria for throttling HPI flow.

This training assures that the instrumentation described above will be adequate for operators to detect and respond correctly to conditions of reduced coolant volume or core voiding.

BY WITNESSES KEATEN AND JONES:

In summary, adequate instrumentation currently exists at TMI-1 to assess core cooling conditions. Even if core water level instrumentation were available, no incremental action can be specified based on such an indication beyond those automatically initiated or required of the operator in response to presently monitored variables. The lack of core water level instrumentation does not, therefore, violate regulatory requirements or pose a threat to public health and safety.

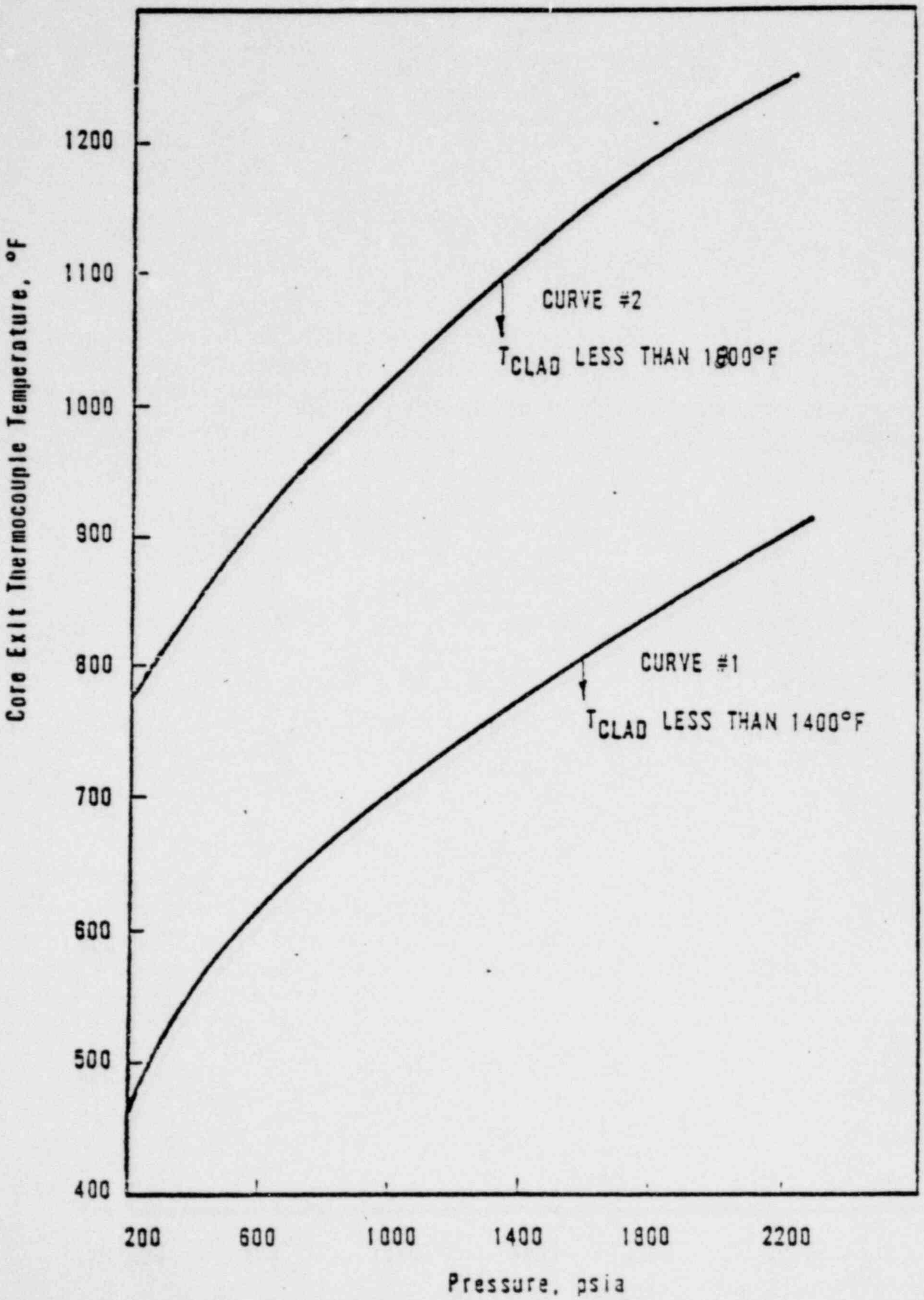
RESPONSE TO SHOLLY CONTENTION NO. 6(b)

BY WITNESS KEATEN:

As explained above in the response to UCS Contention 7 and ANGRY Contention V(B), Licensee has already completed installation of adequate instrumentation for the detection of inadequate core cooling.

FIGURE 1

CORE EXIT THERMOCOUPLE TEMPERATURE FOR INADEQUATE CORE COOLING



ROBERT W. KEATEN

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Education: B.S., Physics, Yale University, 1957. Post-Graduate and Professional Courses in Mathematics, Engineering and Business, UCLA, 1960-1972.

Experience: Manager, Systems Engineering Department, GPU Service Corporation, April 1978 to present. Responsible for the development and application of specialized analytical skills in such areas as nuclear core reloads and fuel management; plant dynamic and safety analysis; system generating plant process computers; control and safety systems analysis, and analysis of plant operating performance for nuclear and fossil plants. Served as Deputy Director of Technical Support at Three Mile Island during the post-accident period.

Program Manager, Light Metal Fast Breeder Reactor Technology, Atomics International Division of Rockwell International, 1974 to 1978. Managed research and development programs performed for U.S. Department of Energy, including programs in reactor physics, safety and component development.

Manager of Systems Engineering, Light Metal Fast Breeder Reactor Program, Atomics International Division of Rockwell International, 1968 to 1974. Responsible for performance of safety analyses, development of safety criteria and development of instrumentation, control and safety systems design.

American Representative to the OECD Halden Reactor Project in Norway, 1965-1968. Participated in research on nuclear fuel performance, application of digital computers to nuclear reactors, and on development and application of in-core instrumentation.

Supervisor of Engineering, Sodium Reactor Experiment, Atomics International, Division of Rockwell International, 1962-1965. Responsibilities included analysis and measurement of the nuclear heat transfer and hydraulic parameters of the reactor core and process systems; specification and installation of nuclear and process instrumentation; design and installation of new control systems.

Senior Physicist, Sodium Reactor Experiment, Atomics International, Division of Rockwell International, 1959-1962. Performed measurements and analyses of the nuclear and thermal parameters of the reactor.

Experimental Physics Group, DuPont Savannah River Plant, 1957-1959. Performed measurements and calculations of the nuclear parameters of the reactor lattices.

Honors and Professional Affiliations:

Member of the Nuclear Power Plant Standards Steering Committee of the American Nuclear Society.

Member and past Chairman of the LMFBR Design Criteria (ANS-54) Standards Committee of the American Nuclear Society.

Registered Professional Engineer (Nuclear Engineering), California.

Publications:

"Analysis of TMI-2 Sequence of Events Operator Response," presented to a special session of the American Nuclear Society Conference, San Francisco, November 1979; and to Edison Electric Institute Conference, Cleveland, October 1979.

"The Role of Instrumentation in the TMI-2 Accident," presented at the American Nuclear Society Conference, June 1980.

Safety and Environmental Aspects of Liquid Metal Fast Breeder Reactors" 35th Annual American Power Conference, Chicago, Ill., May 1973.

"Safety Aspects of the Design of Heat Transfer Systems in LMFBR's" International Conference on Engineering of Fast Reactors for Safe and Reliable Operation, Karlsruhe, Germany, October 1972.

"Safety Criteria and Design for an FBR Demonstration Plant," ASME Nuclear Engineering Conference at Palo Alto, Calif., March 1971.

"Evaluation of Thermocouples for Detecting Fuel Assembly Blockage in LMFBR's," American Nuclear Society Annual Meeting, Los Angeles, California, June 1970.

"A Mathematical Model Describing the Static and Dynamic Instability of the SRE Core II," Reactor Kinetics and Control, AEC Symposium Series 2. (Also published as NAA-SR-8431.)

"Reactivity Calculations and Measurements at the SRE," ANS Topical Meeting: Nuclear Performance of Power-Reactor Cores, September 1963.

"Measurement of Dynamic Temperature Coefficients by Forced Oscillations in Coolant Flow," Trans-American Nuclear Society 5, No. 1, June 1962.

"Analysis of Power Ramp Measurements with an Analog Computer," Trans-American Nuclear Society 5, No. 1, June 1962.

"Reflected Reactor Kinetics," NAA-SR-7263.

Many other reports covering analytical and experimental work.

MICHAEL J. ROSS

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Education: U.S. Navy Nuclear Power School, 1961. U.S. Navy Nuclear Power Prototype School, 1961.

Experience: Supervisor of Operations, Three Mile Island Unit 1, Metropolitan Edison Company, 1978 to present. Responsible for directing the day-to-day operation of the plant to ensure compliance with the conditions of the plant operating license and technical specifications, including supervision of the Radioactive Waste Processing and Shipment Group and coordination of operations and related maintenance activities with the Superintendent of Maintenance.

Shift Supervisor, Three Mile Island Unit 1, Metropolitan Edison Company, 1972 to 1978. Responsible for the management of all operations and maintenance activities, including the manipulation of any controls, equipment or components in physical plant systems on his shift.

Shift Foreman, Three Mile Island Unit 1, Metropolitan Edison Company, 1970 to 1972. Responsible for performance of various pre-operational activities, including preparation of procedures and start-up equipment checks.

Reactor Plant Technician, Saxton Nuclear Experimental Corporation, 1968 to 1970. Held position of reactor operator; additionally, was responsible for training operations staff.

U.S. Navy, 1960 to 1968. Positions held include reactor operator aboard USS Haddo, Instructor at the Nuclear Power Training Unit, and AEC Field Representative at the Nuclear Power Training Unit

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Experience: June 1971-June 1975: Engineer, ECCS Analysis Unit, B&W. Performed both large and small break ECCS analyses under both the Interim Acceptance Criteria and the present Acceptance Criteria of 10 CFR 50.46 and Appendix K.

June 1975-Present: Acting Supervisory Engineer and Supervisory Engineer, ECCS Analysis Unit, B&W. Responsible for calculation of large and small break ECCS evaluations, evaluations of mass and energy releases to the containment during a LOCA, and performance of best estimate pretest predictions of LOCA experiments as part of the NRC Standard Problem Program. Involved in the preparation of operator guidelines for small-break LOCA's and inadequate core cooling mitigation.

MR. BAXTER: And I would note at this point in the proceeding UCS Contention Number 7 and Sholly Contention 6B have been withdrawn by those parties, but we will present the testimony as originally filed with the Board.

I am going to have some questions on rebuttal now to companion staff testimony filed on the Contentions.

From Mr. Lon B. Phillips there are two pieces of testimony which we expect the staff to offer. One piece is nine pages in length and was filed, to my memory, approximately December 1, 1980. And the second piece of testimony, which consists of five pages, was filed on December 22. I assume they will be inserted into the record in that order.

And I will refer to the larger piece of testimony as Mr. Phillips' "first filing," and the shorter atestimony as the "second filing."

BY MR. BAXTER: (Resuming)

Q Looking at the first piece of testimony filed by Mr. Phillips, at page 3 he provides the following definition of inadequate core cooling: "When the two-phase froth level begins to drop below the top of the core, the exposed fuel begins to heat up and will ultimately reach temperatures at which fuel damage occurs. This is inadequate core cooling."

Mr. Jones, giving us your understanding of that definition, would you contrast it with the one presented in your testimony and explain the basis for your explanation, please?

A (WITNESS JONES) Well, first off, I have trouble with the definition that Mr. Phillips has given, because it is somewhat wrong in the sense that it makes the statement that when the two-phase forth level begins to drop below the top of the core and implies that ultimately reached temperatures at which fuel damage occurs, that does not necessarily have to occur.

If you keep the level up high in the core with the steam cooling the rod relatively efficiently, you will not reach fuel damage temperatures.

But even with that, I still have another problem with it, in that really what is presented here, I am not sure what he is using as a definition because there are two definitions presented in my mind, reading this. The first definition is: "Inadequate core cooling is when the two-phase froth level begins to drop below the top of the core." And you can interpret that to be the inadequate core cooling definition that he is utilizing, or he may be us[illegible] the definition that where the two-phase froth level begins to drop below the top of the core such that fuel damage ultimately occurs, which would be indicative of the temperature on the order of 1400 degrees on the cladding.

But irrespective of which definition he is using, they are different from the one that we are using in this proceeding and in our testimony. We have defined inadequate core cooling basically based on the regulations as they exist today. 10 CFR 50.46 defines what acceptable or adequate core cooling is. That is, peak clad temperatures less than 2200 degrees, along with several other conditions that are implied within that criteria, in my mind, defines what the regulatory basis is for inadequate core cooling. That is, if you exceed those limits, you have inadequately cooled the core. That is what we have used.

Now, there is some -- there has been some interpretation or there has been some differences, apparent differences, noted between our definition of inadequate core cooling used in the testimony and the so-called "inadequate core cooling guidelines," because the inadequate core cooling guidelines themselves take action below the 10 CFR 50.46 threshold.

What has been done in developing those guidelines is we r'ated, or the thought process that went into it, was if you have an event which leads to high cladding temperatures beyond what you would normally calculate as a design basis, you know you have something that is beyond the design basis event happening. And the effort of the guideline is to try to present and provide cooling to the core prior to the cladding temperatures reaching the so-called inadequate core cooling limit of 2200 degrees.

CHAIRMAN SMITH: When you say "we," Mr. Jones, you are referring to your employer?

WITNESS JONES: Yes.

BY MR. BAXTER: (Resuming)

Q Turning now --

DR. JORDAN: I guess, Mr. Jones, I really did not understand that last statement. And I guess I just have to ask you to elaborate a little bit. I understood the words, but I do not catch the implications.

WITNESS JONES: Okay. Well, it does not make sense in my mind to put together a procedure that waits to do something when the cladding temperature has reached the so-called inadequate core cooling threshold that is defined in the regulations. So the procedure has been developed to take actions where the event is beyond what you would normally have calculated the event to progress, any small-break LOCA.

For example, the normal design basis calculations performed for the plant to show compliance with 50.46 show peak cladding temperatures on the order of 1100 degrees for a small-break LOCA.

DR. JORDAN: For a small-break LOCA.

WITNESS JONES: For a small-break LOCA.

DR. JORDAN: But that is not inadequate core cooling.

WITNESS JONES: That is correct. Now, the inadequate core cooling guidelines take actions -- start to take actions when you have in-core thermocouple readings which are indicative of a cladding temperature of approximately 1400 degrees.

DR. JORDAN: What guidelines do this?

WITNESS JONES: These are the inadequate core cooling guidelines.

DR. JORDAN: Whose?

WITNESS JONES: These are the ones that have been developed by B&W and submitted to Licensees for their utilization in the development of procedures.

DR. JORDAN: These are the guidelines for dealing with a small-break LOCA?

WITNESS JONES: These are guidelines for dealing with an inadequate core cooling situation, basically, irregardless or irrespective of how you got to it. What we are saying is we do not know of any condition on normal from, let's say, the secondary side of that which leads you to core uncovery. It should not occur. But you could use the guidelines should something happen that somehow the event should progress to that.

But they are mainly coupled to the small-break LOCA procedure for their general use.

DR. JORDAN: But I guess -- what is the significance of 1400? Are you saying that the guidelines themselves say that if the temperature is below 1400 then no action is required, but if it is above 1400 there is action required?

WITNESS JONES: That is correct.

DR. JORDAN: But that means you know when you have reached 1400.

WITNESS JONES: Yes. And what the guideline -- the way the guideline is constructed is, first off, there is an alert or something to alert the operator that the core is starting to uncover, which are superheated temperature indications on either the hot leg RTEs or the in-core thermocouples.

DR. JORDAN: Are you saying that before 1400 is reached or by the time 1400 is reached, the hot leg thermocouple will start to read above normal and is that -- at that time is it only starting to read above normal?

WITNESS JONES: No. We would expect a hot leg thermocouple to respond faster than that. But there are also the in-core thermocouples which will respond even faster than the hot leg RTEs. We have made use of both as an alert statement for the operator to place him out of the normal small-break procedure and into the inadequate core cooling procedure or guidelines. Now --

DR. JORDAN: Let's see. Are there 50-some in-core thermocouples? Do I remember properly, Mr. Keaten?

WITNESS KEATEN: There are 52.

DR. JORDAN: 52. And would you review for me where those are located? I know they are located around the various quadrants, but what actual elevations are those?

WITNESS KEATEN: They are located just above the top of the active core.

DR. JORDAN: Top of the active core, but below the grid plates and below the rods -- let's see -- above the active uranium pellets there is a space with springs and things like that; isn't that true? And are the thermocouples then located about where the active pellets end?

WITNESS ROSS: Our understanding is the thermocouple itself is approximately five inches above the active portion of the core.

DR. JORDAN: And by that you mean the uranium oxide pellets?

WITNESS ROSS: That is correct.

DR. JORDAN: All right.

WITNESS JONES: Now, I forgot where I was.

DR. JORDAN: I am sorry.

WITNESS JONES: So we have gotten to the inadequate core cooling guidelines by getting superheated temperature indications either by the in-core temperature, in-core thermocouples, or by the hot leg RTEs. That puts them into the inadequate core cooling procedure.

At that point in time, the operator is to continuously monitor the in-core thermocouple reading. And as there is a curve that has been developed to correlate the in-core temperature readings, in-core thermocouple temperatures to cladding temperatures --

DR. JORDAN: I did not get that.

WITNESS JONES: The in-core thermocouples will measure -- will not know the temperature of the fuel rod. It measures the temperature of the steam that is coming off the fuel rod. So we have developed a correlation between the steam temperature and the maximum fuel rod temperature and have developed a curve as a function of pressure, which is built into the procedure.

DR. JORDAN: Doesn't that correlation depend upon the rate of flow of the steam?

(Pause.)

WITNESS JONES: Yes, it does. But what you have is a cool boiling situation in the covered portion of the rods, which is essentially determining the steam flow rate coming up, and then that flow goes up and it is superheated by the heat transfer from the rod to the steam. And it superheats.

Now, if the decay heat is down, then you have a lower flow rate. But then again you get lower -- you don't need as much flow, because the uncovered portion of the rods also have less heat to remove.

DR. JORDAN: I see. So that you say there is a good correlation between the reading of the thermocouple in the steam which is being bathed in steam flowing by it from the heating -- steam being heated from the lower end of the core -- there is a good correlation between the temperature of that steam and the temperature of the cladding immediately adjacent, or the cladding where?

WITNESS JONES: It is the peak cladding temperature on the rod. Now, what has been done, in actuality, we have looked at various potential power -- axial flux shapes within the core and have studied them. And in order to make sure that the curve that we generate will bound the more or less a spectrum of power distributions within the core. And we put a number bound around it. And the design basis power shape that is used for LOCA analysis is also the same one that yields the curve which bounds the thermocouple temperatures.

DR. JORDAN: I see.

Mr. Baxter, I am sorry.

MR. BAXTER: That's all right. I wanted to ask one housekeeping question, if I might.

BY MR. BAXTER: (Resuming)

Q Mr. Jones, is this correlation between cladding temperature and reactor coolant pressure-temperature conditions reported in the document "Analysis Summary in Support of Inadequate Core Cooling Guidelines for a Loss of RCS Inventory" in attachment to question 45, supplement 1, part 1, of Licensee's Restart Report, which is exhibit -- Licensee's Exhibit 1?

A (WITNESS JONES) Yes, it is.

DR. JORDAN: I see. All right. Thank you. We may want to come back. I don't really go into any detail now, Mr. Baxter. I was trying to understand what he was saying. So you can go ahead.

MR. BAXTER: I appreciate the difficulty of listening to this orally, and I welcome your questions as we go along.

BY MR. BAXTER: (Resuming)

Q Turning to Mr. Phillips' second piece of testimony, the one filed on December 22, at page 2 of that testimony, Mr. Phillips criticizes reliance upon the saturation meter, because it does not distinguish between anomalous transients which can drain the pressurizer and cause primary loop saturation due to cooling, and shrinking of coolant versus a loss of coolant inventory which could lead to inadequate core cooling. It is the first paragraph in the answer to 22 on page 2.

MR. CUTCHIN: Mr. Chairman, for clarity, is Mr. Baxter now referring to the supplementary testimony of Mr. Phillips that is labeled "Question 1F and Question 2F"?

MR. BAXTER: That is correct.

MR. CUTCHIN: That was filed on January 16, and that is the document with which I provided the Board copies yesterday afternoon.

MR. BAXTER: I am sorry, I have my dates confused.

CHAIRMAN SMITH: So we have three Phillips' testimonies?

MR. BAXTER: No, we don't. We just have two. I was mistaken about December 22.

DR. JORDAN: Two Phillips and one Rubin testimony.

MR. CUTCHIN: That is correct, sir, two Phillips and one Mr. Rubin. Mr. Baxter correctly identified the filing date of the first Phillips, which was December 1. And the second Phillips is the document with which I provided the Board copies after we closed the record yesterday.

MR. BAXTER: I will refer to that as the "supplementary testimony."

DR. JORDAN: All right. Now, identify again then where we are.

MR. BAXTER: I am in the supplementary testimony, page 2, the first paragraph, the answer to the second question.

DR. JORDAN: At the end of the second question. All right.

MR. BAXTER: The first paragraph of the answer.

CHAIRMAN SMITH: What are the first words of that?

MR. BAXTER: "The saturation meter, while providing a basis for initial actions," it is the second sentence. That is the sentence I am looking at.

DR. JORDAN: All right.

CHAIRMAN SMITH: Now I have a document entitled "NRC Staff Testimony of Laurence Phillips," which was received -- undated -- which was received by our office December 2, 1980. Then I have a document entitled "NRC Staff Testimony of Lawrence Philliips, Supplementary Testimony to that of Laurence E. Phillips," filed December 1." And that is also undated. That is the one I received last night.

MR. BAXTER: That is correct. My first question to Mr. Jones went to the first piece you identified. The one I am about to ask goes to the second.

CHAIRMAN SMITH: I cannot find that sentence.

MR. BAXTER: It is on page 2 in the answer to question 2F.

CHAIRMAN SMITH: All right. Go ahead.

BY MR. BAXTER: (Resuming)

Q Let me start over. Mr. Phillips has criticized a reliance upon the saturation meter, because it does not distinguigh between anomalous transients which can drain the pressurizer and cause primary loop saturation due to cooling and shrinkage of primary coolant, versus a loss of coolant inventory which could lead to inadequate core cooling.

Now, to further confuse you I am going to be referring to an answer to our interrogatories, if you have those available. In explanation of why the operator needs to distinguish between these transients, Mr. Phillips stated in response to Licensee's interrogatory number 1, dated January 14 - the answer dated January 14, 1981, that for an anomalous overcooling transient, the operator should determine the cause of the overcooling and correct it.

Mr. Jones, in such a situation, what is the operator's first priority?

A (WITNESS JONES) The first priority of the operator is to initially control the transient and assure adequate core cooling.

Q Does he need to know that there is an overcooling event in order to accomplish that goal?

A (WITNESS JONES) No. He really does not. And the reason for that is if you have an overcooling transient which causes you to drain the pressurizer and get to saturation, you have a situation where you do not want to be. The operator's function at that time, whether it is a LOCA or an overcooling transient, is to restore primary system inventory and pressure by using the high-pressure injection pumps, the emergency core cooling systems, and that there is no need to know what the event is in order to do that.

If you saturate it in the primary loops, you have a situation -- it is just a situation where you do not want the plant to be in, and your goal is to provide the inventory to assure or to restore the subcooled margin in the plant.

Q When does the operator need to diagnose and correct the overcooling event, and how would he do it without vessel level indication?

A (WITNESS JONES) Well, first off, you do not really go and diagnose an overcooling event from a primary system parameter. You diagnose an overcooling event by looking at the secondary side: Do you have too much water, do you have too low a steam pressure in the steam generator? Those are the distinguishing characteristics of an overcooling transient.

In order to overcool, you have to be taking out too much heat, and it would be reflected within the steam generator parameters.

Now, as far as when he needs to do this, basically he has a few items he would want to try to prevent happening. First is to prevent an overfilling of the steam generator so that water flows into the steam lines. That is an indication he could only tell about how fast it is going by looking at the secondary side water level barometer.

If he had a steam line break, for example, he would want to assure that the broken steam line is isolated; that is, that the automatic system functions to stop delivering auxiliary feedwater or main feedwater to the broken steam generator. Again, that is strictly a secondary side-type diagnosis.

The only other indication -- or the only other goals he might have that would rely on primary system indication would be to not fill up the pressurizer and put water on the floor for an overcooling transient, where you put in so much water that he has filled the pressurizer and he now has water exiting the system from the -- through one of the relief valves.

Now, that can be done without a level indicator on the vessel, by looking at the subcooling margin in the plant and the pressurizer level. And again, the action is one that is covered within both the small-break guidelines and the inadequate -- and an overcooling-type transient diagnostic.

He would be allowed to throttle HPI, provided you have subcooled indication in the hot leg and the pressurizer water level approaching or tending to go high -- towards high and possible filling of the pressurizer. Then he could take that action without knowing whether it is an overcooling event or a small-break LOCA.

Q Returning again to Mr. Phillips' supplementary testimony at page 2, this time moving onto the second paragraph, he states that TMI-1 emergency procedure 1202-6B requires the operator to distinguish between an overcooling event and a small-break LOCA.

Mr. Ross, when in the procedure or scenario is diagnosis required?

A (WITNESS ROSS) Our procedures call for that diagnosis at approximately step seven under the immediate action. The actions preceding that are the actions Mr. Jones has discussed, basically verifying the reactor in fact had tripped, verifying that if high-pressure injection is initiated, we do have proper flows, if necessary, and high-pressure injection has initiated automatically, tripping reactor coolant pumps, isolating any obvious leaks, ensuring we have emergency feedwater.

At that point, the operator is given directions to evaluate this accident for an overcooling. He is also given the parameters to look at, those being the parameters that Mr. Jones has discussed, those being feedwater flow, steam pressure, steam generator levels.

Q Mr. Ross, do you agree with the observation Mr. Phillips makes at the bottom of the page that vessel level instrumentation would permit a much quicker and more reliable diagnosis of these transients, the transients being overcooling and loss-of-coolant accident?

A (WITNESS ROSS) No, I do not.

A (WITNESS JONES) I would like to add a little something to that answer, if I could. One of the things -- one of the problems with using a water level indicator to distinguish between an overcooling event and a small-break LOCA of each of the transients have very wide-ranging potential impacts on the primary system, depending on the nature of the overcooling event, for example, or the size of the small-break LOCA.

If you had, for example, a very small LOCA in the system, the primary system will remain liquid solid for a fairly long period of time or could remain liquid solid for several minutes, five-ten minutes, depending on the size of the break. So you could not use the fact that, "Well, my level has not dropped any," to say, "Ah-hah, I do not have a small-break LOCA," when in fact you might.

Additionally, if you have an overcooling event, you can end up, if it is a very rapid overcooling event with the potential of pulling a steam bubble within the reactor vessel head region. And a small-break LOCA ultimately and for the larger size small-break LOCAs, it will occur very quickly, will also lead to steam bubble formation in the reactor vessel upper head. So you cannot distinguigh it from that, you know, not strictly just off a level relationship or the fact that a level has dropped.

And finally, just an example of how it could be confusing to an operator, if you just take the St. Lucie event which occurred, where they pulled a bubble in the upper vessel head during a natural circulation cooldown of the plant, and that was not a LOCA situation. And yet, if they had a vessel level instrumentation -- instrument installed, if the operator was instructed to diagnose a LOCA based on level indicator, he would have then been considering St. Lucie to be a LOCA event when in fact it was not.

CHAIRMAN SMITH: Of course, we would assume that the operator would be aware of bubbles. I mean your operators do know about bubbles; don't they?

WITNESS ROSS: Yes, our operators do know about bubbles. As we have previously testified, they have been trained in the way the bubble reacted in Unit 2. And in particular, they have been trained in the St. Lucie incident itself. That training consisted of a couple of ways the first was immediate notification of the I&E bulletin of St. Lucie, and then later sit-down sessions in the classroom were instituted on the St. Lucie bubble and what the indications were that that staff saw.

So I would answer, "Yes," they have been trained in that particular incident and bubble-type incidents.

DR. JORDAN: You are saying now that the operator would recognize a bubble in the upper head, that the procedures include ways of recognizing and diagnosing a bubble. Could you point to the procedure that does that?

WITNESS ROSS: In particular, the St. Lucie incident involved a natural circulation cooldown, and that particular procedure has been changed in two steps. The first step is to put the cautions in. And I can give you the number. I will have to look it up, to tell you the truth, though. Put the cautions in on what St. Lucie saw.

A final version of that is coming up based on the B&W guidelines instituted on natural circulation cooldown, and it's expected to be through our final review chain very shortly.

MR. BAXTER: We will come back to St. Lucie again in a few minutes.

DR. JORDAN: All right.

BY MR. BAXTER: (Resuming)

Q Looking at the sentence at the bottom, that begins at the bottom and runs over -- bottom of page 2 and runs over to the top of page 3 of Mr. Phillips' supplementary testimony, he states that: "For a small-break LOCA, an orderly cooldown is required, but not necessarily for an overcooling transient."

Mr. Keaten, do you agree with this observation?

A (WITNESS KEATEN) No, I do not. And I do not really understand what Mr. Phillips is driving at here. If we define an overcooling transient in an analogous way to the way that we define a small-break LOCA as being outside the normal behavior of the plant, and then following any such overcooling behavior there is no question but what the plant would have to be shut down and remain shut down for a period of time while we evaluated the transient, and we did the necessary notifications to the NRC and where we assured ourselves that the plant was not damaged prior to the time of restart.

So whether it is a small LOCA, whether it is an overcooling transient, or whether it is any other type of unexpected plant upset, there is no question but what the plant has to be shut down.

DR. JORDAN: May I ask now for a definition. I understand that your definition of inadequate cooling disagrees with that of the staff. But now then, an overcooling event, do you also disagree with the staff as to what is meant by that? And are you saying that every overcooling event, according to your definition, requires shutdown?

WITNESS KEATEN: We are defining an overcooling event, Dr. Jordan, as being something that is outside of the normal expected response of the plant to a transient. I honestly do not know what the staff means by their definition. So I don't know whether there is a disagreement or not.

But we would not define an overcooling event as the type of small cooldown that we normally expect to get, for example, when the reactor trips. We would define it as an overcooling event only if it is outside of the normal expected behavior of the plant.

And yy point is anytime we get this type of deviation from normal behavior, there is no question in my mind but what they have to shut down the plant and do an evaluation prior to restart.

DR. JORDAN: Okay. So any overcooling -- well, how do you recognize an overcooling event? As a lack of pressurizer level?

WITNESS KEATEN: Well, you look at it in two different ways. From the standpoint of the immediate operator response, what he will see is the behavior of the primary cooling system, and he will see a drop in level in the pressurizer, and he will see a decrease in the reactor coolant system pressure.

And as Mr. Jones was saying, at that point in time, he will not know, and he does not need to know, whether he has an overcooling transient or whether he has a LOCA, because the actions, both the automatic and the manual actions, are exactly the same in terms of the ones that need to be taken immediately. And that is to automatically or manually initiate the high-pressure injection system and make sure it is working.

DR. JORDAN: But you said before that although he has a drop in pressure -- in pressurizer level -- and a drop in temperature, as long as it is within -- inside of the normal events, then this is not an overcooling event. And so he has to look around and say, "Now, is this part of a normal event, or is this an abnormal event?" And how does he know?

WITNESS KEATEN: He knows by the extent of the changes which occur. If it is -- if it is what I would call a "normal" overcooling event, the type of small cooldown that occurs associated with a reactor trip, he will get some drop in the pressurizer level, he will get some decrease in the reactor coolant system pressure but not enough to initiate the high-pressure injection.

And what the operator normally does in that case is to stop his letdown flow, and he may increase somewhat his makeup flow, but he still is running with the normal makeup letdown system. He is not relying on the emergency core cooling system at that point.

If it is a major overcooling event, then he will get a much more substantial drop in the level in the pressurizer and a much more substantial decrease in the reactor coolant system pressure. And he may in fact get an automatic initiation of the high-pressure injection. And this in fact has occurred at TMI.

DR. JORDAN: If he gets an ECCS signal, does that mean the reactor has to be shut down, taken cold?

WITNESS KEATEN: Yes, sir. I would say in the environment we are living in today, any time that we get an automatic initiation of the emergency core cooling system, we are facing a very careful evaluation of what happened prior to the time we restart.

DR. JORDAN: Okay. Thank you.

BY MR. BAXTER: (Resuming)

Q Mr. Keaten, I am not sure we got your Dr. Jordan's question. How does he diagnose the overcooling event from the secondary side?

A (WITNESS KEATEN) Yes, I am sorry. I answered only the first part of the question.

Once -- and again, I am talking about a significant overcooling event, and let me define it for the moment as one which results in automatic initiation of the emergency core cooling system. Once he has assured himself that the necessary immediate actions that must be taken, such as the fact that the high-pressure injection pumps have started, that he has the flow rate that he has required, that he has manually tripped the reactor coolant pumps as the procedures require him to, and has done those immediate actions necessary to protect the core, then, as Mr. Ross said, the procedures instruct him to then try to distinguish between whether he has a LOCA or an overcooling event.

And he does this not by looking at the primary system parameters, which will not help him, but by looking at the secondary system parameters, the flow rate, the steam generator level, and the steam pressure.

Q Continuing at the top of page 3 of Mr. Phillips' supplementary testimony, he states that: "In both cases -- that is, an overcooling event or loss-of-coolant accident -- vessel level meter, if available, would provide coordinating information to assist the operator in restoring the water solid primary system (possibly using the upper head vent) and the normal water level in the pressurizer."

Mr. Keaten, what coordinating information might be provided, and how would the operator use it?

A (WITNESS KEATEN) We have not to date been able to identify how this type of information could be used to help the operator, even if it were available.

As Mr. Jones pointed out, the actions that the operator needs to take under a normal loss-of-coolant condition or an overcooling event are relatively simple and are easily performed using the existing instrumentation. Even in the case where an event develops which is beyond the design basis, which is not a normal loss-of-coolant accident but it is something else, B&W has developed the inadequate core cooling guidelines which apply to this case. And the operator is instructed to take action based upon the readings of the in-core thermocouples, coupled with the saturation meter and the other installed instrumentation.

As Mr. Jones also pointed out, even if there were a reliable method of measuring the reactor vessel water level and such instrumentation were installed, we have not been able to find any use that the operator could make of that information.

Frankly, our objection to installing such instrumentation is based upon that fact. In the testimony that you have been hearing on the previous Contention having to do with the human-engineering review of the control room, in carrying out that human-engineering review, one of the guidelines which was suggested to us by our expert consultants in human-factors engineering was that the information which is presented to the operator should be restricted to the information that he will use, and that if used in general terms it is a violation of the human-engineering principles to present him information that he is not going to be able to use, because it tends to overload him and it gets him thinking about things that he is not going to use rather than thinking about those things that he can use.

And so in a situation like this, where we have information that somebody has suggested that might be of use but where we cannot find any use for it, then we are very reluctant to say that this is something that ought to be installed in the control room and presented to the operator in a fashion that may distract him from looking at the things that he should be looking at and that he does know how to use.

So our reluctanct to install such instrumentation even it were available, it is based on the fact that we want our operators to concentrate on the things that they have to do in responding to these events and to concentrate on the instrumentation that we have taught them to use in order to determine what actions need to be taken.

CHAIRMAN SMITH: Mr. Keaten, in that answer you began by saying, as you concluded, that you did not know what use could be made of the level meter. Then you explained that the procedures in place do not rely upon it. And then you came back to, again, you do not know what use could be made of it and you do not want useless information.

But I do not know if you made it clear that you have looked at established procedures to see if they would fit in or if you have begun a new -- a new conceptual look at emergency procedures to see if a redesign of the procedures could make use of the water level indicator.

MR. BAXTER: Mr. Smith, that was going to be one of my questions. Let me ask it now, if I might.

CHAIRMAN SMITH: The reason I ask it now is I did not understand if this question included that or didn't. The emphasis seemed to be on established procedures, and they do not fit in. But that is fine. Clarify it anyway.

BY MR. BAXTER: (Resuming)

Q At page 4 of his testimony Mr. Phillips make the assertion that Licensee has not submitted any analyses or evaluation addressing additional operator actions which could be taken to prevent core uncovery for small-break LOCAs, if information were available for prompt diagnosis of the condition."

Would you review for us what analyses and evaluation Licensee or B&W has submitted to the staff on this issue? Mr Jones or Mr. Keaten?

A (WITNESS JONES) Let me start out first -- go back to the original creation of the guidelines. Back around last May, of 1979, the small-break guidelines were developed. Various revisions were made to those guidelines over the next several months to incorporate things like reactor coolant pump trip and other items that developed as we took a closer and closer look at these guidelines and tailored them to be more plant-specific.

Around November of 1979 the inadequate core cooling guidelines were developed based on existing instrumentation as required by 0578. But there was also another part of 0578 which said to look for additional instrumentation that could be utilized for the detection of inadequate core cooling, et cetera.

A review was performed first of other concepts of instrumentation to try to see if we could identify anything which would be more useful than the core exit thermocouples, the in-core thermocouples, and we could not find any.

DR. JORDAN: By "existing instrumentation," do you mean the in-core thermocouples? Do you also include the saturation meter and the hot leg exit thermocouples?

WITNESS JONES: Yes. The core exit thermocouple -- the in-core thermocouples, I probably used both terms as I go through here. The hot leg RTDs, the t-sat meter, all of those are --

DR. JORDAN: By "existing," you mean those things?

WITNESS JONES: Yes.

DR. JORDAN: All right.

WITNESS JONES: Now, we could not find any instrument that would be worthwhile. But we had basically been directed to look at use of core water level instrumentation. And we examined that issue. We found, first off, that we could not find any instrumentation that was unambiguous or easy to interpret, which was one of the guidelines for selection of an instrument.

CHAIRMAN SMITH: You mean available on the market?

WITNESS JONES: And even some of the concepts that had come up over the last year. We still have not found anything that has been proven to be worthwhile.

And additionally, we did take a look at the existing guidelines to see how possibly could a core water level indication modify those procedures? And we really could not find anything that we would want to change or add as a result of having core water level instrumentation available.

And in fact, there was a lot of concern that putting a core water level instrument in was unsafe. For example, let's take the situation which is a break in the primary system somewhere, say, in the top of the pressurizer, similar to TMI. You get an indication that the pressurizer fills up. Okay.

You then turn around and you look at your core water level indication and remember the core is way down in the system relative to the overall system arrangement. So that level will not even change significantly for quite some time until you essentially drain the entire primary system.

And what happens -- so the operator will see I have my water level going up in the pressurizer, I have a constant water level in the core, gee, maybe I ought to gravel my HPI so I do not put water out on the floor through the pressurizer relief valve because that is not a desirable thing to do.

And if an operator would do that, that would lead you exactly to a TMI-type scenario, because the HPI pumps are not designed to function at -- here is the core starting to uncover now and they will quickly recover the core. That was never their design purpose.

The HPI works by providing an integrated makeup flow to the primary system to help delay the drainage of water from the primary coolant loops, such that the time span that additional water delays the time to the water levels dropping toward the top of the core to a time when the HPI pump is now capable in and of itself to match up with the core boil-off and initiate a refill of the reactor coolant system and core.

And that does not occur for 30 or 40 minutes in many of the small-break transients. In 30 to 40 minuutes of HPI flow, it is something on the order of 15,000 gallons of water. That is a substantial amount of water, and the loss of that water could mean the difference between the core being covered and being adequately cooled to not being adequately cooled.

DR. JORDAN: It seems to me that your definition then of a level meter must be at variance with the staff's, because you are saying the level meter should measure only to the top of the core. And surely, in reading the staff's testimony, I must conclude that their meter goes up into the head, so that they could detect approach to a loss of cooling before it happens.

That was one of the major advantages that they saw. And are you saying that you have not thought about the possibility of a meter that measures the level clear up to the top of the head?

WITNESS JONES: No. I am considering either one within the core region itself or one that spans the entire span of the reactor vessel or even limited from the top of the core to the head of the reactor vessel.

And to point out the problem I have, if you can turn to page 4 of Mr. Phillips' supplementary testimony, at the last -- actually, the whole second paragraph is dealing with the St. Lucie event, and he goes on to finish up with the last sentence, which was: "Additionally, vessel level indication would provide a safe bases for manual shutoff of HPI to avoid overflow to the containment."

Now, in looking at the St. Lucie event -- and I assume in reading this he is talking about the St. Lucie event being in the same paragraph -- you now have a situation where you would have a bubble indication in the primary system within the core, you have a void in the upper head. You have pressurizer level rising, which is what happened at St. Lucie, and this to me indicates using those -- that type of instrumentation that maybe we could shut off the HPI to avoid the overflow of liquid to the containment. And that is exactly the wrong thing to do.

DR. JORDAN: Wouldn't the meters, if there was a bubble up there, wouldn't it show that the level of the water is only up to here and above that it is a bubble, not water?

WITNESS JONES: But, again, taking this in context with the St. Lucie event, which is what he is doing -- that is the way I read that -- he has a bubble in the head. And so what I am interpreting him to be saying is: I have a bubble in the vessel head, and that while I am depressurizing, you know, I am getting level swings in the pressurizer, and that may be the way to prevent me from overfilling the system and putting water on the floor is to throttle back on the HPI, based on the fact that I know I have a bubble in the head, but yet I have the core covered.

You see, my fear is not that you know you have a bubble, my fear is now you know that the core is covered because you measured the head region and there was water there, or you measured the entire span of the reactor vessel and it says there is water 17 feet in the air, which means maybe you have five feet over the top of the core. So you say, "Ah-hah, my core is covered, and so therefore I can shut back on my HPI," when in fact you may lose that accumulated benefit of HPI which is very important to you.

And taking those type of factors into account, I am not even sure the use of vessel meter level is a safe thing to do.

I would prefer the operator strictly rely upon the primary system temperature indications, subcooled or saturated, to determine whether or not to throttle HPI.

CHAIRMAN SMITH: Is that last conclusion a general conclusion, or is it based upon the statement to avoid overflow into the containment is not necessarily a desirable goal?

WITNESS JONES: Well, I agree with the goal to avoid overflowing water into the containment.

CHAIRMAN SMITH: Is that a safety goal or an industrial goal?

WITNESS JONES: I would say it is more of an industrial goal. My own personal feeling, from a safety standpoint, I do not care; they can throw all the water through the pressurizer relief valve they want to keep the core covered, because that is the safe objective, to make sure the core remains covered and does not become uncovered and get damaged.

But obviously, from a commercial standpoint, you do not really want to run your plant to where you start putting water through the valves and putting water all over the floor in the containment. That is not really a desirable thing to do.

CHAIRMAN SMITH: My question was: Was your conclusion that the level meter might be inconsistent with safety? Was it based upon this particular analysis, or was that a general conclusion, counting all of the factors you have taken into account?

WITNESS JONES: I think taking all of the factors into account, the fact that we have indicators should the core become uncovered, which is the exit thermocouples of the core, we have instrumentation that tells us that the plant is in a configuration it should not be in, and that you need to restore inventory -- that is, the saturation meter or just the hot leg RTDs and pressure measurement to tell if the fluid in the RCS is saturated -- and at that point in time, that plant is not in a configuration it is supposed to be in.

Your objective at that point in time is to restore water and to inject all the water you can. You will get on your safety systems and let them inject water and attempt to return the system to a liquid solid condition. If you leave the ECCS on, the emergency core cooling system, you will keep the core covered throughout the event. If you leave them on.

CHAIRMAN SMITH: Before we get too far away from your earlier testimony, I have two clarifying questions. At the very beginning of your most recent testimony, you have indicated that you do not believe that the water level meter would be better than the thermocouple -- in-core thermocouples. And then you expand in-core thermocouples to the other devices mentioned by Dr. Jordan.

Well, that to me is an ambiguous answer. I could take it two ways. One is that we are not talking about alternatives; we are not talking about water level meters instead of thermocouples. Or you could be saying that if it is not better than a thermocouple, then it is superfluous, unnecessary, and it meets that standard that Mr. Keaten is talking about, you don't want it, or whatever other alternative.

But in any event, I believe it was ambiguous. You said that one reason you do not favor water level meters is that they are -- that in-core thermocouples are better. That suggests that we are talking about either-or alternatives. Or it could suggest that since they are not better, they are superfluous and, therefore, undesirable because of the reasons Mr. Keaten mentioned. Or it could be an alternative I have not perceived. And I just wonder if you could clarify.

WITNESS JONES: Well, actually, we believe, first, it would be superfluous. And second -- and this is not in order of priority -- we believe it is superfluous information, and we do believe indeed that the in-core thermocouples are etter.

And jus help illustrate at least the second statement, let's sa at I had a vessel water level instrument that was there and it indicated the core was uncovered. Now, the general types of instruments available to -- available today is simply a DT instrument; that is, it will essentially weigh the column of water in the system. That is about all it can do. And you can infer from the weight essentially where the collapsed or solid water level is.

The problem with that is in the core you do not have a solid water level, you have a two-phased mixture within the core region. So in fact you may have a vessel level indicator even when the transient is progressing normally and safely that will say the core is partially uncovered when, in fact, the two-phased mixture is indeed above the top of the core and the core is being adequately cooled.

That can be distinguished by thermocouple and not the water level instrument. Additionally --

CHAIRMAN SMITH: I was not inquiring about the basis for you believing that thermocouples are superior. I was inquiring about the relevance of that comparison.

MR. BAXTER: Let me see if I understand. Is the question, Mr. Smith: If in-core thermocouples and level meters are equivalent alternatives and yet we say a level meter is unneeded, then also we are thereby indicating that perhaps the in-core thermocouples are unneeded?

CHAIRMAN SMITH: No. I am saying -- who has suggested removing thermocouples? No one. What is the relevance of the fact that thermocouples are better unless they are superfluous and therefore mischievous?

WITNESS KEATEN: Let me give the answer. I think, from all our viewpoints -- and Mr. Jones can talk about it from the B&W standpoint -- I would also like to answer your earlier question, Mr. Smith.

Certainly, in trying to evaluate whether water level might be of use to us, we have not evaluated on the assumption that if you put in a water level meter that you would take out the in-core thermocouples or take out the other instruments. But rather, we have tried to evaluate how would we use it in addition to the instrumentation which is already there.

In my earlier statement having to do with the fact that we believe it is superfluous, I was not relying on the fact that our present procedures do not call for water level. Of course, they would not call for it, because we do not have such a meter. But rather, I was referring not only to the work which B&W has done, but to an evaluation which I personally performed about a year ago, where I took the scenarios of various small-break LOCAs and other events, as developed by Babcock & Wilcox, and sat down myself and tried to say if I had a water level meter in addition to the things I already have, what would I tell the operator to do different or to do in addition to what he is already doing?

And the answer I personally came up with is the same one that B&W came up with. I could not figure out how we would use it.

CHAIRMAN SMITH: The question I am asking now is not to challenge the testimony, but only to clarify it, because I think that one answer could have been ambiguous if the record would be approached at that point. Which now suggests my second question.

Mr. Jones indicated that the concept of water level meters and those on the market are not particularly good or they are not particularly useful. I think it might be helpful if he were to explain what assumptions of the capabilities of water level meters were used in determining how they could have been used in the operating procedures.

BY MR. BAXTER: (Resuming)

Q Is this evaluation that you have been referring to, Mr. Jones, of methods to detect inadequate core cooling beyond those available with existing instrumentation documented in a paper entitled "Evaluation of Instrumentation to Detect Inadequate Core Cooling, Prepared for the 177 Owners Group," dated August 15, 1980, and found in Licensee's Restart Report, Exhibit 1, supplement 1, part 2, the answer to question 95?

A (WITNESS JONES) Yes, it is.

CHAIRMAN SMITH: If those are the assumptions, that is fine.

WITNESS JONES: But even my own look at the instrumentation would be -- let's assume I have an instrument that was perfect, that functioned perfectly, which is the way I looked at it myself, and I could still find no use for it even if it could pick up the two-phase water level, I could find no use for such an instrument that, you know, does anything different than what we have done based on the existing instrumentation.

DR. JORDAN: That raises several questions, and I probably will not be able to remember them long enough to go through them. But one of the things that bothers me is you say you have done in this analysis -- looked at possible meters that would augment the present meters. It seems to me that if you have done this, that this would meet the requirements for the January 1 which Mr. Rubin has -- no, I guess it is not Mr. Rubin. No. Mr. Rubin goes further than that. Mr. Rubin says you do not even have to have the restart requirements, but they say you have not attempted even to meet the January 1 requirements, which calls for an analysis. And yet you say you have done an analysis and it has been submitted. Now, where does the truth lie here?

WITNESS JONES: All --

DR. JORDAN: Should we save this for further --

MR. BAXTER: No.

WITNESS JONES: All I can say to that is that I do not understand the basis for the staff's statement. We have done our analysis and submitted it in August. They have said -- they have apparently come back and said it is either not an adequate analysis or -- what the staff really wants is for us to put water level in. But yet they will not come out and say it. I just do not know any other way to read it.

DR. JORDAN: Okay, that is fine.

CHAIRMAN SMITH: Maybe we will find out today.

(Laughter.)

DR. JORDAN: I guess we possibly will.

WITNESS KEATEN: Excuse me. If I might, I would like to also answer Mr. Smith's question. In the evaluation which I did as far as the characteristics of the water level meter, I considered two cases. I considered a water level meter of the type that Mr. Jones described, that measures the effective collapsed water level, such that if there was a froth, the meter would think it was solid water at a lower level.

And then I alternatively considered one that would actually give the true indications of the top of the two-phase mixture. And as in the case of the B&W analysis, in neither case was I able to identify anything that I could tell the operator to do different based upon that information.

DR. JORDAN: I will come back. I have this analysis in front of me, and I have not had time to read it. I only learned last night that this was one of the documents.

But in reading the short response, I am not sure --

it seems to me all it says is that -- when I read the response, it says what we have is adequate, we do not need to do anything more.

But we will certainly come back to that later, so let's not dwell on it now.

DR. LITTLE: When you say he would do the same thing, does this mean he would do it on the same time basis, too? He could not do anything earlier if he had water level instrumentation than he can do with the present instrumentation?

WITNESS JONES: That is basically correct. And, again, as I said, I have some personal fears that he would use it the wrong way. He could actually use it to do things that would be unsafe.

WITNESS KEATEN: Dr. Little, the problem we run into here is that the operator has to be prepared to meet a very wide spectrum of possible events, ranging from, for example, a very small break to a very large break, depending on which size break actually occurs. And where it occurs in the system, the behavior of either the actual liquid level or the two-phase liquid level varies enormously from one transient to another transient.

And so whether the level is dropping rapidly or slowly is not particularly useful information to the operator, because that may be perfectly normal behavior for the particular event that he has.

It is not a situation where he can look and say if the level is dropping at one foot a minute I am all right, my emergency core cooling systems will take care of it, but if it is dropping two feet a minute I am not all right and I need to do something different. That is not the situation, because the two-feet-a-minute rate of drop may be perfectly normal for a different break, which again the emergency core cooling systems are perfectly capable of handling.

So that is why we ran into the problem of not figuring out how to use the information.

DR. JORDAN: We surely have to get into today the business of the overcooling event. I just do not understand why you say that a level meter would not be useful. If I found that there was an overcooling -- supposing we did have an overcooling event, the level drops, and then it stops dropping. Surely that is valuable information for the operator to know.

WITNESS KEATEN: But, Dr. Jordan, that also occurs for quite a spectrum of small-break LOCAs.

WITNESS JONES: That is correct. And there are -- hold on one second.

(Pause.)

WITNESS JONES: Just to help illustrate Mr. Keaten's point, this is a figure extracted from a previous exhibit, Licensee's Exhibit Number 4. It is not the whole document, but this is just the figure.

(Extract from Licensee's Exhibit Number 4 follows.)

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Docket No. 50-289
(Restart)
Excerpt from Licensee's Exhibit No. 4

Letter dated July 18, 1978 from
J.H. Taylor, Babcock & Wilcox,
to S.A. Varga, NRC, attaching additional
ECCS small break analyses

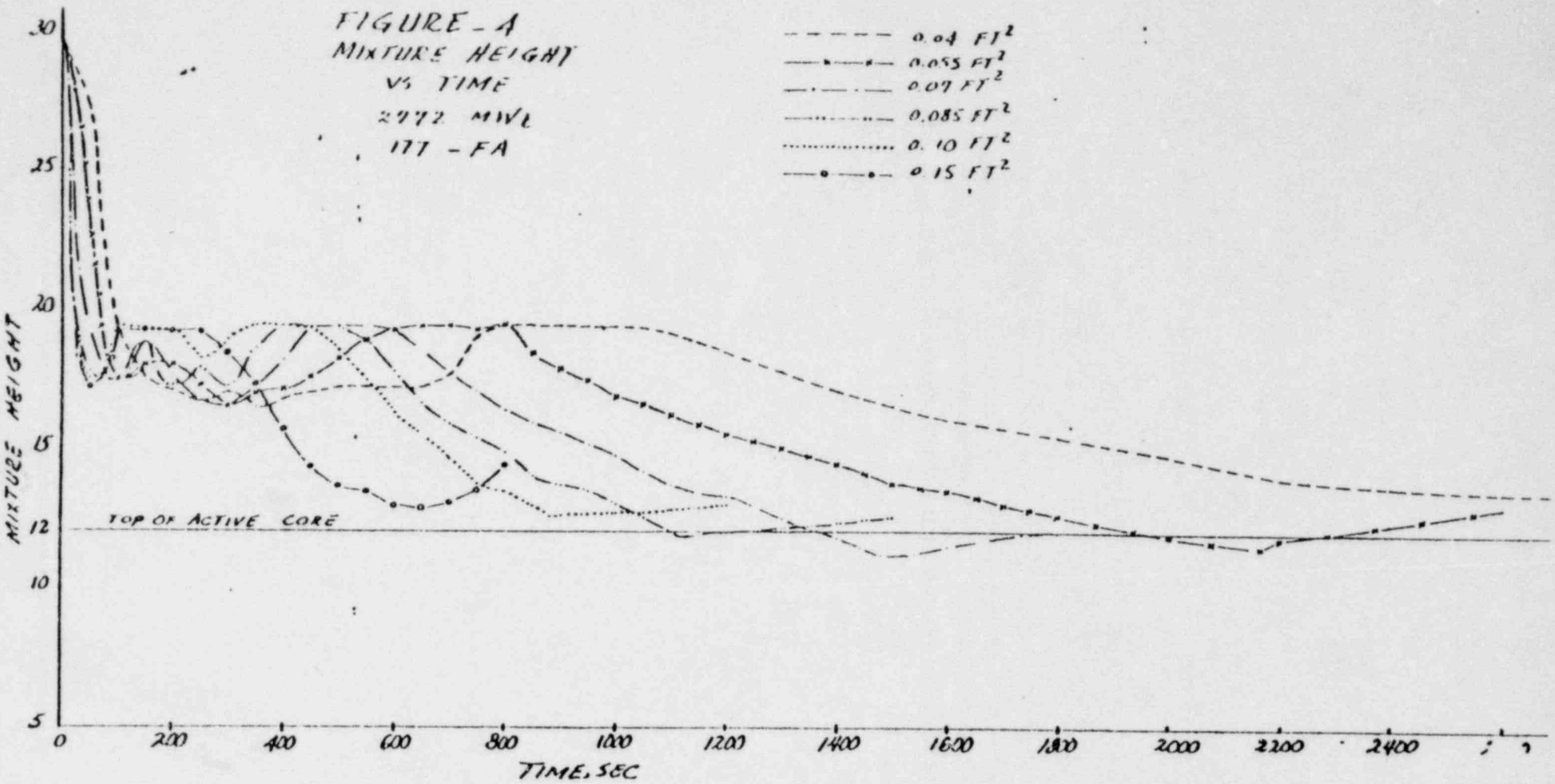
FIGURE - 4
MIXTURE HEIGHT
VS TIME
2772 MWt
177 - FA
0.04 FT²
0.055 FT²
0.07 FT²
0.085 FT²
0.10 FT²
0.15 FT²
TOP OF ACTIVE CORE
MIXTURE HEIGHT
30
25
20
15
12
10
5
0
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
TIME, SEC

WITNESS JONES: And to help show part of the problem you get, this is a spectrum of small-break vessel level response, basically two-phase liquid response for a variety of break sizes.

And as you can see, characteristically, after an initial -- initial drop which is coming down and putting the two-phase mixture level at the internal vent valve, the level will stabilize, basically just stay stabilized at roughly that elevation, and dependent on the size, for quite a prolonged period of time. And if you had a rapid overcooling event which pulled a bubble in the overhead, you would see something very similar to this.

And the other problem is this is just to a .04 square foot break, and if you used a smaller break, then the .04 which is shown here, the actual initial level response drop will be much, much slower, and you would not even see an indication of drop in level for possibly several minutes.

CHAIRMAN SMITH: Before you get too far, I would like to suggest that this be bound into the transcript at the beginning of his explanation, right at the point that he described it as being an excerpt from Licensee's Exhibit 4.

Is that possible, Mr. Reporter?

DR. LITTLE: You are implying water level instrumentation may be more ambiguous than the information you already have?

WITNESS JONES: That is correct, especially in trying to figure out where the transient is going or the initial response of the transient because of a wide variety of conditions you can get for both an overcooling-type accident or event or a small-break LOCA event, depending on its size, location, et cetera. It becomes very difficult to use it as a distinguisher.

DR. JORDAN: Could we start over again? I did not hear the question.

DR. LITTLE: The question was: In the early part of an event, the water level instrumentation may give more ambiguous -- may be more ambiguous than what they would ordinarily have.

DR. JORDAN: Than what you ordinarily have?

DR. LITTLE: With the existing means of --

DR. JORDAN: What?

MR. BAXTER: Existing instrumentation.

CHAIRMAN SMITH: This is the first examination among Board members that I have ever had.

(Laughter.)

CHAIRMAN SMITH: Off the record for a moment.

(Discussion off the record.)

DR. JORDAN: Would you interpret what you think the question is now and then repeat your answer?

WITNESS JONES: The way I understand the question

-- the question, I think, asked: Do you believe basically that water level may in fact be an ambiguous signal early in the transient? And I do agree with that, and especially trying to diagnose between an overcooling and an undercooling event. And, you know, it gets more complicated, depending on the scenario, whether the pumps are running or not, et ceterea. If you are running with the pumps on, you may really have a problem, because the level indicator could say no water initially because of just the pressure drop across the core. So you need to be very careful.

DR. JORDAN: Do steam plants of a site gauge still cold plants?

WITNESS JONES: I do not really know.

DR. JORDAN: Do they have water level indication in a boiler? That is ambiguous information, I guess is what is bothering me.

WITNESS JONES: A boiling water reactor --

DR. JORDAN: Not boiling water reactor. I am talking about a cold plant.

WITNESS JONES: I really do not know about cold plants.

DR. JORDAN: Okay.

CHAIRMAN SMITH: There would not be an equivalent water level -- you have steam generator -- would the --

never mind. Strike that.

DR. JORDAN: There is a boiler. But, okay, go ahead.

BY MR. BAXTER: (Resuming)

Q Let me, if I may, return to the issue raised some time ago by Mr. Smith about what evaluation you have done of additional operator actions or instrumentation, and complete that subject. We were looking at page 4 of Mr. Phillips' testimony. But in supplementation in answer to Licensee's interrogatory number 1, he asserts that: "We specifically have not addressed the desirability of various operator actions to increase safety injection makeup flow when a slow but continuous loss of coolant inventory is indicated."

And Mr. Phillips suggests that one such action may be rapid OTSGD pressurization to 400 p.s.i.g. which would make the accumulator coolant injection available.

Would you assess these criticisms, Mr. Jones?

DR. JORDAN: I can't find it.

MR. BAXTER: It is in the interrogatory responses, interrogatory 1.

DR. JORDAN: Interrogatory 1.

MR. BAXTER: It is in connection -- I linked it to the testimony -- it is connection with the criticism in that interrogatory that we have not submitted analyses or evaluations addressing additional actions.

DR. JORDAN: The last paragraph, which is page 1. Answer to interrogatory 1, is that where we are?

MR. BAXTER: That's correct

DR. JORDAN: All right. Thanks.

WITNESS JONES: Well, I -- I am not so sure that we have said to the staff we have done this little item, looked at this little item, et cetera. But let me just address what is there.

The question is that we have not looked at maybe the use of water level indicator for -- where we know we have or would be picking up indications where we are slowly losing inventory in the system, and then go on and based on that take some other actions in the plant, such as an OTSGD pressurization.

We have, in fact, considered this, not so much from the standpoint of where the level instrument -- but even the initial creation of the inadequate core cooling guidelines, since some of the small breaks utilize the steam generator for heat removal, we recognize that if we drop the steam generator pressure rapidly, we would hope to depressurize the primary system and, hence, increase primary system flow, makeup flow.

But there are problems associated with that, which is one of the reasons why we did not suggest rapid depressurizations within the normal small-break transient.

We do have -- to first describe what we have, we have within the small-break transients, if he knows he has a small break, which he should be able to determine very readily off the existing instrumentation, the operator has been instructed to initiate a 100-degree per hour cooldown of the plant in a saturated -- even though the system may be saturated.

So that in fact we do have actions within the guidelines for these slow, continuous depressurization transients where we are increasing safety injection makeup by depressurizing the plant with the steam generator. That is within the existing small-break guidelines.

Now, the reason we do not like a rapid OTSGD pressurization are a couple. First off, we would like the operator to really evaluate what is happening in the plant. Do you want the operator to rapidly depressurize the steam generator when maybe what he has is a combination tube rupture event with a LOCA? That would not be a good thing to do, because that would increase the leakage flow into the generator and increase off-site dose. We would rather have the operator evaluate secondary-side conditions.

The operator is in the middle of a severe transient, a small-break LOCA. We do not want him to go -- we would like him to get the plant stable, make sure his injection systems are on, et cetera, and then to take control, flow control of the plant to initiate a rapid depressurization in the steam generator, will impose another transient on top of a transient, and we think that might mask other things that are happening to the system. And it is not desirable, therefore, to take a sudden action.

Rather, what we feel is better is for the operator to continue a continous slow cooldown of the plant by depressurization of the secondary side in a controlled fashion. And then, if something happens while in that controlled fashion, he can evaluate it rather than taking a rapid drop and then figuring out now is the primary system doing this because I did it to the plant or is it doing it as a consequence of the original event.

DR. JORDAN: When you say "continuous cooldown by depressurization," you mean the cooling now is taking place by the boiling of the water and the escape of the steam to the break? Is that the cooling you are talking about?

WITNESS JONES: Well, what I am talking about -- when I am talking about the cooldown of the plant, what I am talking about is essentially decreasing its pressure and temperature by bringing down the steam generator pressure, condensing steam within the primary system by that action, and slowly depressurizing the primary system.

The core itself is being cooled by the two-phase mixture and from the continuous HPI flow.

DR. JORDAN: You are talking about using the steam generator then and the condensor for injecting -- rejecting the heat.

WITNESS JONES: Or the atmospheric dump valves, manual control of those sorts of items. But again, we would rather the operator take a cautious look to evaluate all the system conditions and take slow control rather than initiate very rapid transients when they may not be necessary. And they should not be necessary if the systems are functioning properly.

BY MR. BAXTER: (Resuming)

Q Let's return to Mr. Phillips' supplementary testimony at page 3.

DR. JORDAN: Which page?

MR. BAXTER: 3.

CHAIRMAN SMITH: We'll look for a time for a break when it is convenient.

MR. BAXTER: This is fine, Mr. Chairman.

CHAIRMAN SMITH: All right, let's take our midmorning break.

(Brief recess.)

CHAIRMAN SMITH: Did we figure out what to do with that insertion in the transcript? It should be both pages. I think we should amend it. It is an excerpt from Licensee's Exhibit 4.

MR. BAXTER: That is correct. I think Mr. Jones did refer to it as not the complete exhibit but an excerpt.

CHAIRMAN SMITH: Can we physically put the words "excerpt from" and let that be the one that is physically bound in?

MR. BAXTER: Mr. Chairman, I distributed over the break to the Board and parties who are here testimony of Solomon Levy in response to UCS Contention Number 13. That document was served on everyone yesterday by mail, but this is an extra copy.

BY MR. BAXTER:

Q Looking at page 3 of Mr. Phillips' supplementary testimony, approximately the middle of the first full paragraph.

DR. JORDAN: The one that begins "For a small break LOCA"?

MR. BAXTER: Yes.

BY MR. BAXTER: (Resuming)

Q In the middle of that paragraph, "For some small break LOCA conditions, the time interval from the instant of primary system saturation conditions until the occurrence of superheat indication on the core exit thermocouples or hot leg RTDs is in excess of 30 minutes and possibly up to three hours or more."

Mr. Jones, what are the implications of such time intervals?

A (WITNESS JONES:) Well, I do not understand the total implications that Mr. Phillips was trying to draw there. I assume what he is trying to get at is that if you have such a prolonged period of time, the operator could do something to assist the transient or the water level indicator.

Pe[illegible]ly, all it means to me is that you will be then using your high pressure injection water for something on the order -- in excess of 30 minutes and possibly up to three hours after you saturate the system. That is again going back to the primary design function of the HPI. If you reach low pressure or saturation conditions and actuate the HPI, you want to keep that HPI on until such time as you have returned the system to a subcooled state.

DR. JORDAN: But isn't the staff saying that during this period there is a continuous lowering of the water level and it would be useful for the operator to know this before he reaches inadequate core cooling conditions? And are you saying there is just no need for him to know that there is a lowering of the water level, that it does not help him to know it?

WITNESS JONES: Well, in my opinion, yes. And if you go back to that excerpt, the figure looking at the water level, transient as a function of time, you in fact see that although it may be three hours or 30 minutes before you might reach the top of the core, for a substantial period of time -- now, 30-minute-type timeframe. Let me just back up a little.

A 30-minute time frame to expose the upper portions of the core imply that you have a relatively small size LOCA, okay, on the order of .05 square feet or less, looking at this excerpt. And you see the water level indication basically remains in a very tight band for a long period of that time, so you would only even see the indication of water level continuing to drop in the vessel later on in that timeframe, and in fact you do not have this up to three hours indication that he talks about.

You will only see it from a period of time that the level drops from the vent valve location to the top of the core, and that could be much shorter than three hours or 30 minutes, depending on the break size.

Additionally, as you can see from the excerpt, there is a substantial difference in the rate of water level drop between these cases, and as a break size gets smaller and smaller or goes to the larger size, the range is going to get even more spread out to where you will see very slow drops in level, to where you will see rapid drops in water level indication.

I do not know of any way to tell an operator how to interpret what is the proper rate because that rate is dependent on where the break is, how big it is, what the previous power history is on the core, how much HPI may be going into the system. There are so many parameters to correlate that to try to use rate of water level drop as an indicator becomes very difficult to do because, as you see, even from these excerpts, the water level drops for maybe a substantial period of time and then turns around.

I think that would possibly even alarm the operator and make him take other actions. That, indeed, may not be necessary nor desirable. One of the examples of this is in the inadequate core cooling guidelines. One of the actions we take is to restart the reactor coolant pumps. Now we know that use of the reactor coolant pump has a potential safety hazard associated with it, so you do not want the operator to start the reactor coolant pumps when it is not necessary.

But if you are in a condition where you have indeed exposed the core beyond where you expect to have been, at that point in time you really are trying to find other ways to cool the core, and at that point the judgments remain that the risk of a potential ultimate loss of the pump later on in the transient is far outweighted by the benefits associated with turning on the pump and getting a slug of water into the core and helping to rapidly decrease the temperature.

So, although there is time there for the operator if he had such a level instrument, I have no idea how to tell him to use it based on level drop.

BY MR. BAXTER: (Resuming)

Q In the last paragraph on page 3 of Mr. Phillips' supplementary testimony, the first sentence in part, he takes the position that the trend of level indication would provide valuable diagnostic information on the nature of the transient before the level drops into the core.

This position is amplified on in the staff's response to Licensee's interrogatory number 4 and the part of that answer that is on page 4 of that document, where Mr. Phillips says that with respect to diagnostics, the primary coolant inventory versus time data could be compared to event scenarios considered in the safety analyses to infer the type and location of the break or to provide evidence of anomalous behavior which could otherwise be undetected.

I have two questions, then. Why would it be necessary, if it is, to provide positive and direct indication that a LOCA exists, as Mr. Phillips asserts earlier in that answer to interrogatory number 4; and second, is the suggestion of comparing inventory versus time data with event scenarios a practical suggestion for an operator in the control room?

A (WITNESS JONES:) Well, on the first question, which is why is it necessary to provide a direct indication that a LOCA exists, first of all, I do not believe a water level instrument would do it. But I believe the existing instrumentation already provides it to you. Additionally, it really is not necessary, especially in the early portion of the transient, to be able to tell whether it is a LOCA or a non-LOCA event.

If you have a transient which leads to saturation, you want to get the high pressure injection on to resupply makeup fluid or water to the system, to assure that adequate inventory will be maintained so that the core will remain cool. So, in fact you do not need to be able to tell it right away. You don't need to have a meter which flashes up and a flag that says you have a LOCA. You do not need to know that right away.

On saturation you need your HPI on, which will be performed automatically. And secondly, you keep it on until it is there and the operator has the time based on the existing information and has sufficient information to diagnose that he has a LOCA in the plant.

DR. JORDAN: Mr. Jones, whether or not a level meter may be the best way, woulin't it be useful to have some indication of approach to inadequate core cooling, and by approach I mean exactly what the staff has been saying, well long before you reach the place where the core is uncovered. Wouldn't it be helpful to have a device that says we are losing inventory, we should be doing something about it if we can?

WITNESS JONES: The problem with that is, as you can see from that excerpt, you lose inventory for a substantial period of time, depending on the size of the break.

DR. JORDAN: Yes, but wouldn't it be helpful to have this information?

WITNESS JONES: Let me just try to tell you what the ICC guidelines really look like and the transition between the small break LOCA guidelines -- I used an anachronism -- ICC meaning inadequate core cooling, how you make the transition from the small break guidelines to the inadequate core cooling guidelines.

When we get to superheat conditions in the core, they really do not tell the operator to do, at least initially, anything different than what we had already told him to do. It is basically verifying you have your HBI on, verifying that you have cooling to the steam generator, verifying that you are cooling down with the steam generator, actions of that nature, and then to take some actions to prepare him for potential steps later on, such as verify that the core flood tank isolation valves are open, because there are places in the small break procedures -- in the guidelines, excuse me -- which would allow you to close the isolation valve on the core flood tank, at least within the guidelines.

So that is the alert stage for the operator. That is his advanced warning, because we do have safety analysis cases which do uncover the core somewhat.

DR. JORDAN: Which do what?

WITNESS JONES: We do have some of the safety analyses cases which do uncover portions of the upper regions of the core by about a foot. So we wanted him to be ready at that stage, get into the procedure and be ready to take the action, and then to perform the functions if the system continues to degrade from there and to monitor the in-core thermocouples to see whether he is continuing to degrade or whether he has a small degradation and then starts a recovery process.

Then he goes back when he recovers the core. He goes back, transfers back into the normal small break guidelines again, and we feel that is adequate advanced warning to the operator for the potential onset of inadequate core cooling conditions.

DR. JORDAN: Suppose I had a meter that read reactor inventory. Would that be a helpful meter to have?

WITNESS JONES: From an analyst's standpoint, if you had a reactor vessel water meter it would be useful to me to go and later on look at the event and benchmark computer codes, et cetera. From an operator's standpoint, I do not really see that much of a benefit to be gained from it.

And again, I might be -- personally, I just worry a little bit about the operator confusion aspects. But, you know, I cannot really testify exactly as to the type of confusion it might give him. But certainly I feel that it would tend to confuse him.

DR. JORDAN: Mr. Keaten or Mr. Ross?

WITNESS ROSS: I think on the subject of operator confusion you must realize this meter would sit there reading the same at all times during power operation. It would read the same whether at 100 percent power, whether I make a reduction to 50 percent power, whether to 15 percent power. It would read the same if I had a normal reactor trip, whereas the other indicators would move. The operator would see them move. He would grow reliant on the other indicators and know what they mean.

As far as value to the operator or directing his actions on this indicator, I see little value in it.

DR. JORDAN: Okay.

WITNESS KEATEN: Dr. Jordan, let me see if I can say the same thing in slightly different words. The problem that we have found in trying to decide how we would use this kind of level information if it were available, even assuming that there were an ideal level meter or inventory meter and so forth is that we have not been able to figure out how to correlate between what we are seeing as a level and whether or not there is, in fact, an approach to the condition that we are referring to as inadequate core cooling.

The important thing to recognize is that under LOCA conditions, the emergency core cooling systems have been designed to protect the core, and really, the role of the operator is simply to see that those systems are working the way that they are supposed to, and we really do not want him to do something different.

The inadequate core cooling procedure or guidelines and the procedures that we have developed from that are directed at a situation that is basically a nonmechanistic situation in which for some reason although we have had a loss of coolant accident, things are not proceeding the way that the analyses say that they will, and so that we are getting into a new beyond-the-design-basis type of event.

In this situation we are asking the operators to take further and very unusual actions, actions which, as Mr. Jones has pointed out, in fact have some possbility of being detrimental. But we are willing to take them because we are now in a hypothetical situation beyond the design basis.

In looking at how that could be related to reactor vessel inventory or water level, we ran up against the snag that for many of the LOCA events, as the figure shows, in fact the water level drops down to the top of the core, and in that situation we do not want the operator to do anything.

We want him to continue to rely on his emergency core cooling systems to recover that. And it is only if we get some indication that the emergency core cooling systems are not going to be capable of doing their job that we would want the operator to do something different, and that cannot be water level because they can be doing their job just fine and you still see a drop in the water level.

What we have been able to identify, however, is that if we see temperatures on the in-core thermocouples that are sufficiently higher than we expect, that it indicates that the cladding temperature is going above the range predicted by the analysis.

Now we have an indication that things are not going the way the analysis predicted. That is the reason we were able to identify actions based on that and why we have not been able to identify any actions based on water level.

DR. JORDAN: This recommendation, of course, came out of the TMI-2 incident. But I guess most everyone has the feeling that if there had been a meter that said the inventory is disappearing, gentlemen, would those operators then have turned off the ICS system deliberately? I guess -- I am sure your feeling is now with the instrumentation that you do have, that the operator will not take that course. But he did have the in-core instrument. He did have in-core thermocouples, and in spite of what those read, he did turn it off.

So why wouldn't an inventory meter be helpful?

WITNESS KEATEN: Dr. Jordan, let me point out first that we are not asking the operators to look at the in-core thermocouples to decide whether or not to throttle the high pressure injection. We are asking him to look at his primary system temperature and pressure, or alternatively, at the saturation meter. And it is true that he did have primary system temperature and pressure. He did not have a saturation meter such as we now have.

But the other thing he did not have is the very extensive training and very clear procedures that are now provided to him that really explain to him what he needs to do, and this is why you have heard me say on previous occasions that in my viewpoint the real lesson from the accident is not that we need additional hardware, it is to make sure we have the operators use correctly the hardware that is there.

DR. JORDAN: I have heard you say that before, Mr. Keaten. Thank you.

DR. JORDAN: Okay. I believe the second question -- I will try to paraphrase it becaue it has been a while -- is can we use the primary coolant inventory versus time data compared to the safety analysis predictions and therefore be able to pick up an anomalous behavior?

Well, I find that very difficult to believe it could be done because first off, as we have already stated, the water level changes dramatically and differently depending on the size and location of the break. But it is not only that. Even if I could put it in and somehow could correlate it to say system pressure, for example, there are other things to recognize.

First off, it makes assumptions about certain equipment status, so much feedwater flow to the generator, so much HPI flow to the system, the generator pressure control of safety valves instead of what would probably be controlled out, which is the turbine bypass valves. When you put all the various parameters that are in the analysis, core decay heat that we have assumed versus what would actually be there, when you put them all together you have one jumbled mess. And to try to compare what is going on in the plant, certainly I would find it very difficult myself personally to sit there and look at the thing with my experience and try to say yes, it is that break and that is where it is and it is anomalous.

You would have to, at best, have to program it into some computer and then have some parametric variations on all the other parameters that could affect the analysis, and it would be very difficult even to probably draw that correlation and present it to the operator and say you have got something wrong.

He could find out quicker by looking at the HPI flow, comparing it to what he should have and saying, hey, I do not have enough flow. I either better get that flow up or be prepared to take further actions because this has a potential of degrading. That is a better indicator of a lead to inadequate core cooling than looking at the water level drop because it is direct and simple.

BY MR. BAXTER: (Resuming)

Q Would you consider the use of vessel level measurement for the kind of diagnostic exercise Mr. Phillips has suggested to be an unambiguous and easy to interpret indication?

A (WITNESS JONES:) No, sir.

CHAIRMAN SMITH: Could you with your t-sat meter and in-core thermocouples and all of the other data that would be available to you calculate the water level at a given moment in a given event?

WITNESS JONES: It is possible from the in-core thermocouples to predict the water level, the two-phase mixture level in the core. You would have to have some sort of a calculational routine to do it because you would have to account for the power shape and the core decay heat that was on in the plant at that time in order to make that calculation. But such a calculation could be performed.

That's it.

BY MR. BAXTER: (Resuming)

Q What use might the operator make of that calculation if he could get it?

A (WITNESS JONES:) Off-hand I know of no real use to him because he would have to wait for first the calculation to come up if he was going to use it to take an action. The in-core thermocouples are responding at the time and it is a faster thing, it is a faster indication availabale to the operator than going through the calculational routine, although that may take a small period of time.

But you know, since you are backing it out of the in-core, I do not see how it provides you anything really additionally useful, more useful than what you already have from the in-cores.

Q Looking at the same sentence in the last paragraph on page 3 of Mr. Phillips' supplementary testimony and also the interrogatory 4, he asserts that water level indication would also provide indication when safety injection makeup is adequate to prevent core uncovery and would provide positive information when primary system inventory recovery commenced.

Mr. Jones, what would the operator do with this information?

A (WITNESS JONES:) Well first off, I do not see any way, based on the level information as it changes during the transient, the operator would be able to tell whether he had adequate safety injection flow because even the analyses which have been performed, assuming the flow is there, show the water level dropping with time. So it is difficult to assess that.

Certainly a level instrument could tell him when primary inventory recovery is commencing, possibly; but again, there is nothing that he would do with that because his objective is still to restore the plant, ultimately, to a liquid, full primary system loop condition. That is his ultimate objective, to return the system to a subcooled state.

Personally, all I see that positive information about the recovery would tell, would do would be give me maybe a slightly warm feeling in my stomach, is about it. I cannot see an operator doing anything with it because the rate of recovery, just like the uncovery of the core, is dependent on break size and location. The rate of recovery has the same parameters affecting it.

Q Mr. Jones, I take it from your earlier testimony you have examined the NRC staff's report on the St. Lucy 1 natural circulation cooldown on June 11, 1980, which is cited by Mr. Phillips on page 4 of his supplementary testimony. Is that correct?

A (WITNESS JONES:) That is correct.

Q Both to help explain your earlier testimony and to answer my next question, I think it would be appropriate to very briefly summarize what that event was, if you would.

A (WITNESS JONES:) Well, basically what happened at St. Lucy was they had a loss of component cooling water to the reactor coolant pumps, which therefore they lost cooling to the pump seals, and to prevent damage to the pump seals, they tripped the reactor coolant pumps and then initiated a plant cooldown on natural circulation.

At some time during this cooldown process, they apparently pulled a void within the primary system.

Specifically, it is believe to have been located in the upper head of the vessel. That was indicated to the operators because the level swing occurring in the pressurizer was rather large and could not be explained by the fact that they were just changing the injection location from the cold legs of the system to the pressurizer as a repressurizer spray.

The significance of the St. Lucy event was basically the fact that a bubble was drawn in the head.

Q Does St. Lucy 1 have a Babcock and Wilcox nuclear steam supply system?

A (WITNESS JONES:) No. The St. Lucy plant is a Combustion Engineering plant.

Q Would you assess, then, the contention by Mr. Phillips on page 4 of his supplementary testimony that this event at St. Lucy 1 somehow illustrates a need for water level indication?

A (WITNESS JONES:) Well, I do not see the event really showing the need for water level indication. In fact, I think the event just tends to point out one of the problems with using water level as a distinguisher between overcooling and a small break LOCA, as I have discussed just previously.

What the staff has stated in the report and is stated in Mr. Phillips' testimony is that the operators apparently did not initially recognize the fact that they had drawn a bubble in the head of the reactor vessel, and they have somehow concluded that possibly unsafe operator actions could have been taken and that if they had vessel level instrumentation they would have known about the bubble in the head.

I have had trouble trying to figure out what unsafe operator actions could have been taken because there are no details in the report, but additionally the operators were able to figure out they apparently had a bubble in the vessel head because they controlled the plant, and certainly it was easily recognizable by the level swings that were occurring in the pressurizer, and it was a direct indication, in fact, that they had a void in the primary system.

Now, as far as possible unsafe operator actions, the only action that I can conclude that might have been taken, reasonably have been taken is that they continued to cool down the vessel, cool down the plant, which allowed the bubble in the vessel head to expand and some of the steam to be drawn into the reactor coolant system loops.

That would not necessarily cause a safety problem in the sense that you would now have hot steam coming in contact with cold water and would condense, number one, and additionally it would be easily recognizable on the temperature and pressure indications for the plant because if that happens, you would have had a rapid pressure decrease as a result of the condensing of the steam, plus you would have seen the temperature in those legs increase as a result of the mixing of the hot water with the cold water; and they ultimately would have lost their saturation margin long before they would have -- I cannot say long before, but certainly before they would have had a loss of natural circulation in the loops because the fluid would have gotten hotter in the hot leg, would have been indicated, lost their 50 degree subcooled margin, would have had to restore HPI, which would have been what they would have had to do to maintain 50 degrees subcooled, was to repressurize the plant.

Since St. Lucy, as Mr. Ross has already stated, the St. Lucy event has been fed back into the operator training to assure they understand the potential for void formation during one of these events.

DR. JORDAN: Do they now recognize a St. Lucy event by virtue of the fact of level oscillations? Is that the tip-off now?

WITNESS JONES: That would be a primary way of recognizing it, yes.

BY MR. BAXTER: (Resuming)

Q Mr. Jones, the staff report concludes with a list of recommendations. Did the author of that report include any recommendation that the vessel level indication should be installed or imply anywhere in the report that it would have been helpful?

A (WITNESS JONES:) No, it did not. Basically the recommendations for the plant were along the lines of procedural aspects and a few of the equipment problems noticed during the St. Lucy event. There was no recommendation to put in the water level indicator.

Q Beginning at the bottom of page 4 and moving on to the top of page 5 of Mr. Phillips' supplementary testimony, he asserts that "vessel level information is important and possibly essential to proper emergency procedures relating to use of the reactor vessel head vent required by the TMI Action Plan."

Is B&W developing operator guidelines on the use of these head vents in operating B&W plants, and will these guidelines recommend any correlation or use of water level in the reactor vessel?

A (WITNESS JONES:) We have developed or at least have partially developed the set of guidelines for use of the vents, and at this time we have not determined that there is any need to have water level indication to assure proper utilization of the head vents.

Q Both the original testimony filed by Mr. Phillips and his supplementary testimony cite and rely in part as the basis for the staff's position a letter of September 24, 1980 from Mr. Eisenhut of the staff to Mr. Arnold of Metropolitan Edison Company, with the attached staff evaluation of the Babcock and Wilcox position regarding additional instrumentation for detection of inadequate core cooling for B&W reactors.

If there is anyone who needs a copy, I have extras.

DR. JORDAN: I have it but I have to find it.

(Pause.)

DR. JORDAN: If we could borrow a copy or two, we have copies but --

CHAIRMAN SMITH: The more copies we have, the greater our chances are of finding any given one.

(Laughter.)

BY MR. BAXTER: (Resuming)

Q On page 3 of the staff evaluation attached to the letter I cited, near the bottom of item one, the staff states that its review of available LOFT and semiscale data during large and small break loss of coolant tests and comparison to calculated values of measurable parameters such as differential pressure lead the staff to believe that correlation of measurable parameters to the advent of inadequate core cooling is feasible.

Can you endorse this belief on the part of the staff, Mr. Jones, on the basis they cite and why?

A (WITNESS JONES:) Well, what they are basically saying is, or the way I interpret it is they are trying to state that because we have been able to predict these experiments, you should be able to predict what will happen in the plant during one of these accidents.

There are numerous problems with taking this to a plant. First off, these are controlled experiments, and while I am not going to disagree with the statement that they have been able to correlate in reasonable agreement the calculated parameters, pre-test or even on a post-test evaluation to the measured parameters, it is difficult to see how you could use that in a plant because it is not a test.

In performing these evaluations it is important to note that they have been able to -- first, it is a well -- they try to control the test reasonably well, and they have had some experience in calculating these experiments so they have been able to so-call tune the model to make sure the parameters in general match up, so when they change something, they should be able to pick up that effect within their calculations and it should match reasonable well with the experiment.

The calculations which are done for a nuclear plant have got to cover a wide range of conditions. These are nice specific tests with specific input assumptions on the system performance. For a nuclear plant you would, again, as I tried to point out before -- you have such a large degree of parameters to try to mark up that it would be impossible, in my opinion, to really correlate it.

Just to point out some of the potential fallacies or some of the effects of system interactions on predictions to calculated values, I would like to note the recent L36 test that was done. L36 was a small break loss of coolant accident where they kept the reactor coolant pumps running. I think it was E, G & G in Idaho performed a pretest prediction of this experiment with some assumed system performance based on how the secondary system had performed in previous experiments, and what they calculated was following a tripping of the pumps there would be sufficient inventory in the system that the core would not uncover.

When the experiment was run, one of the valves, or at least the indications to date that I have seen say that one of the valves which has historically had a problem of not closing tightly closed tightly in this experiment and the system depressurized slower, and when they tripped the pumps, instead of having enough water to keep the core covered, the core was way uncovered. It uncovered to a substantial degree, and that is a systematic feedback that can occur at a plant -- a valve sticks or it does not stick or whatever -- that you have to account for.

That is why the safety analyses that are done rely on certain values of safety valves and they do not take credit for some of the equipment like the turbine bypass valve. They do not take credit for the atmospheric dump valves. They take credit for the safety valves on, say, the steam generator, when in fact if you have an accident, it is quite proabable that you would have either the atmospheric dump valve or the turbine bypass valves available to you, and that already knocks your comparison right off. So you cannot really correlate things directly.

DR. JORDAN: Mr. Jones, your statement about the difficulty of applying the LOFT experiments and the codes and so on that are developed there to a nuclear plant sound exactly to me like the critics during the rulemaking hearings on emergency core cooling, which claimed exactly that: that it was impossible on these small-scale experiments and these codes to predict what would happen in a big plant.

WITNESS JONES: No.

DR. JORDAN: Are you agreeing?

WITNESS JONES: I am not saying that. What I am saying is that what they have done is analyze the specific tests. Okay, they are running the test and they analyze the test, either in a pre- or post-test mode. They are also doing best estimate calculations versus what I will call the licensing or evaluation model-type calculation designed to overpredict the normal consequences of the event.

DR. JORDAN: That is true.

WITNESS JONES: They attempt to control the experiment to verify that the codes indeed have proper formulations of the equations so that it indeed can predict system response and system behavior with the right set of input conditions for the plant.

All I am pointing out with the LOFT test is that here is a case where they made an assumption on a system performance based on previous history. They were trying to do a best estimate. They expect the valve will perform the same way it always has, and suddenly the valve performs actually the way it was designed to function, which was not to leak, and they get two different answers.

So if you apply this to a nuclear plant, what you get is should the valve perform differently than what is assumed or the other equipment functions which we are not allowed to take credit for, the comparison is no longer any good from the standpoint of an actual prediction of how the level changes with time.

The calculations are conservative from the standpoint that they do not take credit for the equipment which would help, such as a bypass valve or the atmospheric dumps which keep the steam generator pressure down. Rather we relied only on the safety system itself, which is the steam safety valve, and we further assumed that they stayed and functioned exactly right from the standpoint that they did not stick open.

If they stuck open, the consequence of the accident would be less severe than what we analyzed. So we would mock up the systems in their design configuration, take some assumed failures, not take credit for certain systems that might be available to help you, and we make assumptions on the core decay heat and various other things that all together make the safety analyses really not very good for telling you exactly where that plant is going to go.

But given all the inputs, I think we could do a reasonable job of predicting where the level would go if we knew all the inputs beforehand.

BY MR. BAXTER: (Resuming)

Q Turning to page 4 of this evaluation, item 4 states "The staff finds the position that additional instrumentation is not needed because necessary operator actions will be taken based on existing indicators to be unacceptable. If all actions available to the operator had been taken and the system is continuing to lose coolant due to equipment malfunction or some unknown system condition, the operator should be clearly informed of the situation.

"It is probable that additional actions such as detection and correction of the unknown malfunction or initiation of system depressurization to utilize low pressure coolant injection sources could be taken by the operator if circumstances warranted such action."

Mr. Jones, would you assess for us how vessel level indication might aid the operator and be used to detect and therefore correct an unknown malfunction?

A (WITNESS JONES:) Well, I don't really see a way that the level indicator could be used to correct an unknown malfunction or even detect it because, as I said, you have such a wide spectrum of water level decreases with time that you cannot really determine from rate of change or anything like that what is happening, because if I took the smallest break, say, that I have analyzed and take away the water level or take away the high pressure injection flow, for example, its range of change in level would look just like a smaller break -- a slightly larger break, and I have no real way to distinguish it.

The best way to address this type of an opinion would be to say if I have inadequate HPI flow because I am not getting the flow on the meter that I should be getting, then either fix it or, if you know you cannot fix it, then depressurize the plant rapidly. But you do it based on the indications that I am not getting the flow I need rather than the level drop.

That is a much better way to do it from a systems standpoint of detecting malfunctions and what to do if you have multiple system failures. That is a better approach than looking at a water level approach and, in fact, in my opinion is more direct than trying to infer it from water level.

Q Continuing with that paragraph, the staff states that "even if operator actions are not keyed directly to level indication, the information derived from such an indicator would be valuable in assisting the operator and supporting emergency operations staff to assess the situation and to prepare for those actions required upon indication of the existence of inadequate core cooling."

Mr. Keaten, I have two questions for you on the basis of that statement. One, what assistance might such information provide supporting emergency operations staff; and two, are you in favor or do you agree with the idea that such instrumentation should be installed even if operator actions are not keyed directly to it?

A (WITNESS KEATEN) In the process of trying to evaluate whether or how such level instrumentation might be used if it were available, as has been stated here earlier this morning, we have not been able to figure out any method of using it. We have a situation that if we get a small break LOCA, we expect to see the level to drop.

And depending on whether it is a big or small break LOCA or a smaller small break LOCA, it will drop faster or slower. Many of these LOCAs we expect to see the level drop down to about the top of the core. So if we had a level indicator and we saw the level dropping and saw it getting close to the top of the core, that still would not tell us if there was anything unusual. It could be a perfectly normal loss of coolant accident that the emergency core cooling system was going to accommodate, so we would not know what different to do than we are already doing.

It is not the situation were that kind of a level drop can be identified as being abnormal or can be identified as being a forerunner of the approach to inadequate core cooling. It may be a perfectly normal event, and therefore, you know, let's say we have a level indicator but it was in the technical support center instead of in the control room so the technical staff was evaluating it. I don't know what they would do with the information.

Your second question is --

Q Before you move to that one, Mr. Keaten, I am not sure whether the staff when they say emergency operations staff are implying emergency planning personnel, but if that were the implication, would the information be of use to them?

A (WITNESS KEATEN) It is the same thing. If you are talking about the initiation of the emergency plan, the notification of the state and counties and so forth, TMI has developed a set of action levels that are used to identify what classification of an event is in progress and to use this to start the notification process. On a loss of coolant accident as defined by either a high building pressure and initiation of emergency core cooling or by loss of saturation margin is an action level to trigger a site emergency.

If we had level indication, I do not know what else we would do in the way of notification. Again, if we have a loss of coolant accident, we expect to see the level drop, and it drops further or not as far and faster or slower depending on what kind of a loss of coolant accident is concerned. To my mind in terms of the relevance to the emergency plan, once we knew that we had a loss of coolant accident, I do not know what would be done additionally or differently with the level instrumentation.

The next step beyond a site emergency and the final severe level of declaration is a general emergency, and for the most part, the declaration of general emergency tends to be tied to actual radiation levels, releases of radiation, and I do not see any way it would make technical sense to me that one could use the level instrumentation if it were available to decide whether or not you should go from a site emergency to a general emergency.

As far as the second question you asked, do we favor installing it even if it cannot be used, as I explained earlier, we would like to restrict the information available to the operators to that information which they need and will take actions from, so that we increase our assurance that they are looking at the right thing. And if we overburden them with information, all we do is make their job harder.

So, if we cannot identify how they are using this instrumentation, then I for one am opposed to the idea that it is something that should be presented to the operator.

(Board conferring)

BY MR. BAXTER: (Resuming)

Q To sort of sum up, I would like each member of the panel to answer the question whether in your view reactor vessel water level indication would provide an unambiguous, easy to interpret indication of inadequate core cooling. You may or may not choose to expand or repeat testimony you have already given.

A (WITNESS JONES:) No, I do not believe it is; but there is one other reason that had not been brought out previously I would just like to quickly touch upon. A given water level in the core, say ten foot water level in the core, even if you could measure it exactly, still does not tell you whether or not that core is being adequately cooled or not because if the power shape is such that the power is peaked to the bottom of the core, then you have very low power in the upper portions of the fuel rod.

You do not have that heat to remove, and in fact the steam cooling will keep that rod very cool and thus the superheat temperatures being reached, say, on the in-core thermocouples will reflect that fact. If, however, you are at ten feet and you have a power shape skewed toward the top of the core, then you could indeed -- I am not sure ten feet would do it, but you would get very elevated temperatures if you had a large peaking factor in the very top of the core because you have a lot of energy removed and the in-core thermocouples would pick up higher degrees of superheat coming out of the top of the core for a given core water level.

So, in fact, the water level does not even tell you whether or not it is being adequately cooled without knowing more facts about what is happening in that core, and that you have to interpret even if you are down into the core at that period of time. So I just totally disagree with the fact that water level is an unambiguous or easy-to-interpret indication of inadequate core cooling.

A (WITNESS KEATEN) Well, as I have said earlier, I agree with Mr. Jones. I think the way I look at it is that the configuration that we want the reactor coolant system to be in is the configuration that it is designed for, and since this is a pressurized water reactor, it is designed to run water solid. In the instrumentation that is used to measure such saturation margin, we have an indication, a very clear indication of whether the system is subcooled or whether it is, in fact, in saturated conditions.

If it is at saturated conditions, the real goal and almost the only goal of the operating staff is to resort to subcooled conditions, and the way that is done is to add water. And the amount of water that is added is determined by the design of the emergency core cooling system, and it is designed to add adequate water for the total spectrum of breaks that might occur, as well as for other types of inventory loss accidents such as an overcooling accident.

So what the operators need to do in those cases is make sure his emergency core cooling systems are functioning as they were designed to, or if, for example, one pump does not start even though that is within the design basis, nevertheless he should try to start the second pump in order to try to add more water.

His indication that he has successfully completed that comes not from a water level indication because, as Mr. Jones pointed out earlier, that might be in fact misleading. It comes from the fact that he has reestablished a subcooling so that he has returned the reactor coolant system to the configuration appropriate for a pressurized water reactor.

So given that as the design basis and the operating basis of the plant, and given that we have not been able to identify anything that he would do differently or in addition to the things that he is already instructed to do based upon the presently installed instrumentation, I see no incentive for adding water level instrumentation, and I see some disadvantages if that were to be done.

So I personally would recommend against it.

WITNESS ROSS: I guess I definitely see a disadvantage to burden the operator with an instrument that he does not use in his normmal duties every day. He doesn't respond or indicate nor does he take any actions based on it, I might add, that are different than is already planned.

CHAIRMAN SMITH: No one has discussed whether the physical presence of such a meter inside the reactor would have an undesirable effect.

DR. JORDAN: He means a sensor.

CHAIRMAN SMITH: A sensor, yes.

WITNESS KEATEN: That's right, we have not, Mr. Smith. One reason is that there are a wide variety of different designs of instrumentation that have been and are being considered for use in measuring water level or something equivalent to water level, and they vary radically in terms of whether they have anything inside the vessel or not, and if they do have something, what it is. So it really is impossible to address on a generic basis.

Some of the designs, at least in principle, are thought might be able to work without anything inside the vessel. There are others that rely on sensors. So that is the type of consideration that has to be done on a case-by-case basis depending on what particular design is being considered.

DR. JORDAN: Could you tell us what is involved in -- could you tell us what INPO and NSAC are doing about this? Are they making any analyses, doing any experiments, are they addressing the problem at all that you know of?

WITNESS KEATEN: I cannot address INPO and NSAC, but I can tell you what EPRI is doing. In general EPRI has been following work done by various organizations around the country, and in addition they have done some limited work on their own directed toward a couple of concepts.

To the extent that we are familiar with what is done -- and we have made some attempt, at least, to stay aware of what is going on around the country, realizing that this is a topic that a lot of people have a great deal of interest in -- we have not seen any convergence of opinion toward what is a good or acceptable device for measuring water level.

Now, one of the NRC staff witnesses at the recent ACRS subcommittee meeting on TMI-1 restart indicated that he believed that the staff was -- and I do not remember his exact words -- but along the lines that the staff had encouraging feelings with regard to two approaches, one which measured the water level by the differential pressure technique, and the other which used the heated thermocouple technique.

I believe it is a true statement that both of those are being evaluated as part of the LOFT program. On the other hand, we have heard different evaluations of those and in some cases conflicting evaluations from other people. So, while there is work going on, it is at least my personal opinion that at the moment there is not a method of measuring water level that has uniform and universal acceptability.

A part of the problem is a definition of the conditions under which it is desirable to measure water level. For example, if a determination were made that water level measurements would be used only under circumstances where the reactor coolant pumps were turned off, then a measurement that used differential pressure would be much more meaningful than would be an attempt to use that same method when the reactor coolant pumps were running.

So, to sum up at least to the extent of my knowledge, Dr. Jordan, I would have to characterize all of these efforts as still being in the development stage.

DR. JORDAN: Supposing the NRC requirements were pointed not at a measurement of water level but a measurement of inventory level. What would be your answer then? Would it be the same or would you say you just do not see any way of doing it?

WITNESS KEATEN: I think my answer would be similar. For example, if it was desired to measure the inventory inside the reactor coolant vessel -- the reactor vessel -- and if the only time that I wanted to do this was when the pumps were turned off, which is the normal LOCA scenario, at least to date it is, then it is conceivable to me that a technique that used a differential pressure measurement from the bottom to the top or some portion thereof of the reactor vessel, properly temperature compensated, might be used as an indication of the inventory in the vessel.

DR. JORDAN: Isn't that what is used in the pressurizer?

WITNESS KEATEN: Yes, sir, that is what is used in the pressurizer; that is correct.

WITNESS JONES: But there is a big difference between the pressurizer and the core during one of these transients. The pressurizer is going to tend to be, if you put the break in a nonpressurizer location -- that is, put the break in the pipes -- the pressurizer water level is just going to be a measure basically of where the water is and the actual level in it, while in the core region you have all the frothing --

DR. JORDAN: What?

WITNESS JONES: All the frothing mixture, swelling.

DR. JORDAN: If you have a break.

WITNESS JONES: If you go -- yes, if you have a break you will create steam, trap bubbles within the water, and the only thing the DP cell will be able to pick up for you is how much water is within this region, but it will tell you nothing about where the actual level is. That would be very dangerous to do anything with because, again, you may have more than enough mixture to cover the core, and te solid water level is down below the top of the core, and that is not an unusual condition to be in during our analyses.

DR. JORDAN: I guess I have some feeling that it would be useful for the operator to be able to know instantly whether he is dealing with a loss of coolant, say a small break loss of coolant, or an overcooling event.

Now, you have told me that if there is a loss of level in the pressurizer, than one of the things he does is to immediately look at the secondary system.

I guess what I need to know is what happens in the case of an overcooling event such as a steam line break, a small or large, which produces a loss of level in the pressurizer. What does the operator do about this? How does he know it? How does he respond?

WITNESS JONES: Let me try to just generally characterize all overcooling events rather than hit a specific steam line break or feedwater overfill because the basic dynamics of the transient is essentially the same. There are details which are significantly different, but the basic system dynamics are the same.

DR. JORDAN: All right.

WITNESS JONES: Both events -- any overcooling event is a result of taking more power out of the primary system than the core is producing basically. It is a mismatch between what should be taken out of the secondary side under normal circumstances and what you have here is more coming out. What that does is it cools down the primary system temperatures below what you would normally expect the primary system temperatures to be at following a reactor trip.

That means you have additional shrinkage or compressing of the water because it is at a lower temperature, and in order to replace the shrinkage occurring in the primary system, the pressurizer level falls. And because the pressurizer level falls, the system pressure falls. And for most overcooling events of any significant magnitude, and certainly steam line breaks, you will get a high pressure injection signal or an emergency core cooling system signal to actuate.

DR. JORDAN: One quick one. Would the operator have turned off the coolant pumps?

WITNESS JONES: At that point in time he would then go and turn off the cooling pumps.

DR. JORDAN: All right.

WITNESS JONES: Under the present instructions.

Now, depending on the severity of the overcooling transient, it is possible that the pressurizer may drain and become completely empty of water and indeed start passing steam into the primary system where it would be condensed, at least for some period of time, within the primary system water.

If it is a very severe overcooling transient, such as both steam lines breaking, for example, he might even produce a void within the primary system, within the U bends of the candy cane of the hot leg, and also possibly in the reactor vessel upper head, and the system presure for this transient would be extremely low. It would drop to around 1000 psi or so. It could even go lower, depending on the case that you are analyzing.

Now, the basic operator response during this transient initially is to do nothing except verify that the reactor has tripped -- trip the reactor coolant pumps, that is an operator action, assure the HPI comes on when actuated, and then to go and verify that he has secondary side cooling and to evaluate for an overcooling transient.

Now, those three steps or four steps are the same steps the operator takes initially for a LOCA. There is no difference. Verify the reactor has tripped on the appropriate signal, verify the ECCS works, trip the reactor coolant pumps and assure secondary side cooling. So all four steps are exactly the same.

Now, when he evaluates the secondary side parameters -- is the level to high or the pressure too low, indicative of an overcooling event depending on the nature of the event -- the operator than has to terminate it. If it is too much feedwater flow he throttles back a valve or trips a pump. If it is a steamline break, he assures that the steam line break detection system isolates that steam line, isolates the broken generator and controls properly the unbroken generator.

If it is a LOCA, he would look at those parameters and assure they are being maintained at a proper value. So just doing that confirmation step, he will automatically pick up if it is an overcooling event.

Now, the subsequent actions of the operator are to basically leave the HPI or the ECCS on until the system returns to a subcooled state. At least that is the significant one. There are for the LOCA procedure, of course, many other steps the operator can take, such as cool down the steam generator, et cetera.

But the most important thing is that he leaves the HPI on, and that is the same whether it is an overcooling or a LOCA situation until he has returned the system to a subcooled state, has the pressurizer level rising, and at that point in time he can throttle back on the HPI while continually assuring he maintains adequate subcooling in the system.

And then there are again some other actions which then go forward from that, depending on what the nature of the event was. So all of this period of time, as far as the operation of the adding inventory to the system, they are the same for all intents and purposes.

DR. JORDAN: Is there a difference in the case of an overcooling event that the operator wants to try to make sure that he does not somehow inject a slug of cold water into the pressure vessel, thereby exceeding the pressure temperature limits on the vessel? Is there something that he should avoid doing in the case of an overcooling event that is different from the case of a LOCA and therefore additional information is needed in order to avoid slugging the reactor vessel itself with a slug of cold water?

WITNESS JONES: Off-hand, the only difference that I can come up with is that if he has an overcooling event and he is returning the plant or allowing the plant to heat back up following the original transient response for an overcooling event, the operator is directed to open up his letdown line and to control the steam -- if it is a steam line break, the good steam generator, the broken one, or control both steam generators to try to prevent the water level rise in the pressurizer from filling it up during the subsequent heatup process because he might have added a lot of water from the high pressure injection system which then just had to be removed from the system at this point in time because now you are swelling again in the system and you could potentially fill up the pressurizer.

So he is allowed to take those other functions. But other than that, I do not know of any. As far as a cold slug of fluid, when you turn on the HPI you are putting cold water in the primary system. You are thermally shocking the HPI nozzle. But that is not a real concern. That is part of the design.

If it is a severe overcooling event, there will be nothing you can do about it, possibly, by violating the temperature and pressure limits of the vessel. You want to try to get a handle on it before it happens. But if it is a steam line break, you will violate the pressure, temperature --

DR. JORDAN: There is nothing himself that the operator can do.

WITNESS JONES: Not off-hand.

CHAIRMAN SMITH: Okay. This would be a good time to break for lunch. Let's take --

MR. CUTCHIN: I wanted to ask, Mr. Chairman, if we are approaching the end of Mr. Baxter's examination of these witnesses. I would like to get a feel because there has been a lot of testimony this morning and I believe we are within striking distance of wrapping this issue up by the end of the day.

I would like to have a little extra time at lunch to go over with my people some of the things that have gone on here this morning.

CHAIRMAN SMITH: Mr. Baxter.

MR. BAXTER: I only have three or four more questions. I was going to move to a different subject, though, so this would be a convenient breaking point, but I can do it now if you would like.

MR. CUTCHIN: If you wanted to break now and we could prepare on everything that has gone on, it probably would not create a problem with the additional three or four questions.

CHAIRMAN SMITH: How much time?

MR. CUTCHIN: Were you normally planning to come back at 1:00? I would suggest maybe 1:30.

CHAIRMAN SMITH: All right, sure.

(Whereupon, at 12:07 p.m., the hearing was recessed, to reconvene at 1:30 p.m. the same day.)

AFTERNOON SESSION

(1:32 p.m.)

Whereupon,

ROBERT W. KEATEN

MICHAEL J. ROSS

ROBERT C. JONES

recalled as a witnesses by counsel for the Licensee, Metropolitan Edison Company, having previously been duly sworn by the Chairman, were examined and testified further as follows:

MR. BAXTER: Mr. Chairman, I have determined that I have no additional rebuttal questions, so the witnesses are available for cross examination.

BOARD EXAMINATION

BY DR. JORDAN:

Q I will probably have questions after the staff has cross examined. But there was one thing I was curious about. On September 26 last year the Commission came out with a proposed rule on interim requirements related to hydrogen control and certain degraded-core considerations.

Among the features of the proposed rule, the requirements of the proposed rule, is the following: By January 1, 1982, each boiling and pressurized light water nuclear power reactor -- oh, yes, if you would like to read that, it is on almost the end of the proposed rule, which is page 33, according to my document, but yours will be paginated differently.

Now, do you find at the end, do you find Chilk's signature at the end of the proposed rule? All right, up from that, the paragraph above that is G.

MR. BAXTER: It is F, isn't it, Dr. Jordan?

DR. JORDAN: I must be talking about -- there are two proposed rules on degraded core cooling, and I suspect we are talking about different ones.

CHAIRMAN SMITH: I think you probably came up with the early one, which was not even a proposed rule but it was -- well --

WITNESS KEATEN: This was taken from the Federal Register on October 2. It's item G, labeled "Training to Mitigate Degraded Core Accidents."

CHAIRMAN SMITH: No, that is the early --

DR. JORDAN: The last paragraph in mine is "Training to Mitigate Degraded Core Accidents." That is right. But just above that I have item 4 and item 3. Item 3 I was starting to read from. Do you see that?

WITNESS KEATEN: Yes.

BY DR. JORDAN:

Q It says -- and I will repeat: "By January 1, 1981, each boiling and pressurized light water nuclear power reactor shall be provided with instrumentation such that a reactor vessel water level indicator which supplies to the control room a recorded unambiguous direct indication of inadequate core cooling. The indication must cover the complete range from normal operation to complete core uncovery and give advanced warning of th roach of inadequate core cooling."

Now, my first question, this proposed rule was put out for comment and comments were due in before November 30 -- November 3, 1980. Did B&W or Met Ed comment on this? And if so, what were their comments?

A (WITNESS JONES) I do not know from B&W's standpoint.

Q I see. All right. Okay. I note that you would then differ with the requirement to have a reactor vessel level indicator.

Do you also disagree with the second sentence, which states: "The indication must cover the complete range from normal operation to complete core uncovery and give advanced warning of the approach to inadequate core cooling"?

Do you feel that such -- that there is a need for such instrumentation which will give advanced warning of inadequate core cooling and will cover the complete range from normal operation to complete core uncovery?

A (WITNESS JONES) Well, it is the B&W position right now, I believe, that the in-core thermocouples cover the range from normal operation to accident conditions, provide direct, unambigous, direct indication of inadequate core cooling, and also provides the approach to it. That is the B&W position right now.

DR. JORDAN: All right. Fine.

(Board conferring.)

CHAIRMAN SMITH: Did you cover complete core uncovery in your answer? Which was a part of the question, is the only reason I raise it.

WITNESS JONES: I believe the in-cores would also be satisfactory for complete core uncovery. The only hesitancy I might have on that would be if you really have the core totally uncovered for a prolonged period of time, I am not sure that the thermocouples could withstand the excess of temperatures which could result. That is my only limit on that.

DR. JORDAN: Okay. That is all for the present. Go ahead, Mr. Cutchin.

CHAIRMAN SMITH: How do you want to proceed, Mr. Cutchin, Mr. Adler, is there a preference?

MR. CUTCHIN: I have no preference.

CHAIRMAN SMITH: You have a preference, Mr. Adler?

MR. ADLER: Yes. Mr. Dornsife had prepared for this, so I prefer Mr. Cutchin to go.

CHAIRMAN SMITH: All right.

CROSS EXAMINATION

BY MR. CUTCHIN:

Q Mr. Jones, I will be asking you several questions to begin with, based on your written prefiled testimony. I will be referring first to page 4 of that prefiled testimony. In the middle of page 4, the second full paragraph reads that, "Obviously, the goal of assuring adequate core cooling is ongoing and is achieved by maintaining subcooled conditions in the reactor coolant system (RCS) or, in the absence of such conditions, by providing sufficient reactor coolant inventory."

How do you propose that reactor coolant inventory be measured after subcooling is lost?

(WITNESS JONES) Well, it is not really -- we are not proposing to measure reactor coolant inventory. Rather, what that statement is meant to mean is if you do not maintain subcooled conditions in the reactor coolant system, you should provide high-pressure injection to the system to maintain sufficient inventory.

The calculations show that if you keep the HPI pumps running, you would provide enough inventory to ensure adequate core cooling.

Q Now, referring to page 5, the parenthetical statement which begins at the end of the continued paragraph at the top of the page, it says: "At TMI-2 saturated conditions were indicated several minutes after the reactor tripped and if sufficient injection flow had been maintained, the core would have been adequately cooled and not damaged."

Do you not believe that had an indication of reactor water level been available, it might have aided the operators in recognizing that sufficient injection flow was not being maintained and thereby possibly enable them to have prevented core uncovery?

A (WITNESS JONES) Well, first off, you have to remember initially in the transient you have the reactor coolant pumps running for something on the order of the first 100 or 101 minutes, I believe it was. The reactor coolant pumps remained continuously operative, at least two of them. Under that situation, something that measures water level would have been quite useless because of the slow delta phi, et cetera.

I am not sure what kind of signal you would have got. You would have got a pressure drop across the core which would have had to somehow compensate out. And the fact that it was a two-phase fluid would have also had to be taken out. It would be difficult.

Now, after -- after the pumps were tripped, I am not so sure I could really say what the operators would have done. It might have been an aid, but, you know, the operator would have been faced at that time with a full pressurizer level, as he was, an indication that the core was uncovered, if he had a vessel level meter.

Which one of those two the operator would have then chosen to believe, I just cannot guess at. He might have chose the vessel level. He might have chosen to still believe the pressurizer level. So I really cannot give you a good answer. Just that it might have been an aid.

Q Now, you have seemed in several instances during your testimony today to base your measurement of core water level primarily on DT cells. And I believe in the document which was filed by Babcock & Wilcox, entitled "Evaluation of Instrumentation to Detect Inadequate Core Cooling," prepared for the 177 owners group, which was referred to this morning, you list a number of methods which were examined by B&W in their evaluation of instrumentation to detect core water level or inadequate core cooling.

Did you in that evaluation consider use of heated thermocouples?

DR. JORDAN: Was that heated thermocouples?

MR. BAXTER: Heated thermocouples.

WITNESS JONES: In the evaluation in the report that we referenced earlier, we -- I do not believe we examined -- looking at it, there is no statement as to that heated thermocouples were evaluated. I know of the concept of using heated thermocouples to detect water level in the upper plenum. And I believe some of the vendors are looking at this.

But, in general, it still comes down to the same thing. I do not believe that any level measurement technique offers any additional action or any additional benefit.

BY MR. CUTCHIN:

Q It appears that you at least -- the position that is being taken by the Licensee and B&W is that superheat detection via the thermocouples, the in-core thermocouples, is an adequate means of detecting inadequate core cooling. However, again, on page 5 of your prefiled testimony, you refer in the first full paragraph in the center of the page, you make the statement that, "As explained below, this indication -- meaning the superheated reactor conditions -- would allow time for corrective action before the limits of 10 CFR 50.46 would be exceeded."

You say, "Again, core water level instrumentation would not provide a basis for any additional action." And you point out that, "During the accident at TMI-2, indications of superheating in the reactor coolant system were available."

And I am asking the same question: Wouldn't it have been of some use in helping the operators to believe other indications that they had available to them but did not use, if they had had an indication of core water level?

A (WITNESS JONES) As I said, it is just highly speculative. I cannot really answer the question. It may have. And again, he may not have believed it, because it would have been an instrument that would have read off scale for -- during normal operation continuously, and one that may have moved around during the transient. I don't know whether he would have believed it or not. I just do not know.

Q Mr. Ross, as one responsible for operations, do you have an opinion on that question?

A (WITNESS ROSS) You must remember that the training evolutions we have gone through have been significantly different since the accident. Thermocouples were availble, but they were not trained on the thermocouples. Today they are.

I still feel with no training available, with the vessel level displayed to them, the fact that it did not move, did not respond when the plant did significant changes, like a plant startup or a plant shutdown, the operator would not have relied on that indication.

Q Mr. Keaten, just so you won't feel left out, this morning in your testimony, I believe, you indicated that the procedures for Three Mile Island Unit 1 have been modified to aid the operators in recognizing TMI-2 type events and the recent St. Lucie-type events.

Are you now comfortable with the belief that core water level indications would not be helpful to the operator in recognizing other types of anomalous situations that might occur? You seemed pretty positive that they would not aid him in recognizing those two. But are there others?

A (WITNESS KEATEN) It is, of course, very difficult to give a positive answer to a question which says is there some unknown thing out there. And so I think I can only answer in terms of the work that has been done, which has been to attempt to identify any use that we would make or, more accurately, any use the operator would make, of such information.

And as I said this morning, we have not been able to identify any circumstance where we could identify any action that he would take that is new or different compared to the actions that he is already instructed to take based upon the existing information.

I cannot sit here and swear that there is -- you know, that we have all knowledge in the world. Obviously, we do not have all knowledge.

My concern, though, is that I am reluctant to see

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us install any instrumentation that we do not know how to use it, when at the same time we are installing something that we do know could be confusing to the operator and at least has the potential for leading him to take the wrong action.

MR. CUTCHIN: Thank you. I have no further questions of these witnesses, Mr. Chairman.

CHAIRMAN SMITH: Mr. Dornsife.

MR. DORNSIFE: I just have to say, Mr. Chairman, you paid me back this morning; you destroyed my cross-examination plan.

CHAIRMAN SMITH: I am sorry, I did not hear Mr. Dornsife.

MR. DORNSIFE: I said you paid me back this morning; you destroyed my cross-examination plan.

CHAIRMAN SMITH: As a matter of fact, no. There were questions that Dr. Jordan pointed out were on your cross-examination plan. So we will complete it if you wish.

MR. DORNSIFE: I still have questions.

(Laughter.)

MR. CUTCHIN: I have the same problem. The Board saved me a lot of time as well, Mr. Chairman.

BY MR. DORNSIFE:

Q I have a general question to start off. Maybe Mr.

Keaten is best able to address this. Your scheme of things, the way you envision now with the procedures and instruments that are available, is it true that the t-sat meter is relied upon to detect approached inadequate core cooling and the thermocouples themselves are used to determine inadequate core cooling is taking place? Is that basically how the scheme works?

A (WITNESS KEATEN) It is along that line. But let me be careful in the way that I use the terms. If we define, as has been done in the testimony here, inadequate core cooling to refer to that condition in which the fuel temperatures exceed the limits of 10 CFR 50.46, then the inadequate core cooling guidelines, as they appear in the TMI procedures as taken from the B&W guidelines, are used as an indication of the approach to inadequate core cooling.

They show conditions where the fuel temperature is higher than would have been predicted from the LOCA analysis but still below the limits imposed by 10 CFR 50.46. And the actions that are called up in the inadequate core cooling guidelines are designed to prevent the temperatures from reaching or exceeding the limits of Part 46.

Still using the terms in the same way then, I think it would be accurate to say that the t-sat meters are used as an indication that there has been a substantial loss of reactor coolant inventory either due to shrinkage or due to a LOCA or whatever.

That requires the operator to add coolant to restore the normal subcooled conditions in the reactor coolant system. This may be done automatically by the emergency core cooling system, or it might be done manually by the operator, depending upon the particular event.

So, really, in summary then, I would view the t-sat meters as the indication that when the subcooling margin is lost, that inventory needs to be added and the core thermocouples as an indication that for an unknown, undefined reason that the ECCS system is not operating as designed and that there is starting to be an approach in the direction of inadequate core cooling.

Q Mr. Jones, are you generally familiar in detail with the TMI-2 accident scenario? Thermohydraulically, anyway?

A (WITNESS JONES) Yes.

Q Maybe it would help to have you describe when -- if these instruments were available during the TMI-2 accident, when they would have indicated that the condition would have existed, like, for example, when the t-sat meter would have first identified that you are approaching saturation conditions and when if a water level instrument were available, when that would have first indicated that the core level was going down and when the thermocouples first started going -- increasing in temperature, and the whole -- all the instruments that are now available, how would they have responded?

A (WITNESS JONES) Okay, let me take a crack at it.

Well, following the initial event, the primary system remained subcooled for about five to 5-1/2 minutes, something on that order. At that point in time, saturated conditions were reached in the hot leg, indicative of something that inventory needed to be restored.

Now, all during -- from the beginning of the accident to about 100 minutes into the accident, the reactor coolant pumps were operating. And with the reactor coolant pumps operating, I do not believe any level detections came, such as a water level based on the DT cell or something of that order, or a heated thermocouple. I am not so sure either of them would have picked up any core or loss of coolant inventory in the system because the pumps were on.

Now, 100 minutes after they tripped the reactor coolant pumps, at that point in time, I believe what you would have seen if you had a level instrument, you would have seen that you had a bubble somewhere within the reactor vessel head, probably down near the hot leg nozzles, maybe even a little lower. The hot leg thermocouples, or RTDs, at that time were reading saturated -- the in-core thermocouples, although there were no readings available, were also going to be indicating saturated conditions at that time.

Now, it is debatable as to exactly when the core uncovered. So I cannot give you a good estimate on when the various instruments would have indicated superheated conditions. But I would estimate that certainly be' . e 110 minutes; that is about 10 minutes after the reactor coolant pump trip, the hot leg RTDs were indeed superheated.

Now, between 100 and 110 minutes, it is possible that the in-core thermocouples would have read superheated conditions. In the -- well, it would definitely have, and it probably would have been earlier than the hot leg RTDs due to the transport time differential to move from the core up the hot leg to the hot leg RTD.

At that point in time, you would have had superheated conditions, and the operator would be starting to use the inadequate core cooling guidelines. As far as thereafter how the instruments would have read, it is difficult to say. But they would have passed the various threshold sometime over that period.

Q Once the core heat-up began, how long did it take before fuel damage started occurring from the time heat-up first began? Do you have any idea?

A (WITNESS JONES) There are various estimates, but I believe in general it would be about 40 minutes, somewhere about 30 to 40 minutes after the core uncovered. You probably have some fuel damage; that is, ruptures of the cladding.

Q Was that the 1400 degrees you were talking about basically?

A (WITNESS JONES) Basically around 1400 degrees, something on that order.

Q It starts occurring at 1400 degrees, but it is not -- it is not cladding reaction with water, is it?

A (WITNESS JONES) No. To try to define that, the 1400 is more of a ballpark number. What you have is you have a stress built up across the fuel rod from the internal pressure of the rod and the external pressure in the reactor coolant system. If the temperature reaches a certain point, the stress temperature combination reaches a certain point, the cladding will rupture

Somewhat previous to that it will start to expand and become caustic, and it will -- shortly thereafter it will rupture if it continues to heat up.

Q I guess the obvious next question is why then is the limit 2200 for ECCS analysis and not 1400 where damage begins to occur? What is the differences? Why the higher limit for ECCS?

A (WITNESS JONES) Well, the ECCS limits deal with the ultimate structural integrity of the cladding. Okay. The fact that the cladding ruptures and balloons and can cause some core blockage is not necessarily an unsafe condition. You can cool the core in a blocked condition, and in fact you are required to analyze by Appendix K, Cladding Ruptures and Subsequent Feedback on the Transient and the Clad Heat-up; 10 CFR.46 limits that deal with basically the structural integrity of the cladding are the 2200 degree temperature limit, in combination with the 17 percent local oxidation limit of the cladding.

What those two in combination mean is that if you satisfy those two and thermally shock the cladding -- that is, suddenly cover the cladding with cold water, below those thresholds, the combined threshold, the cladding will remain ductile and will not shatter, become embrittled.

Above that it has a potential for embrittlement. At the time that was essentially an embrittlement limit today -- based on knowledge today, they are a very conservative set of limits, but that was the basis for the limits at that time.

Q In your analysis of inadequate core cooling, did you come across any potential scenarios where the margin to saturation would not be an adequate indication of actual conditions in the core? Is there any possible scenario where that could give you a fleeting indication?

Like, for example, could pockets of hot air trapped near the RTDs give you an accurate reading of the actual conditions?

A (WITNESS JONES) Well, for the initial indication of saturated conditions, no, I do not know of any situation where you get high temperature readings locally, excessively high temperature readings locally at the hot leg RTD that would not be indicative of overall system saturation, or at least a good portion of the system being saturated.

If you had gone through a complete accident and had come through, uncovered the core, went through an inadequate core cooling situation and then recovered the core, it is quite possible at that time that the Haflich RTD could be reaching superheated conditions due to a gas trap, such as hydrogen generated from metal-water reaction, where it would be reading, say, superheat while the core exothermocouples would be reading saturated, and in fact, the recovery process is more based on the getting the transfer from the inadquate core cooling guidelines back to the normal small break guidelines based on the recovery indications from the in-core thermocouples, not the hot leg

RTD.

Q How about the other way around? Are there any possibilities of the core being saturated, but the RTD's seeing sub-cooled temperatures?

A (WITNESS JONES) It is possible that you could have local regions in the core be in a saturated condition, such as the hot rod being in somewhat saturated conditions while you have subcooled indications on the hot leg RTDs. But that does not mean it is not providing adequate cooling, but that is a situation there that you could -- is possible.

I am not sure about ATWS, anticipated transients without scram. It may be possible in those type of transients where you could get into that type of situation, voiding in the core while at the same time being subcooled on Haflich RTDs.

I would expect that to be somewhat momentary. There are probably other reactivity type control rod withdrawal transients or control rod ejection type transients, where again you may get the local boiling within the core, yet still see subcooled conditions on the hot leg RTDs, but for that case, like the other, that would still provide adequate cooling of the core.

Q How about if noncondensable gases were saturated? How would that affect the various -- especially the pre-fat indication?

A (WITNESS JONES) Well, I think -- that is what I was getting at before. If you went through an inadequate core cooling situation and a generated hydrogen or you had some other noncondensable gases in the system, it is possible that that gas will form a block in the reactor coolant system such that as the water level tries to rise, it is sitting at the RTD and would be reading superheat. Yet the core could indeed be covered with water.

Q Are any of you gentlemen familiar with some of the interviews of the operators concerning why they did not believe the hot leg temperature readings?

A (WITNESS KEATEN) Yes, I am.

Q Are you aware specifically of one of the interviews with Leland Rogers, the B&W representative who said the reason he did not believe the hot leg temperatures is because he saw those temperatures higher than saturated conditions of the previous time.

Do you recall any of that interview?

A (WITNESS KEATEN) I am not specifically familiar with the interview with Lee Rogers. No, I am not. For the benefit of the Board, he was a B&W representative at the plant.

I have talked to some of the Met Ed operators.

Q Can you maybe give us some analysis of why they did not believe it?

A (WITNESS KEATEN) Of the persons that I have talked to, the majority of them have said that they were not aware of the actual readings, although they were aware that the computer printout was showing question marks.

Q I am talking about the hot leg temperatures.

A (WITNESS KEATEN) Oh, excuse me. I am on the wrong topic. Pardon me.

Q Wasn't the range of the TMI to hot leg RTDs up to about 800 degrees?

A (WITNESS KEATEN) There was one that went up that high, but the normal range was not that high. It is 620. Actually, the operators that I talked to that were in the control during the critical portion early in the accident -- I have not had any of them tell me that they did not believe the higher readings.

You have to realize that Mr. Jones has said for the first 100 minutes when the coolant pumps were still running, the coolant was in fact being cooled and the temperatures were normal. When the operators turned off the final pump in the A-loop in an attempt to establish natural circulation, shortly thereafter the temperatures in the hot leg RTDs started going up, and the operators did see this and they did attempt to respond to it.

They tried to respond to it by attempting to restart the reactor coolant pumps, recognizing that something adverse had happened when they turned off the pumps, and this is where they got into the situation where they found they could get the pump to turn.

But they had no indication there was pumping fluid, so they were in fact trying to respond to those elevated readings.

The other thing is that the operators have testified that in the timeframe where they turned o[illegible] the reactor cooling pumps -- and we do not have accurate data on this -- that in this timeframe they did take action to reinitiate the emergency core cooling flow.

Now, this is a portion of time where we [illegible] missing some of alarm summary data, and so I do not know times and intervals, but this is what the operators remember that they had done.

So I think a reasonable conclusion is that they did believe them, at least in the sense of recognizing that they needed some actions that they were not taking.

Q The main problem, they were just not trained to do the right things to respond to that.

A (WITNESS KEATEN) That is certainly a part of it. I think the other part of it is that by this time sufficient inventory had been lost that the actions were not as clearcut as they would have been much earlier in the transient.

Q Going to the t-stat meter itself, will this be a safety grade instrument, and if not, what is lacking on being a safety grade instrument?

A (WITNESS KEATEN) Let me double-check.

(Pause)

In general let me tell you that the instrument is not fully safety grade, but it is sort of the category that I might characterize as nearly safety grade.

The equipment will meet the environmental design conditions. It is seismic.

DR. JORDAN: It is what?

WITNESS KEATEN: Seismically qualified. The separate requirements are met.

DR. JORDAN: So it is redundant?

WITNESS KEATEN: It is redundant. There are two t-stat meters, one for each of the hot legs.

BY MR. DORNSIFE:

Q Are there two separate readouts or just one?

A (WITNESS KEATEN) There are two separate readouts. The equipment in the cabinets -- that is, the electronics that drives the output and so forth is in fact nuclear safety related and Class I-E, but I believe the difference is in the indicators and perhaps the cabling from the signal conditioning equipment to the indicators.

I do not believe that those are safety grade. But they are very high quality equipment.

Q I guess that was my next question. These are digital readout?

A (WITNESS KEATEN) These are digital readouts.

Q Are these the only digital indicators you have in the control room?

A (WITNESS KEATEN) No, they are not, and certainly new equipment that we are installing, there is more than one digital.

Q I am wondering how a digital meter compares in reliability, say, to another type of meter, that that is a hard-wired type of device. Is there any difference in reliability between those digital and any experience in those types compared to a gauge, let's say, or something -- something on that order that is more familiar on control room panels.

A (WITNESS KEATEN) I am not terribly knowledgeable in this area, Mr. Dornsife, but it is my impression that the model digital display meters that can be purchased now are in fact a high reliability display unit, and they have a tremendous advantage in terms of the precision with which they can be read, and that is the reason that we are going to digital indicators in several of the applications.

Q I assume from human engineering, too, they are easier to read.

A (WITNESS KEATEN) Yes, that is true, particularly where it is necessary to read them accurately.

Q Why was the 5^ degrees subcooling chosen for the limit for the procedures? What is the basis of the 50 degrees?

A (WITNESS JONES) I cannot remember the specific numbers that went into coming up with this, something on the order of 50 degrees, but basically what it was was we wanted to assure an indicated or a real subcooled margin of at least 10 or 20 degrees.

I cannot remember the exact number, but we wanted to be several degrees away from the saturation line, and then on top of that, what we placed was instrument errors for the hot leg RTDs and the system pressure indication, which translates back into a temperature -- potential temperature error.

Those were utilized, and I do not remember how the exact number came out, 40 or 45. It came out in that ballpark for the instrument we were looking at -- using, so we chose 50 to be a nice, round number ultimately.

Q You say in your testimony there is an alarm associated with a t-stat meter. Is that alarm on the main alarm panel, and is it going to be the one -- one of the alarms that is especially coded?

A (WITNESS KEATEN) Mr. Dornsife, you are right at the limit of our design. As a matter of fact I was on the telephone on exactly that subject during the luncheon break. It is my present anticipation that that probably will be one of the ones that is color-coded. It will certainly be enunciated in the control room, and we are still in the process of debating just where it is.

Let me also take the opportunity to add to my previous answer. In looking at the information I have with me with regard to the extent that it is safety grade, I believe that it is also true that at least in the modifications that we are doing prior to restart, that the centers that we are using for the saturation are perhaps not safety grade sensors, although the new cabinets and signal conditioning equipment that is being installed are safety grade.

Q You say the sensors?

A (WITNESS KEATEN) Right.

Q RTDs?

A (WITNESS KEATEN) RTDs and/or the coolant pressure. You understand the in the system there are both safety grade and non-safety grade RTDs and pressure signals and so forth. And I believe from what I can read in the STD here that the sensors that are being used at the present time for the t-stat meter are non-safety grade sensors.

Q So they are not the same -- the same sensors that go to the safety system?

A (WITNESS KEATEN) That is correct.

Q Was there a problem with making the same sensors as far as saturation? Is that why you chose non-safety sensors? It would seem that this instrument has to be a very reliable insurance because it is the basis basically for the small break loss of coolant procedure, and it should be a very important instrument.

I am wondering why you chose not to use safety-related sensors.

A (WITNESS KEATEN) I think there are probably a couple of answers to that. One is that as you can understand, in tying into equipment which is presently going to the protection of the system, it is necessary to be extra careful and make sure nothing is done that would interfere with the capability of the protection system to perform its function.

Furthermore, in this particular case -- and maybe others could answer clearer -- it is my impression that the non-safety grade sensors and the 50 grade sensors are pretty much the same thing, other than perhaps paperwork and so forth.

It is certainly true that we want this to be a highly reliable instrument, and we indeed to have it a highly reliable instrument, but let me also point out that if for some reason this meter or meters are not available, the operator still does have available to him the safety grade temperature instrument and the safety grade pressure instrument, and he can himself calculate saturation margins so there is a safety grade fallback position if this meter were not available.

DR. JORDAN: Well, Mr. Keaten, in that connection wouldn't it also -- isn't it also designed to have the t-stat meter completely separated from the meters which tie into the safety system in that the operator is controlling from the t-stat meter and to have that tie in with the same meter that are used in the safety equipment seems to me would be a gross violation of the separation of safety equipment from operating equipment.

WITNESS KEATEN: I think it is correct, Dr. Jordan, that there are some advantages to having separate instrumentation. It is my understanding of the regulations that in fact it would be possible to have the t-stat meter use the same sensor using appropriate isolation devices and so forth in a way that would meet, for example, IEEE-279.

But personally I am just as happy to have it use a different sensor.

DR. JORDAN: I guess you are right, because I remember now complaining about the integrated control system also getting information on power from the meters that also provide for the scram, and to my mind I think that is a violation of 279.

BY MR. DORNSIFE:

Q How about power supplies to these sensors? How reliable are they?

A (WITNESS KEATEN) They are powered from either onsite or offsite power, so loss of offsite does not lose the function.

Q Are they used for anything else, these sensors? Are they tied to the ICS or something?

A (WITNESS KEATEN) I think they probably are used for normal indication. I am sorry. I just do not know whether these are the same ones that feed the ICS or not. They might be the same ones that go to the ICS.

Q About the backup capability of determining t-stat on the computer, what sensors does that use? Do you know? Are they the same, or are they the safety-grade sensors?

A (WITNESS KEATEN) I believe that our intent had been to use the same sensors to feed the computer as is feeding the t-stat meter, to independently calculate the saturation margins so we would not use the output from the saturation margin meter itself, but would use the same temperature and pressure information to calculate the saturation margin.

DR. JORDAN: Mr. Dornsife, I guess I do not know how a t-stat meter works. know it works with indications -- taking as input the pressures and temperatures, of course. Is there a simple -- could you describe simply how it works, because I notice, for example, in some of the -- either the questions that were asked somewhere or other, there was a problem raised as to whether you were approaching from the pressure side or the temperature side, and it reminded me that I did not understand exactly how the t-stat meter worked.

WITNESS KEATEN: The ones we are installing prior to restart, Dr. Jordan, are meters which will indicate the temperature saturation margin, and so basically the way the meter works is to take the reactor coolant system pressure and calculate the saturation temperature associated with that pressure, and then subtract that from the actual measured temperature.

DR. JORDAN: I see. Where is that calculation done?

WITNESS KEATEN: It is done in a separate electronic unit that goes with the equipment.

DR. JORDAN: I see. It has the steam tables built into it.

WITNESS KEATEN: Yes, sir, and I'm not sure whether it is a digital or analog calculation.

DR. JORDAN: Okay.

BY MR. DORNSIFE:

Q The next line of questioning I have involves the core exit thermocouples. First of all, what -- what typically is their range? What will be their range prior to restart of the instruments? What is their accuracy?

A (WITNESS KEATEN) I believe to the best of my memory that the computer software that we are using to translate from mid-evals to degrees Fahrenheit goes to -- it is on the order of 2200 degrees Fahrenheit. Maybe it is 23. I am not sure exactly, but it is of that kind of number.

Q Do you have any feel for their accuracy?

A (WITNESS KEATEN) These are standard type K-chromel-alumel thermocouples, and my experience with them has been that the accuracy is very good, assuming that the thermocouple is working.

I am not sure I have a good number to give you, frankly.

Q Is the only way to read them from a computer? Are there any other ways of reading the temperatures besides the computer?

A (WITNESS KEATEN) The only hard-wired method that we have at the moment is to the computer. However, it is possible for the operator to take a portable meter and go to the terminal board, for example at the input of the computer multiplexers and read the millivolts there, and convert that using the table to temperature.

Frankly, we are right now in the act of evaluating just how quickly and readily that could be done, and whether or not there may not be another method.

Q That was my next line. I was wondering how much you were going to pursue that considering that was a major problem during the TMI 2 scenario, people believing those readings and actually taking the volt meter readings from the hard-wired portion and transferring them to numbers, and making any sense out of them. I wonder how far you are going to pursue that.

A (WITNESS KEATEN) We are going to pursue it. I think there are two different reasons why we are going to pursue it. One is our own incentive. I have, I think, in earlier testimony told the Board that we tried not to be in the situation in TMI where we relied on the computer as the only method of getting information.

Here is a case where we are tapping on the operators' being able to read the in-core thermocouples and so, in line with our own philosophy, we are going to explore how practical it is for him to do it with the portable meter, or whether there might not be another method.

In addition to our own requirements, I might point out that new reg 737 imposes requirements for a backup method of being able to read at least a portion of the thermocouples and also specified a time within which a certain number of thermocouples must be able to be read.

And so we will also be addressing that requirement in new reg 737.

Q That was my next question, since you brought it up. Are you going to be meeting all the requirements on Attachment A in new reg 0737? Are you committing to do that for the in-core thermocouples?

A (WITNESS KEATEN) Honestly, Mr. Dornsife, we are right now in the process of evaluating those requirements. As you are aware, Attachment A has quite a few different requirements. We certainly will have to address all of those requirements.

There are a couple of them that I in my own mind right now am not sure how to meet, so we are presently -- have already had dialog with the NRC staff on the subject of those requirements, and the subject of the schedule on which we would be meeting the requirements.

We are starting to address them.

Q Is it your position at this point that these in-core thermocouples would be he principal way of detecting inadequate core cooling?

A (WITNESS KEATEN) Yes. Yes, that is our position, but let me also state again, as I am sure you recognize, that inadequate core cooling scenario that we are talking about is an event beyond the design basis of the plant, and -- but in that case, yes, we have given instructions in the procedures that the operators should monitor the in-core thermocouples as a method of detecting that non-design basis event.

Also, I would like to -- just to make sure the record is clear -- as far as the problem that was faced by the operators in using the in-core thermocouples during the TMI 2 transient, the problem was the computer software which had an algorithm that translated millivolts into temperature readings only up to a temperature of 700 degrees Fahrenheit, and that of course has been corrected at TMI 1.

Q So is that basically the only difference between the thermocouple installation on TMI 2 and TMI 1, just the fact that you extended the range on the computer? Are there any differences besides that?

A (WITNESS KEATEN) In terms of changes that we have made, I would say, no, not really. The situation is a little bit different between the two units in that as you may remember, the in-core thermocouples in unit 1 were not connected up prior to the accident.

And so we had to go through the act of rewiring them into the control room and connecting them to the computer, which had been done at TMI 2 and not been done at TMI 1. In the process of doing this, there were some details that resulted in some differences in the installation, the physical installation, particularly the fact that the thermocouples at TMI unit 1 did not terminate in a reference junction, as they had at -- or a cold box as they had at TMI 2.

Therefore, for the installation at TMI unit 1, we measure the temperature at the physical location where the reference junction or the transition from chromel-alumel is located, and then we analytically do the calculation to correct for the temperature at that location, rather than having a fixed reference junction, as TMI 2 had.

But these are the details of the installation. As far as the readout in the control room, it is via the computer as far as the reference readout is concerned.

Q As far as the reliability, are they capable of being powered from onsite and offsite power supplies, and generally a very reliable piece of instrumentation?

A (WITNESS KEATEN) The thermocouples themselves, in my experience, are very reliable pieces of instrumentation. They are among the most reliable pieces of instrumentation.

The power supply consideration really comes up in association with the computer, and yes, it is capable of being powered from either onsite or offsite powers.

Q Thermocouples themselves have no requirement for power. They are self-powered devices.

A (WITNESS KEATEN) That is right. They do generate the millivoltage themselves. My only uncertainty in that is I am not exactly sure what we are using to measure the temperature at the position of the reference junction, and if that is an RTD it would require power.

Q Do you have any idea what the design, the maximum design temperature gives when they would start melting and no longer be giving useful information?

A (WITNESS KEATEN) It is my understanding -- in fact, it is my experience that type K thermocouples have been used usually inadvertently to measure temperatures as high as 1,500 degrees Fahrenheit, although that is certainly well above the normal high range of such thermocouples.

My memory is that something around maybe 1800 degrees is considered the normal upper limit. Even that is pretty hot.

Q The last instrument you identified then as being an additional way of determining inadequate core cooling is the expanded range, RCS hot leg temperature. Is this instrument separate from the p-sat hot let temperature instrument?

A (WITNESS KEATEN) It is certainly a separate instrument, and neither myself nor Mr. Ross are 100 percent certain whether it uses a different sensor or not. I do know that it is not one of the safety grade key hot sensors.

Q It is -- its range, I believe, was 620. Is that not correct?

A (WITNESS KEATEN) Yes, that is the range as it was, right.

Q On page nine of your testimony, the second full paragraph, right near the end, you kind of summarize what you said previously, and you say that all these instruments are useful -- I am paraphrasing now -- are useful for identifying when inadequate core cooling is approaching, and specify the operator action necessary to properly enhance core cooling.

Specifically, I am looking at the ways of properly enhancing core cooling. Are there any safety grade systems or components that are called out in the procedures to enhance core cooling when you get into this situation? Are they all on safety systems?

A (WITNESS JONES) Well, the objective of most of the actions which are being taken to bring the -- bring the system pressure down to allow other safety systems to actuate, but in general we are relying on non-safety pieces of -- non-safety grade pieces of equipment, although like the emergency feedwater system which is a safety grade system from the standpoint of a LOCA.

Q But that by itself would not -- you would need things like the atmospheric dump valves and the condenser bypass valves.

A (WITNESS JONES) That is correct. What I'm saying is although we are using non-safety grade pieces of equipment, some of the support equipment can be safety grade, but in general we are relying on non-safety grade equipment in that period of time.

Q On that same page of your testimony you indicate three -- four ways of determining or four actions that are necessary for inadequate core cooling and I am wondering are procedures -- are these actions alternatives, or are they all in sequential steps?

MR. BAXTER: Excuse me. I think Mr. Dornsife said these are actions for inadequate core cooling. I think the testimony is they are actions taken during an approach to attempt to avoid it.

WITNESS KEATEN: Yes, and let me expand on that, but first to answer your direct question, no; the initation of high pressure injection, the maintenance of the correct generator level, the tripping of the reactor coolant pumps -- if there is an engineered safety feature actuation and monitoring the exit core thermocouples are actions which all need to be taken. It is not one of them but all of them.

Let me also then call your attention to the final sentence, which is that no further action is requried for design basis accidents. In other words, this is not really addressing the approach to inadequate core cooling that would go beyond the normal LOCA analysis, for which we would be taking the actions Mr. Jones was describing, such as the pressurizing of the steam generator.

These are the acts associated with a LOCA which is within the design basis.

BY MR. DORNSIFE:

Q The next page, the bottom of page 10 and the top of page 11, these are in fact steps for inadequate core cooling. Is that not correct?

A (WITNESS JONES) Yes.

Q The same question. I would like to ask the same question. Are these alternatives or are they sequential? Is the operator first instructed through the first step and then to proceed through the rest of the steps? Or if the one were to be stopped at that particular step?

A (WITNESS JONES) No. These have been constructed for the operator to do them all basically. The order is really not that important in my mind, with the exception of like, on page 10 I list four items, but most important are one and two, and I am not really -- it does not really bother me if they are done in a different order.

The opening of the PORV, it is listed as an as-necessary step, so I would not want to open to the PORV to depressurize the plant. I would rather use the steam generator first, and then the cooldown with the steam generator after you continue the initial action, but the first two could be interchanged.

Q Based on the analysis that you did following the TMI 2 accident, what in your opinion is the reason the plant could not be depressurized using the PORV? Was it because the non-condensable gases primarily that were generated?

A (WITNESS JONES) Could you reference me to a time period?

Q When the operator is following what is assumed to be initial core recovery and then attempts to go to a decay heat removal recirculation cooling? When they tried to reduce pressure by opening PORV, why couldn't they achieve the conditions necessary to activate that system?

A (WITNESS JONES) Okay. I would not necessarily believe it is the non-condensable gases. I just think simply what you reached at that point in time was a matter of -- as the system pressure decayed, you get less and less energy removal or steam or non-condensable gas removal out of the valve while you still are creating steam in the system.

And then you reach the balance pressure, at which flowout and steam generation more or less equalize.

Q So that is a reason why one of those steps previously is not sufficient. You would need more than one because a PORV itself would not necessarily be adequate to do that.

A (WITNESS JONES) Correct.

Q I have some questions now on potential vessel level indications and type of indications. Mr. Jones, are you -- are the positions of the licensees at the B&W plants similar to Met Ed's in that they feel that the t-stat meter is totally adequate; they do not need to add any additional instrumentation?

A (WITNESS JONES) As far as I know, right now the position of all of the B&W operating plants is not to add additional instrumentation?

Q How about other meters?

A (WITNESS JONES) I am not really that familiar with what other vendors or what other plants are utilizing. I have heard that Combustion Engineering -- and they have recommended to some of its operating units that they put on heated junction thermocouples, but that some of the units are rejecting them because they just do not see a use, and they do not think they really work.

They had questions about whether they were a good instrument or not, so I do not think there is a clear picture. I don't have a real good picture as to what all Combustion Engineering plants are doing.

Q Mr. Keaten, you had described the two ways you are familiar with that people are looking at are the heated thermocouple that Mr. Jones just described at a differential pressure measurement.

A (WITNESS KEATEN) Excuse me. I believe my reference, or at least what I intended to have said at the time was that those were the two methods that one of the NRC staff members had described to the ACRS as being the method that the staff thought might be the near term candidate for being qualified in the eyes of the staff.

We have also looked at some other approaches, such as the use of neutron detectors, for example. We have considered the possibility of sonor type devices, but we have not found any that appeared to us to meet the reasonable criteria of being reliable and unambiguous.

CHAIRMAN SMITH: What is the priciple of the heated thermocouple?

MR. KEATEN: The idea, Mr. Smith, is that if a thermocouple is provided with a current that is flowing through the junction, as long as it is covered with water the water will keep it cool, and so it will read the normal temperature of the water.

If the water is then gone, the thermocouple will start heating up, indicating a loss of water.

DR. JORDAN: What happens if it is in two-phase fluid?

WITNESS KEATEN: Dr. Jordan, I think that is the sixty-four dollar question. My interpretation of one brief sentence that appeared in the report of the most recent LOFT tests, in which they had heated thermocouples in the upper region --

DR. JORDAN: Which report?

WITNESS KEATEN: This is a report on the most recent LOFT test.

DR. JORDAN: LOFT. Thank you.

WITNESS KEATEN: It was that -- as I say, my interpretation of this brief sentence was that the heated thermocouples did not work to indicate level or equivalent level in the two-phased mixture that existed while the reactor coolant pumps were in operation during the test, and it was only after the pumps were turned off that the heated thermocouples gave a good indication.

But I emphasize for the third time that that is one sentence that I saw. I am certainly not prepared to support that in any depth.

DR. JORDAN: One would think that a heated thermocouple woud almost be the ideal instrument because it essentially simulates the situation in the core. You have heat being generated, heat being removed, and if you put a heated thermocouple in the same medium, then it ought to respond very similarly.

WITNESS KEATEN: I think there may be a problem, Dr. Jordan, with respect to the relative heat fluxes. In the case of a heated thermocouple you have a low current and therefore a relatively low heat flux, whereas -- I am thinking of heat flux in terms of transfer to the surrounding medium -- whereas in the case of the fuel rods I would expect that the heat flux would be considerably higher.

DR. JORDAN: I would have thought that rather than -- well, if you are speaking of flux as kilowatts per square centimeter, wouldn't they be somewhat similar because the thermocouple is such a small --

WITNESS KEATEN: My understanding is -- I would not have thought so. Maybe I really better say I do not know.

DR. JORDAN: All right. Of course.

WITNESS JONES: There has been some further work -- I have seen some of the work that has been done on the heated junction thermocouples, and they have found some problems with them to date, especially at high pressures where the steam and water density were close at high pressures. They have trouble in telling when it is steam or water. The type of changes that they see are very small, and they get some noise with these heated junction thermocouples because you do have some spattering of water here or there and it is somewhat indistinguishable from the noise.

DR. JORDAN: It becomes more and more difficult.

WITNESS JONES: Yes. This was my information. It is something on the order of two or three months old, but what they showed was that under certain conditions they seemed to be pretty good, but as the pressure got up they became more and more questionable. The last I heard, it was still under development. They were still being examined.

BY MR. DORNSIFE:

Q This proposed device, would it be a thermocouple on both levels? Is that the principle?

A (WITNESS KEATEN) If the idea is to use it to measure level, then that is how it would have to be done; that is correct.

Q So you probably need a large number of these thermocouples to get a fairly good range of levels.

A (WITNESS KEATEN) There is a trade-off between the range, the number of thermocouples and the resolution. If you are happy with one every foot, well, then you measure 20 feet with 20 thermocouples.

Q Other than the R&D work, are there any applications that you are aware of that actually use these devices or any other devices for level measurement in this type of an environment where you have a large density difference, possibly, heat source, that type of thing?

I guess my question is: is the state of the art still really a differential pressure meter? Is that really the state of the art right now in the industry or all industries that you are aware of?

A (WITNESS KEATEN) In the kinds of environment that we are talking about in a pressurized water reactor, I am not aware of anything different from a differential pressure meter which has been used in a production reactor. In other types of environments, of course, other things have been used, but I am not aware of any in a nuclear environment.

Q If you were to install a differential pressure device on the reactor vessel, would it require additional penetrations in the reactor vessel?

A (WITNESS KEATEN) I am not sure I have a completely unequivocal answer to the question. It is my impression, Mr. Dornsife, based upon some early work that was done, I think partly by B&W and partly by ourselves, that some type of differential pressure indicator probably could be installed using existing penetrations. What I am not sure about is just which range it would cover.

Q I guess, for example, there is probably drain devices on the bottom of the vessel that could be used if you wanted a full indication. Is that correct?

A (WITNESS ROSS) By design the vessel is designed to be undrainable and there are no drain connections on the bottom. There are other connections that could possibly be used, like in-core thermocouple connections.

A (WITNESS JONES:) Well let me, just to bring maybe a little more closure to this, when we looked at the use of differential pressure transmitters in that August 15 report, it states directly there that if you put a total reactor vessel differential pressure measurement device, you would require new penetrations in an in-core nozzle and one at the top of one of the control rod drive mechanisms. You would have to make some new penetrations.

Q Would that cause any safety problems, in your opinion, having that type of an open penetration? For example, let me give you a scenario. Let's say you had a differential pressure device having a small instrument tube on it and that tube were to rupture and you had a leak at the bottom of the pressure vessel. Would that be, although small, an accident or a leak that could cause problems beyond the design basis?

In other words, if you wanted to refuel to get a stable condition, you would have to flood the containment up to the top of the fuel.

A (WITNESS JONES:) If you got a break, if had that pipe break and the size of the break was one in-core nozzle, that would be capable of being handled by the makeup system and retain the system in a solid configuration. You would ultimately to refuel probably have to fill the cavity, as you state. If it was several in-core nozzles, I am not sure what kind of problems that would cause.

Q But wouldn't, though, based on the current configuration of the in-core thermocouples, wouldn't having a pressure sensor be a different configuration whereby you would have pressurized water outside the vessel compared to just the tube, the nonpressurized tube, so it would present some special problems that -- it is not a pressurized instrument outside the vessel, is that correct?

A (WITNESS ROSS) The in-core itself is a pressurized vessel and the pressure boundary is located outside of the reactor vessel, that is correct.

Q So it would have the same potential problems for a break.

A (WITNESS ROSS) I could not say it would have the same potential. I said the potential would exist.

A (WITNESS JONES:) Now, let me just make one comment. I get the impression from your question that use of these in-core thermocouples are heated junction type thermocouples. If you are trying to get at those as not requiring additional penetrations, you know, a unique feature of the B&W design -- you know, like we have our instrument tubes come from the bottom of the vessel up, while the Combustion Engineering design, their instrument tubes come from the top down, and it is one of the reasons why they are able to recommend such an item, because they already have the existing instrument tubes and penetrations available.

But something like a heated junction thermocouple installation would cause problems in that it would require potentially additional vessel penetrations for us.

Q Mr. Jones, didn't you say you could potentially use one of the thermocouple nozzles for a differential pressure meter? That is where I was getting at, not the --

A (WITNESS KEATEN) Yes, potentially you could. It is at the bottom of the vessel. I think the point is that while you could use that one for a DP-type indicator, since it terminates basically at or just above the type of the core, you could not use it for a heated thermocouple that was up higher.

Q You also could not use that for an indication at the bottom of the core; is that also true? There is no way you could use that thermocouple without removing some of the internals as an indication of the full range of the vessel. You would have to have an additional penetration then for that type of an instrument.

A (WITNESS KEATEN) I think that it would require some significant changes in order to use that type of an instrument over the full range. I am not sure we know exactly what those changes are. But certainly the reactor was not designed with that in mind.

A (WITNESS JONES:) If we used in-core thermocouples inside the core, you know, by somehow putting them in the instrument guide tubes, they would not have to be heated junction thermocouples because they would see directly the hot steam because the core is getting hotter as you are progressively going up from the two-phase mixture if you are in this type of situation, while what the present utilization concepts of the heated junction thermocouple is to try to tell where the water level is from the top of the core to the top of the vessel head, where is it steam and where is it water; and again, for us that would require at least additional penetrations.

Q Are you familiar with the way the BWR measures vessel level, any of you?

A (WITNESS KEATEN) Yes, I am.

Q Is it a direct measure of the core level or an indirect measure of the core level?

DR. JORDAN: Did you say BWR?

MR. DORNSIFE: Yes, boiling water reactor.

WITNESS KEATEN: I am going to give my answer in the context of Oyster Creek, which is a specific BWR that I am familiar with, and actually there is more than one level measurement. The ones which are used during normal operation and for most of the safety system actuation actions are a differential pressure type of device but they do not measure the water level in the core. They measure the water level in the downcomer, which is expected to be related to the water level in the core.

The reason that that is done is that trying to measure the water level in the core and during normal operation of a BWR with all the frothing and so on is a very, very noisy signal, and so the normal control -- and the early protective actions are based on the water level in the downcomer.

Now, there is a measure of the water level in the core which does come into play in certain safety circuits, but these are actuation signals that would be expected to come into play only with the reactor shut down. And that was at the time that Oyster Creek was licensed and built, that particular signal was felt to be of such little use that in fact it was not read out in the control room.

Q Was it also a differential pressure device?

A (WITNESS KEATEN) Yes.

Q If you could conceivably measure the level in the downcomer -- you could do that in this vessel, is that not true, although it would not give you indications at the top of the vessel? It would give you some indirect measurement. I am wondering how, in your opinion, how accurate would that be during accident scenarios? Would that give you a fairly decent representation of the level in the core?

A (WITNESS JONES:) Well, conceivably I guess you could say you could measure it. It certainly would be difficult from trying to get the instrument in. But if you had a stagnant loop situation, that is, the pump is off and you measure the water level in the downcomer, it would be in reasonable agreement with the cold water level probably from a solid liquid standpoint now. Probably it would be high by about a foot to two feet. You could probably measure downcomer water level and get an inference.

Now, I am saying that is if you could somehow figure out how to get the instrument in, but you could do it. You know, you could get useful -- I will not say useful because I do not think it is useful information, but you could get information relative to the water level in the core.

Q Based on the problems with frothing in the core, wouldn't it conceivably be just as useful as a direct core level measurement?

A (WITNESS JONES:) Yes. Yes, it would be. If you took a DP across the downcomer or took a DP across the core, both of them would both suffer from the same problems that I testified to earlier.

Q Mr. Ross, in your portion of the testimony on page 15, you state that 16 hours are spent on heat transfer and fluid dynamics.

DR. JORDAN: Just a minute. Let me find it.

BY MR. DORNSIFF: (Resuming)

Q Does that include any theory concerning the effects of noncondensable gases in the system, how they may alter some of the instruments or they may affect some of the instruments?

A (WITNESS ROSS) That particular portion did not. It was theory of heat transfer. We had covered that under different lecture series. Not included was the Unit 2 accident analysis.

Q You have covered the effects of noncondensable gases on instruments.

A (WITNESS ROSS) That is correct.

Q The possible effects it might have.

A (WITNESS ROSS) That is correct.

Q Figure 1 of your testimony, just kind of looking at it in detail, I am kind of curious. First of all, did those curves, the 1400 curve and the 1600 curve, if you are right on that curve, does that mean the cladding is at 1400 or 1600 degrees?

A (WITNESS JONES:) Well, first off, the top curve is a bad reproduction of the number. That is 1800, not 1600.

Q That answers my question, then, because there was a 200 degree difference between that and a 300 degree difference between that and an in-core thermocouple reading.

A (WITNESS JONES:) That is 1400 and 1800 degree curve.

CHAIRMAN SMITH: Go ahead, Mr. Baxter.

MR. BAXTER: You are amending the testimony or correcting it, then, to make 1600 1800?

WITNESS JONES: I can only see the 1800. I am so used to seeing this copy of the figure that I read it as 1800. I am sorry. It is an old habit, I guess, looking at that number, because it always seems to get wiped out every time I make a copy of it, and I instinctively read it as 1800.

CHAIRMAN SMITH: It is correct as presented.

WITNESS JONES: As 300. I think the figure it was made from said 1800. It just did not reproduce well.

MR. BAXTER: The 1600 that is below curve two, it says less 1600 degrees F. It should read less than 1800 degrees F.

CHAIRMAN SMITH: That is what we understood. It seems to be quite clearly a "6."

MR. BAXTER: It does to me, too, but the history I do not think is all that relevant as long as we get it right.

CHAIRMAN SMITH: " 1 are amending the -- whatever, you are amending figure 1 to show 1600 degrees Fahrenheit.

WITNESS JONES: Yes.

DR. JORDAN: He is making it clear.

MR. CUTCHIN: Mr. Chairman, perhaps if the reporter, not having bound that yet, could make that correction in the copy before it gets bound in, it might be more useful than being hidden away in the transcript.

DR. JORDAN: He anticipated you.

BY MR. DORNSIFE: (Resuming)

Q In NRC's supplemental testimony from Mr. Phillips, the one filed supplemental to testimony filed December 1, on page 3 of that testimony, the last paragraph, the second sentence, it says, "The level indication would also provide evidence that the core is covered during recovery from a TMI-2 type flow block condition, even though superheat may persist at the core exit thermocouples."

Could your present instrumentation do that, perform that same function? Is that something that only the level instrumentation could uniquely perform as stated there?

A (WITNESS JONES:) No, I think the level -- I think the core exit thermocouples will provide you information that would tell you that the core was indeed covered because even during the TMI acci nt, after they recovered the core you had many of the thermocouples along the outside of the core indicating very cold temperatures, and had this zone of superheat conditions reading out for the hot portions of the blocked portions or apparently blocked portions of the core.

So, in fact I think the instrumentation available tells you where you are relative to the core. But additionally I would like to point out that even if this happens, even if you had a level instrument that told you that the core was recovered, your actions still do not change. Your attempt is to continue to fill that primary system and return it to a totally subcooled state; and that although if you had a level instrument that told you this, that you had blockage in, say, the center of the core, it does not help you any or you do not change things because you still want to fill the system back up and return it to its normal configuration.

Q On the next page of the testimony, page 4, the paragraph about the St. Lucy event, and from reading the scenario that is attached to the answers to discovery, in your opinion -- are you familiar with that transient?

A (WITNESS JONES:) In a general sense, yes.

Q Is it your opinion that the primary reason that that void occurred in the reactor vessel head was due to the fact that the plant was depressurized -- went through a rapid depressurization to try to get it down to a cold condition? Was that one of the main reasons?

A (WITNESS JONES:) That is my understanding, that basically the operating staff at the plant tried to rapidly depressurize the plant and the metal heated up or kept the hot water in a more or less stagnant region of the upper head, warm, and they did not give it enough time to lose some of its heat through the metal and out to the containment itself; that it ultimately flashed when the pressure got down low enough.

Q And the way they reduced the pressure was by using a system that is unique -- well, unique to -- not necessarily to that system but a CE system where you can use auxiliary spray from a charging pump to reduce the pressure in the pressurizer. Is that how they reduced pressure?

My question -- well, assuming that was the way they did it, does that capability exist on TMI-1 and could that transient have occurred on TMI-1?

A (WITNESS JONES:) I do not believe that capability exists at TMI-1, and I think you are correct that they -- I know at least during certain portions of the transient they were using spray into the pressurizer. I was just trying to check whether it was all the time. Since you are stipulating it for the question, I will say that we cannot get in under that assumption.

Q My question was: without the capability of spraying, other than using the differential pressure across the reactor coolant pump, could you have gotten into that situation on TMI-1?

A (WITNESS JONES:) Well, not directly, but if you initiated -- no, you could not get there unless you were using the DORV to depressurize the system because they did not have any reactor coolant pumps. You could not have gotten this transient.

CHAIRMAN SMITH: Mr. Dornsife, when there is a convenient place in your cross examination, we can take our afternoon break.

MR. DORNSIFE: I am very near the end, so could we continue for a couple more minutes?

DR. JORDAN: I guess before we do that, I am failing to understand something. The B&W reactor can reduce the pressure by its spraying into the pressurizer.

WITNESS JONES: Yes, it can. But the specifics of the St. Lucy event, the one where they have tripped all the reactor coolant pumps, which is your source of your spray injection to the pressurizer --

DR. JORDAN: In the case of St. Lucy it was not; in the case of B&W it is? Is that what you are saying?

WITNESS JONES: I think they probably have both. I am not sure about all of the reactor coolant pumps. But generally speaking that is the normal situation, but they have the additional capability of spraying using the charging pump through the pressurizer spray line or an auxiliary spray line of some sort.

DR. JORDAN: I see. I guess I got the impression that the boiling occurred as a consequence of the cooling. Yes, and then of course the consequence was a reduction of pressure that went along with the natural convection cooling. And they arrived at a condition as a consequence of cooling and reducing the pressure whereby they got --

WITNESS JONES: Yes, generally what it was was by depressuring the plant in a rapid fashion, the hot pocket of fluid located in the upper head just turned to steam when it dropped below the saturation pressure.

BY MR. DORNSIFE: (Resuming)

Q Again on page 4 of that testimony at the bottom where it talks about vessel level information possibly necessary or essential for using the reactor vessel vent valves, you had said that B&W was currently in the process of developing guidelines for using that valve. I am just curious of what kind of guidelines you are planning on giving for using that valve without relying on vessel level.

WITNESS JONES: Well, to give you an example of how we will use the head vent, let's say we have something like a St. Lucy event, for example. Now, it is quite clear from looking at the fact that the pressurizer level goes off scale or has rapid changes while they are switching the location of the charging pump that there is a void in the head and they can open the vessel head, use the high pressure injection system to maintain system pressure and presurizer level. While the head is continuing, the gases or steam in the head is being discharged through the vent.

When you would fill the level or fill up the reactor vessel, you will get a change in the flow quality from steam to water, and you will get an increase in the pressurizer level according to that because the volumetric flow change from steam to water is quite dramatic. You might only be putting out, say, four or five pounds of steam but in fact be putting out an equivalent volume associated with the replacement liquid of, say, 100 gallons per minute. These numbers are just not even real numbers but just to try to point out an example.

Now, when you change to a water flow from a steam flow, say four pounds per second, and it is now up to, say, ten pounds per second or whatever, you will get an increase in the flow rate. But that may be associated with 50 gallons per minute. So now suddenly you are putting in 100 gallons per minute, you are only discharging 50, and so the pressurizer level will increase at the rate of 50 gallons per minute.

So when it makes the transition from steam to water, the pressurizer level will start to increase. At that point in time you can close the vent and continue to cool down, and if a bubble reappears, you just take the action again.

Q What if there are voids in the other parts of the system?

WITNESS JONES: I was just trying to give an example of it. There are two ways -- we are going to be installing on the B&W plants -- GPU is installing both the vents on top of the hot legs and a vent on top of the reactor vessel head. AT least I believe they are installing them; let me put it that way.

You can use the hot leg vents to vent the rest of the system, and the general way of using the vents is to vent the hot legs first in order to aid in the refilling process of the primary system, and you can tell when you had filled up the hot leg by the hot leg RTD measurement. That is, you will return to a subcooled situation and get indications of natural circulation flow if you fill it up, if you fill up the system.

At that point in time you then go and use the heat vent in a similar fashion to what I just described.

Q And on the last page of that testimony, page 5, there is a sentence, the last sentence of the first paragraph at the top, "The staff is aware of at least one vent design for which the designer insists on the coupling of level information to the vent."

Are you aware of that, details of that, why they feel that is a better way of doing it than, let's say, your proposed method?

A (WITNESS JONES:) While I do not know the details of why they chose it, again, when we were developing the guidelines we were looking at, you know, what the existing instrumentation was, and then, you know, what the objective was. If we felt we needed level information to really use the vents properly, we were to identify it, and we found we just did not need it.

I cannot find the information that I was given. I believe the other vendor is Westinghouse, but I do not think the staff provided us with details as to the why.

MR. DORNSIFE: I would like to explore that with the staff.

WITNESS JONES: They give us some supporting information to one of our interrogatories concerning use of the head vent for another vendor, and it said -- all I have here, and I quote, "The designer is Westinghouse. They have stated that vent operating procedures will permit the vent to be open only when a steam level in the head is indicated by the reactor water level instrument, and the vent is to be closed when the steam bubble has been vented."

That is the only details I have.

BY MR. DORNSIFE: (Resuming)

Q One of the recommendations of the St. Lucy accident concerned specifically the B&W design, and it read that consideration should be given to the potential for the formation or accumulation of vapor in the candy cane of B&W reactors, particularly in the inactive loop, when natural circulation cooldown was being accomplished with a single steam generator.

My question is: even if you had void formation in the candy cane, could you still not remove enough decay heat using two-phase natural circulation?

A (WITNESS JONES:) Well, that would depend on the size of the bubble if there was one in the candy cane. If the bubble was large enough it could block the circulation path and yet small enough not to have a condensing surface exposed in the generator, you could have an interruption of circulation.

Now, you should not be able to form that bubble if you went into a situation like St. Lucy where you had a bubble in the system, depressurized and the bubble tried to transport itself into the hot legs. Just the fact that you are trying to maintain 50 degrees subcooled in the loop during this process, the hot steam or gases escaping into the hot leg will mix, will show up on the hot leg RTD as a loss of the 50 degree subcooling, and you could just reinitiate HPI, and really what you would do is hold the pressure for a while.

Additionally there will be ultimately vents at the top of the candy canes so that if you got a local bubble trapped up here, it could be vented off. In fact, this is an advantage for the B&W system as opposed to the Westinghouse system or the Combustion Engineering, and they have 15,000 U tubes that are impossible to vent should a big gas bubble accumulate up there, while we can vent the gases off through the hot leg vent.

Q I guess, going back to the first part of your answer, you said that if the steam bubble was not big enough, that you could not get a condensing surface. I was under the impression that as long as you had exposed surface in the steam generator tube of cold water, with a steam bubble in there you could have nothing but vapor in the loop and still have two-phase natural circulation.

A (WITNESS JONES:) Yes, yes, that is correct. But what I am saying is if you go and look at the 180 degree bend at the top of the hot leg, if a bubble forms right across that bend, right at the bottom part of the elbow, you form a bubble there and there would be no way to transport liquid from one side to the other. Okay.

Now, until you drain down -- I don't know the exact distance, but probably about 10 feet -- you have not hit the top -- the cold surface of the steam generator tubes yet. So a bubble of that order of magnitude is the size bubble that could block or interrupt a circulation pattern for a period of time.

This was discussed in the small break LOCA testimony where you could have this interruption after forminng the small bubble, waiting for it to grow forming the condensing surface, yet during that entire period of time the core is being adequately cooled because it has water over it.

Q I just have one more question. You had mentioned briefly the LOFT experiment. You mentioned some familiarity with some of the details, and one of the things that interested me was you indicated that they have different types of level instrumentation on that particular level. You mentioned they ran some experiments using the heated thermocouples.

Are there other types of level devices, and what has their experience generally been as far as reliability, and whether they work or --

A (WITNESS KEATEN) It is my understanding that they have normally used a DP type of level indicator as well as the heated thermocouples, but I am not really familiar in detail with that, Mr. Dornsife. I have just seen this recent report that came out and it crossed my eye that they were describing the difficulty or appeared to be describing the difficulty that they had had with the heated thermocouple until they had turned off the pump.

A (WITNESS JONES:) They do have a DP cell. I think the heated junction thermocouples were a special test they were running. The normal type of level instrument that they use, I believe, is a conductivity probe. That is the normal type of level instrumentation they use. I would not call it instrumentation unambiguous by no means in that I have seen times where they indicate, say, five feet of liquid or two-phase mixture, a gap of no reading indicative of steam, and then a spurious couple of marks here and there indicating maybe two-phase up in that region, which Iusually consider to be spurious.

Whenever they have tried, or in general most of the times I have seen them give you inventories in the system, they have always done them on a mass flow balance rather than by use of any of their level instruments. It has usually been mass -- an integrated leak flowout, that sort of stuff.

Q So your testimony is basically even on the LOFT test there is no unambiguous way of determining the level of the vessel.

A (WITNESS JONES:) In general if you could plot and get used to the conductivity probes and their response, you can learn to ignore them or throw over the fliers or recognize if you have five feet of liquid, a two-phase mixture and then a couple of points where it is steam and then another ten feet of water, it means there is probably water or two-phase in between there.

When you get used to that you can plot some trend information which may be useful for them out in Idaho, but to give that to an operator unprepared for it, I would be very hesitant.

Q I guess that was kind of my next question, the last question. Do you know whether the operators use that information for their procedures at LOFT?

A (WITNESS JONES:) No, I do not. And they have -- all I have heard are some second-hand remarks, which is they have been -- the capability of putting up their predictions on a screen and comparing it to some of the actual plant data, and so they could tell where it is going. Now, this is, as I said, second hand, and I would not put much -- you know, that is to the best of my knowledge.

Additionally, they have some override capability that if they do not like the way the transient looks, they are capable of pressing a button and bringing on additional safety systems if the thing looks to squirrely for them.

MR. DORNSIFE: I have no further questions. Thank you.

CHAIRMAN SMITH: Let's take a 15-minute break, please.

(Recess.)

CHAIRMAN SMITH: I think we are ready for the redirect.

REDIRECT EXAMINATION

BY MR. BAXTER:

Q Do you have a correction, Mr. Keaten, that you would like to make to your testimony?

A (WITNESS KEATEN) Yes, I do. During the break, Mr. Phillips pointed out to us my reading of the document on the most recent LOFT test had misled me. He is much more familiar with the details of LOFT than I am. I had read what was in this document as referring to a heated thermocouple level probe. And he points up that, strictly speaking, those thermocouples they have in there are not heated thermocouples in the normal sense we use the term "heated thermocouples." I guess they receive some heat from the energy stored in the vessel rather than being heated on electric current.

And so the results here may not be particularly representative of what a more carefully designed heated thermocouple might do. I indicated earlier I was drawing conclusions based on one sentence, and apparently I was a little bit misled by what that sentence said.

MR. BAXTER: I have no further redirect examination.

CHAIRMAN SMITH: Any further cross examination?

MR. CUTCHIN: None by staff, sir.

BOARD EXAMINATION

BY DR. JORDAN:

Q I am going to refer you to staff testimony by Mr. Rubin.

MR. CUTCHIN: Mr. Chairman, if it might help the Board, the staff has now decided that we will not be offering Mr. Rubin's testimony. You may still have the same questions, but we will not be offering that testimony.

DR. JORDAN: Yes. This does --

MR. CUTCHIN: I might tell you why.

DR. JORDAN: Yes. I guess I better find out why. Is it because Mr. Rubin is no longer of the opinion?

MR. CUTCHIN: There are several reasons. If you look at the purpose for which that testimony was originally filed, it was to describe the status of the inadequate core cooling procedure review at the time the testimony was filed, which was December 1.

If you will turn to the last page, which is page 3, the question is: "What action does the staff plan to resolve the remaining concerns?"

And the staff is stating that it will require prior to startup that there be inadequate core cooling emergency procedures which are adequate to allow operation, and the staff will ensure that this comes about either through revision of existing procedures or putting into place of the ATOG, meaning "abnormal transient operational guideline," procedures.

The status between staff and Licensee is that these procedures are still not resolved, and we are working on them. There is in process a letter from staff to Licensee setting forth the results of the staff latest review and some interchanges between Licensee and staff. And, of course, the Board and all the parties will be served with that, hopefully, within the next week or so.

So there was really nothing other than to update that status, and it does not really help the record as far as substance is concerned.

BY DR. JORDAN:

Q I guess then the only question I have either of you or of these witnesses is: Has the -- will the requirements of NUREG-0758 -- and I believe it is section -- what is it -- 2.13? Is that it? Is that the section we are talking about with respect to the Lessons Learned, which has two parts to it?

One of the parts is the short-term requirement that there be a meter for measuring inadequate core cooling or for measuring approach to saturation. And I guess the question is whether that first section either has been or will be complied with completely prior to restart.

(WITNESS JONES) Dr. Jordan, you stated 0758. Do you mean 0578?

Q 0578. I mispronounced it -- I misspoke.

A (WITNESS JONES) I believe you are talking about 2.13 part B.

Q Part B. And then there is a B1 and B2.

A (WITNESS JONES) This one is not broken up. The copy I have is not broken up into B1 and B2. Though it does call for two distinct action levels.

Q All right. Now, the first part is something, I believe, an action required prior to restart. Well, let me find it.

(Pause.)

MR. CUTCHIN: Dr. Jordan, that matter is addressed in the staff's safety evaluation report, starting at pages C8-14. And it is my understanding that the procedures that I was just referring to as having been the subject of Mr. Rubin's testimony are the very procedures that are based upon existing instrumentation.

DR. JORDAN: Right.

MR. CUTCHIN: And the saturation meter. And those are still under development, and we will require that they be in adequate shape prior to our saying it is okay to restart.

DR. JORDAN: I guess also the staff is requiring that there be a level meter, too. And I suspected you were going to have the same success with respect to the short term as you were with the long term.

MR. CUTCHIN: I do not believe that staff is requiring a level meter short term.

DR. JORDAN: No, but long term they are. And it sounds to me as though you are going to have a very difficult time getting the Licensee to agree that there is such a meter. Are you also going to have the same problem in getting the Licensee to agree with the short term? You say you are going to require it -- what if they do not? Are they going to do it? I would like to hear them say that they are.

MR. CUTCHIN: I thought you were saying either they or us. I was trying to clarify the situation. I was not trying to cut off your question.

BY DR. JORDAN:

Q Do you see my problem?

A (WITNESS KEATEN) As far as the short-term hardware aspects are concerned, we have already committed to do those things that are called for, the saturation meter, the connecting of the in-core thermocouples, the expanded range hot leg temperature indication. We have committed to those. We presently have that work under way. And it will be installed prior to restart.

Q That was my impression, precisely, but I was -- having read Mr. Rubin's testimony, I have some questions.

A (WITNESS KEATEN) Where Mr. Rubin's question is concerned, as Mr. Cutchin has indicated, this has not been a disagreement on hardware but a disagreement on the wording of some of the procedures.

Q I see.

A (WITNESS KEATEN) We have received Mr. Rubin's comments. Some of those comments have been incorporated. For example, the inadequate core cooling procedure 1202-39 has very recently been revised and, I believe, does incorporate some of Mr. Rubin's comments.

We intend to continue to interact with the NRC staff until we have a procedure that we all believe are acceptable.

DR. JORDAN: Very well, we shall leave it at that. If there is any problem, we would want to hear about it.

(Pause.)

BY DR. JORDAN:

Q The testimony of Mr. Pollard in this Contention UCS 7 has been withdrawn. However, he does raise one or two questions which I would like to get your opinions on.

On page -- do you have his testimony -- page 7-11. Item 3 on that page, the last sentence reads:

"Furthermore, the measurement of temperature in the hot legs is of no use during the bleed-and-feed mode of core cooling, because it is not in the cooling water flow path."

Is this true?

A (WITNESS JONES) That is true. That the hot leg temperature indicators, if you are in a feed-and-bleed mode of cooling without the steam generators, will not be necessarily in the direct path of the fluid flow. What would be used in this mode of cooling would be the core exit thermocouples for providing HPI throttling guidance from the standpoint of prevention of thermal shock, brittle fracture concern of the reactor vessel, which has previously been discussed.

When or if the steam generator is restored as a heat sink, then the hot leg thermocouples would become the primary means. Again, the objective is still always to try to return the primary system to a liquid solid configuration. Now, if you are in the bleed-and-feed mode, it may not be possible to do that, depending on where the break is in the system. But you can use the in-core thermocouples to provide guidance on throttling; that is, if they get adequately subcooled, so subcooled that you have to worry about potential for brittle fracture of the vessel, then you would throttle the HPI flow so as to depressurize and prevent that concern, prevent a potential brittle fracture of the vessel.

But if you restore the heat sink you would still monitor the in-core thermocouples from that standpoint for brittle fracture measurement or as to provide indications to help you prevent it. But you continue on HPI until you get that system returned to a subcooled state.

Q Well, the t-sat meter would not be useful then during this period of time; is that right?

A (WITNESS JONES) That is correct. Well, let me -- I am having trouble with the word "useful," whatever. If it is reading saturated or superheat, even though it is not in the flow path, it is an indication that you have an off-normal situation in the primary system, and the instruction based on the instrument is to continue to provide HPI flow. And then if the in-cores start to indicate excessive subcooling to the point where brittle fracture of the vessel starts to become a concern, then you can throttle your HPI while maintaining subcooled conditions within the reactor vessel itself.

Q Well, the question is whether or not -- would not, under those circumstances of bleed-and-feed, would a level meter be of value to the operator during that operation?

A (WITNESS JONES) I would say "No," from the standpoint that if he had a level meter and he is in this mode of operation, again the only time you would want to throttle HPI is to prevent the brittle fracture concern for the reactor vessel, which implies a subcooling condition within the vessel.

And a level meter would not be able to tell you how much that vessel is subcooled. The thermocouples are much better equipped for that, and you want to maintain the subcooling in the vessel such that you do not even get near boiling, you know, a minimum of 50 degrees subcooled but less than whatever the value is on the specific limit curve for brittle fracture of the vessel between those two limits.

And under those conditions, the level instrument should be reading a flat value. It would not be able to tell you the difference.

Q I see. Okay. Well, on page 12, the next page, he refers to the item that I now remember where I saw; namely, that the meter can display either temperature or pressure margin, that you say in your case you display temperature margin.

A (WITNESS KEATEN) As far as the meter being installed prior to restart, which is the continuous indication to the operator, that is correct.

On the other hand, the readout via the computer will be able to read in either temperature or pressure.

BY CHAIRMAN SMITH:

Q Does that mean that the computer will tell you how close to saturation you are in terms of pressure, too?

A (WITNESS KEATEN) Yes, sir. It will do it with either temperature or pressure.

BY DR. JORDAN:

Q He says that if the operator makes the simple mistake of thinking that the temperature margin is displayed when in fact the pressure margin is displayed, he can well fail to maintain an adequate margin to boiling. Is that true?

A (WITNESS KEATEN) I am not exactly sure of the condition that Mr. Pollard was referring to there. But in our case, since he does not have that option with the hard-wired instrument, it always displays the temperature margin. So there is no danger of confusion there.

As far as the output from the computer is concerned, our computer output, as you are well aware, is labeled as to what it is. And so it would say "temperature margin" or "pressure margin." So I really do not think that that mistake is likely to occur.

Q The next item had to do with the nature of whether safety grade or not, but I think we have already covered that item today.

Those are the only questions I have concerning Mr. Pollard's testimony.

(Pause.)

Dr. Little points up on page 7-14, the middle of the page, Mr. Pollard says as follows: "Therefore, I conclude that instrumentation provided to directly measure reactor water level should be used to automatically initiate any necessary protective actions." I believe I also have that one sentence underlined in my testimony, too -- in my reading of Mr. Pollard's testimony also.

Would you care to address that statement?

A (WITNESS JONES) Yes. I will take a crack on it.

First off, the normal protective systems or emergency systems utilized for a normal small-break scenario, the emergency core cooling systems and the reactor protection systems for just tripping the reactor out, now, there is no way that I know of that you could get to a low-water condition within the reactor core without having already reached one of the various reactor protection system inputs; that is, either low pressure or high temperature or low flow.

There is no way that I know of that you could get to low water level in the core without having at least reached the reactor trip signal.

Q Which would start the high-pressure injection system?

A (WITNESS JONES) No, that would just trip the reactor.

Q I see. All right.

A (WITNESS JONES) From the high-pressure injection system standpoint, under the typical licensing calculation assumptions made with the emergency feedwater system operable, you will get actuation of the emergency core cooling system prior to even reaching saturated conditions in the primary system loops.

That is, you would actuate your HPI prior to losing all subcooling in the primary -- in the hot legs or in the reactor coolant system.

Q I see. You expect the ECCS to be triggered prior to the 50-degree margin?

A (WITNESS JONES) I am not so sure about the 50 degrees, because that is a combination including all the various instrument errors. From a real-world situation, what you would expect the actual fluid conditions to be, you would have actuated the ECCS prior to actually losing subcooling. So that you should have all of your emergency systems on long before this.

If you do not have the steam generator, again the key indicator that we would utilize is saturation for the operator to manually initiate HPI for certain small-break scenarios, where you have a very small break with normal steam generator cooling. The operator is instructed to actuate the HPI based on saturated conditions.

that also occurs long before you get to the core water level uncovery. So that from the normal safety systems that are designed to protect the plant from ever uncovering the core, those functions are already taken and will be done long before the water level is reached.

Now, as far as automatic reactuating, protective actions based on a water level signal for an inadequate core cooling scenario, unless that water level instrument could be designed to be highly reliable, to be very accurate, I certainly would not advocate use of it to take some of the additional operator actions we are contemplating, because they can have unsafe ramifications if they trigger and come on earlier than they are supposed to, such as the reactor coolant -- start of the reactor coolant pumps in the middle of a small-break LOCA does have a negative benefit, and if it is not necessary to take that step, you should not.

So I disagree with his statement.

(Pause)

Q On page 14 of your ^ ~timony, the short paragraph near the top of the page beginning, "We disagree," the second sentence says, quote, "We also disagree with UTS and ECNT that vessel water level indication should be installed at TMI 1."

Was that ECNT that you meant there, or was it NGREE?

MR. BAXTER: You are right.

CHAIRMAN SMITH: I think that is a better question to Mr. Baxter, perh_ps.

MR. BA.TER: Mr. Keaten and Mr. Jones, do you have a correction to make on page 14, the penultimate part of your testimony there where it says ECNT. Should that read NGREE?

WITNESS KEATEN: I believe it should be NGREE rather than ECNT. I would like to see the correction made.

DR. JORDAN: All right. I do not believe I have further questions. I have asked quite a good number of them, and most of them it seems to me have been covered. So that is all I have.

CHAIRMAN SMITH: All right. Is there anything further of the panel?

MR. CUTCHIN: No.

CHAIRMAN SMITH: Mr. Dornsife?

MR. DORNSIFE: Yes, I have two short questions concerning Dr. Jordan's questions.

CROSS EXAMINATION ON BOARD

BY MR. DORNSIFE:

Q Mr. Pollard's concern about the thermocouples not being in the circuit for feed and bleed, in the case of feed and bleed aren't there, temperature instrumentations available on the pressurizer that could indicate the same as the hot leg RTDs and perform the same purpose.

A (WITNESS ROSS) Assuming the flow path is followed all the way up through there, you do have two temperature indicators located in the pressurizer itself, and they would indicate the water temperature coming into the pressurizer.

Q Do you know physically where the hot leg RTDs are located? Are they right on the vertical part of the hot leg, or are they on the candy cane, or where physically are they?

A (WITNESS JONES) They are located on the riser section, the long vertical section on the hot leg, approximately five to ten feet from the bottom of a 180 degree bend, right where it makes the turnover towards the same generator.

It is somewhere in the five -- it is closer to the five-foot - I believe away from that.

Q Is it conceivable under certain circumstances that the one t-stat meter could be reading different than the other one?

A (WITNESS JONES) Yes.

Q What would be the operator guidance if that were the case?

A (WITNESS ROSS) I think the operator guidance would be consistent with what we are telling him in all cases that we give him guidance on, verify your parameters by redundant indications; make decisions based on redundant indicators and indicators that you feel are functional, where you end up in a tie, where you believe your worst indication, and take action accordingly.

Q So you are saying that -- let's say, an example that a void started forming in the inactive loop, and you had natural circulation in the other, and the t-stat meter in the inactive loop approached such a region, and the other indicated subcoolingd, that the operators would then only look at the one in the inactive loop and not terminate high pressure injection based on the reading in the inactive loop.

Is that what you are saying?

A (WITNESS ROSS) See in this case to look at other indicators, in-core thermocouples -- he would select and isolate temperatures. He would then look at the other leg. At that point if he had TH subcooled on one look, if he had thermocouples subcooled on one loop, he had all indications that the core was in good shape, he would then have met his prottling criteria.

He would say it was an out-of-service instrument. He would have made the decision based on redundant indicators and functional indicators and said it is out of service. I have met my prottling criteria on the instruments that are operable, those being instruments that redundantly indicate the core is in a safe condition.

Q Do you feel this is clearly indicated in the procedures?

A (WITNESS ROSS) To put every conceivable combination in the procedure, of course you cannot do that. I feel it is clearly indicated in his operator training that he is to use redundant indicators, and I think it has been clearly stressed in the OARP training we have done.

OARP is an abbreviation for operator accelerated retraining program.

MR. DORNSIFE: I have nothing further.

CHAIRMAN SMITH: All right. Gentlemen, you may step down. Thank you.

(The witneses were excused.)

CHAIRMAN SMITH: We are changing witnesses, and Dr. Little has pointed out an area of completion about the scheduling. You had indicated, Mr. Baxter, in your letter of October 30 that you would present the testimony of Mr.

Groton in response to Board question number 11 at about this time, I guess.

Now, you have it following this.

MR. BAXTER: No. We were going to try and do it. Mr. Groton would have been available today, and he will be available Friday. He will not be available tomorrow, unfortunately. I have discussed it with Mr. Ketchen, and we think it would be advantageous to have his witness and mine appear roughly the same time.

We do not think that the hearing on this issue is going to go past maybe mid-morning tomorrow, and I did not think you would want to wait until Friday to hear Mr. Groton.

So what we thought we would do, unless you do want to stay over to Friday for that would be to schedule Mr. Groton and Mr. Jensen on Board question 11 as the very first item on Tuesday the 27th. We do not think it will take a long time, and then go into Class 9 accidents.

It would still follow this issue in the record, but we are going to have some days off again.

CHAIRMAN SMITH: All right.

(Board conferring.)

MR. BAXTER: I did want to advise the Board that I served Mr. Groton's testimony on Board question number 11 by mail on October 30, so you might check to see if you have it, and if you do not I will give you a copy before you leave.

CHAIRMAN SMITH: That is not the problem with that much lag. It is when we receive it -- it is when we are here and it is delivered up there. A week does not make any difference.

All right. With that arrangement then, the Board would prefer the arrangement of beginning the first thing on the 27th instead of staying over for Friday.

MR. CUTCHIN: I have no preference. I apologize. I was not paying attention at the moment. Could someone repeat the question?

DR. LITTLE: We just arrived at a decision, if it is okay with you, to have the Board question 11 the first thing on January 27.

MR. CUTCHIN: Yes, ma'am. I had understood that to be the case a moment ago. Mr. Baxter and I had agreed that that would be the preferred thing to do unless the Board had a different preference.

CHAIRMAN SMITH: Now, the Board got into it. We prefer that approach.

MR. CUTCHIN: That is fine with the staff.

CHAIRMAN SMITH: Okay. We have Mr. Phillips.

Whereupon,

LAURENCE E. PHILLIPS

was called as a witness, and after having been duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. CUTCHIN:

Q Mr. Phillips, do you have before you two documents, both bearing the caption of this proceeding, the first being entitled, NRC Staff Testimony of Laurence E. Phillips Regarding Reactor Water Level Instrumentation (UCS Contention 7, Sholley Contention 6B, and NGREE cont ntion 5B)?

A Yes, I do.

Q Did you also prepare a copy of your professional qualifications which I have now given copies of to the Board, and which I have attached to that original testimony?

A Yes, I did.

Q Do you have any corrections to make to this first document?

A Yes, I have some corrections.

Q Would you identify them, please?

A On page four in the last two lines of the answer to Q-8, if you would inject the phrase -- now ends in saying, the core is uncovered, exists, and that should read, when the core is uncovered and although provide indication that the core is covered when no superheat exists.

MR. BAXTER: I am sorry, Mr. Cutchin. I need to ask to have that again. I just did not get it.

MR. CUTCHIN: To insert between the words, uncovered, and exists, quote, and also provide indications that the core is uncovered when no superheat, end quote. "Exists" is the word that was still there. Corrections have been made in the reporter's copy.

WITNESS PHILLIPS: On page five, the second paragraph, the second line, water. The w should be lower case where it say, water level instrument.

On the same page, the answer to question Q-10, the fourth line, insert between the words, sensors and cold leg, the word, and. It should read, sensors, comma, and cold leg and hot leg.

On page six the answer to Q-11 at the end of the third line, it reads, reactor coolant system temperature. It should read, insert between system and temperature the words, pressure, comma, reactor vessel outlet, temperature, comma.

That is all the corrections I have.

BY MR. CUTCHIN:

Q Did you also prepare a document bearing the caption of this proceeding entitled, NRC Staff Testimony of Laurence E. Phillips, supplementary testimony to that of Laurence E. Phillips filed December 1, 1980, regarding reactor water level instrumentation (UCS Contention 7, Sholley Contention 6B and NGREE Contention 5B)?

A Yes.

Q Are there any corrections to be made to this document?

A None.

Q As modified, are these documents true and correct to the best of your knowledge and belief?

A Yes.

Q Do you adopt them as your testimony in this proceeding?

A I do.

MR. CUTCHIN: Mr. Chairman, I offered the documents just identified, which include the two pieces of direct testimony plus one copy of Mr. Phillips' professional qualifications be received into evidence and bound into the transcript at this point as it is read.

CHAIRMAN SMITH: If there are no objections, the testimony is received.

(The documents referred to above were received in evidence.)

Corrected copy

Layer 1/21

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
Metropolian Edison Company	)	Docket No. 50-289
(Three Mile Island Nuclear	)	
Station, Unit 1 Restart)	)	

NRC STAFF TESTIMONY OF LAURENCE E. PHILLIPS

REGARDING REACTOR WATER LEVEL INSTRUMENTATION

(UCS Contention 7, Sholly Contention 6(b) and

ANGRY Contention 5(b))

Q.1. Please state your name and position with the NRC.

A. My name is Laurence E. Phillips. My position is Section Leader, of the Thermal-Hydraulics Section, Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulatory.

Q.2. Have you prepared a statement of professional qualifications?

A. Yes, a copy of this statement is attached to this testimony.

Q.3. Please state the nature of the responsibilities that you have with respect to the Three Mile Island Nuclear Station, Unit 1.

A. I am responsible for managing the review of licensee responses to the requirements of TMI Task Action Plan II.F.2 in particular and Inadequate Core Cooling requirements in general.

Q.4. What is the purpose of your testimony?

A. The purpose of this testimony is to respond to three similar contentions; UCS contention 7, Sholly Contention 6(b), and Angry Contention 5(b)

UCS Contention 7 reads as follows:

"NRC regulations require instrumentation to monitor variables as appropriate to ensure adequate safety (GDC 13) and that the instrumentation shall directly measure the desired variable. IEEE 279, 4.8, as incorporated in 10 CFR 50.55a(h), states that:

> To the extent feasible and practical protection system inputs shall be derived from signals which are direct measures of the desired variables.

TMI-1 has no capability to directly measure the water level in the fuel assemblies. The absence of such instrumentation delayed recognition of a low water level condition in the reactor for a long period of time. Nothing proposed by the staff would require a direct measure of water level or provide an equivalent level of protection. The absence of such instrumentation poses a threat to public health and safety."

Sholly Contention 6(b) reads as follows:

"The Commission's Order and Notice of Hearing dated 9 August 1979 are insufficient to provide the requisite reasonable assurance of operacion without endangering public health and safety because these short-term actions do not include the following items:

b. Completion of installation of instrumentation for detection of inadequate core cooling:"

Angry Contention 5(b) reads as follows:

"The NRC Order fails to require as conditions for restart the following modifications in the design of the TMI-1 reactor without which there can be no reasonable assurance that TMI-1 can be operated without endangering the public health and safety:

(b) Installation of instrumentation providing reactor operators direct information as to the level of primary coolant in the reactor core."

Q.5. What may cause inadequate core cooling?

A. An example is a small break LOCA. However, our objective is to detect the symptoms of inadequate core cooling and to initiate and monitor the progress of corrective action independent of the cause of the condition and of other preceding actions which should have precluded the ICC condition.

These contentions allege that until the instrumentation to detect inadequate core cooling includes a means for direct measurement of core water level, it will not be safe to allow TMI-1 to restart.

Q.6. What is the relationship between the reactor water level and the inadequate core cooling?

A. As long as the reactor is shut down and the two phase froth level (the swollen water level due to the presence of steam voids) is above the top of the core, cooling is adequate. When the two-phase froth level begins to drop below the top of the core, the exposed fuel begins to heat up and will ultimately reach temperatures at which fuel damage occurs. This is inadequate core cooling.

Q.7. What conditions must be monitored to provide a direct indication of inadequate core cooling?

A. The most direct indication is an increase in fuel temperature or surface temperature of the fuel cladding. In addition, the two phase froth level without pumped flow or the coolant quality with pumped flow provides indication of conditions associated with inadequate core cooling.

Q.8. Why isn't instrumentation to directly measure these parameters required?

A. There are serious problems concerning the feasibility and reliability of the design of a commercial PWR core to incorporate thermocouples attached directly to the fuel cladding surface. Such a design would greatly complicate the installation and replacement of fuel assemblies. In addition to the coolant level and quality measurements, there are indirect indicators (e.g., core exit thermocouples) of inadequate core cooling and of conditions which could lead to inadequate core cooling (e.g., saturation meters) which can be monitored more reliably and can provide equivalent information to the operator if properly processed for display and incorporation into emergency procedures. Core exit thermocouples provide an indication of the magnitude of steam superheat when the core is uncovered ~~exists.~~ and also provide indication that the core is covered when no superheat exists.

Q.9. Does the absence of direct water level measurement instrumentation conflict with GDC 13 and 10 CFR 50.55a(h) of NRC regulations as inferred by UCS Contention 7?

A. GDC 13 states: "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges."

For TMI-1 and other PWRs, the normal water level range in the reactor coolant system is within the pressurizer and is maintained by the pressurizer level control system. For transient and accident conditions resulting in low coolant inventory, the high pressure and low pressure safety injection systems are initiated on low pressure signal to reflood the primary system. Reactor vessel water level is not an appropriate input to the safety injection system since the corrective action is initiated by a low pressure signal well in advance of core uncovery.

Regulation 10 CFR 55 a(h) applies only to protection systems. Reactor water level instrumentation, if installed, will be used for monitoring and operator actions only and will not provide input to protection systems. Additionally, regulation 10 CFR 55a(h) is applicable only to construction permits issued after January 1, 1971. The TMI-1 construction permit was issued in 1968.

Q.10. What instrumentation is currently availabe at TMI-1 to indicate inadequate core cooling?

A. The instrumentation available at TMI-1 that could indicate inadequate core cooling includes core exit thermocouples which indicate coolant superheat associated with excessive fuel cladding temperature, reactor coolant pressure sensors, and cold leg and hot leg resistance temperature detectors (RTDs) which provide inputs to compute the margin to coolant saturation conditions, subcooling meters which will display the margin to saturation, and reactor coolant pump current which provides indication of increasing coolant quality while the pumps are running.

Q.11. Have emergency procedures for response to inadequate core cooling in TMI-1 using existing instrumentation systems been approved by the staff?

A. The licensee has proposed emergency procedures for inadequate core cooling (EP 1202-39 and 1202-6B), which rely on the information available from the core exit thermocouples, reactor coolant system pressure, reactor vessel outlet temperature, and the new saturation (subcooling) meters to identify the approach and existence of inadequate core cooling and to specify the operator actions required to prevent or recover from inadequate core cooling. These procedures are under review by the staff and revised submittals have been requested from the licensee. Although we have not completed our review at this time, we are confident that procedures acceptable for TMI-1 restart without reliance on water level measurement can be developed. The staff has found ICC emergency procedures based on instrumentation similar to that which will be provided prior to TMI-1 restart to be acceptable for other PWRs while a level measurement system to further enhance the operational safety is being developed. We will keep the Board apprised of our progress in the development of acceptable procedures for TMI-1 restart.

Q.12. How are the existing instrumentation systems at TMI-1 being upgraded to accomplish the inadequate core cooling response objectives?

A. Prior to restart of TMI-1, we will require an upgrading of those existing instrumentation systems, information displays, and operating procedures which relate to the detection of and response to inadequate core cooling conditions. These modifications in conjunction with improved operator training will substantially enhance the capability of the operator to recognize and respond to conditions of inadequate core cooling.

The licensee has committed to install a primary coolant saturation meter, and has also described two short-term modifications to existing instruments to satisfy the requirements of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, July 1979).

These changes involve routing the in-core thermocouple signals outside of containment and connecting the 52 in-core thermocouples to the plant computer (for display purposes), and providing an extended range for reactor outlet (T_H) temperature measurement (from 520° -620°F to 120° -920°F). This modification will be made to four T_H channels, two in each reactor coolant loop. In addition, it is intended to isolate the new wide range T_H signal from the existing control signals. These signals will then be seismic Category I and separated for use as redundant signals. It is anticipated that all modifications required for existing instrumentation will be implemented prior to TMI-1 restart.

Q.13. What are the design criteria for instrumentation for detection of inadequate core cooling? Will the TMI-1 instrumentation meet these criteria prior to restart?

A. All instrumentation in the final inadequate core cooling monitoring system must be evaluated for conformance to TMI Task Action Plan II.F.2 as clarified in NUREG-0737, November 1980, including Attachment 1 (In-core Thermocouple criteria) and Appendix B, "Design and Qualification Criteria for Accident

Monitoring Instrumentation," of the cited document. Such an evaluation will be required prior to restart and deviations must be justified.

Q.14. What additional instrumentation, if any, will be required for detection of inadequate core cooling for TMI-1. Will any additional instrumentation be installed prior to restart?

A. The staff requirements for additional instrumentation to detect inadequate core cooling are presented in NUREG-0578 and supporting documents (D. Eisenhut (NRC) letter to All Operating Nuclear Power Plants On "Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident," dated September 13, 1979; H. Denton (NRC) letter to All Operating Nuclear Power Plants on "Discussion of Lessons Learned Short-Term Requirements," dated October 30, 1979) and are further clarified in NUREG-0737. These documents state that the licensee shall provide a description of any additional instrumentation or controls proposed for the plant to supplement those devices cited in the preceding section of the requirement (2.1.3.b) which give an unambiguous, easy-to-interpret indication of inadequate core cooling.

The clarification provided with the Denton letter to operating nuclear power plants specifically states that "The evaluation is to include reactor water level indication." The purpose of the evaluation is to determine the feasibility and practicality of direct water level measurement in the core and to allow the licensee to select the best of several instrumentation concepts and/or data processing schemes, for obtaining a clear indication of inadequate core cooling conditions.

Metropolitan Edison and Babcock & Wilcox have proposed the use of existing instruments (that is, no additional instrumentation) to provide an indication of inadequate core cooling. The basis for this proposal is that there is no need for additional instrumentation, particularly a reactor vessel water level measurement, to determine inadequate core cooling. We have reviewed the justification for no additional instrumentation and have found it unacceptable (D. Eisenhut (NRC) letter to R. Arnold (Met. Ed. Co.) on "Lessons Learned Short-Term Requirements 2.1.3.b Instrumentation For Detection of Inadequate Core Cooling - Additional Information," dated September 24, 1980).

We will require that Metropolitan Edison provide an acceptable response to the Inadequate Core Cooling requirements, as detailed in our clarification for TMI-2 Task Action Plan II.F.2 (NUREG-0737), by January 1, 1981. Identification of new instrumentation to be installed and the schedule for installation cannot be provided until we review the submittal.

Based on the current status of our review, it is likely that a water level measurement system will be required, but not necessarily prior to restart. In the absence of such a system, the NRC staff will assure that the instrumentation and procedures provided for detection and response to inadequate core cooling conditions are adequate for safe operation.

Laurence E. Phillips

CORE PERFORMANCE BRANCH
DIVISION OF SYSTEMS INTEGRATION
U. S. NUCLEAR REGULATORY COMMISSION

PROFESSIONAL QUALIFICATIONS

I am employed as a Section Leader of the Thermal-Hydraulics Section in the Core Performance Branch of DSI.

I graduated from the University of Cincinnati with a Chemical Engineering degree in 1954. After serving two years as an officer in the United States Army, I have been continuously employed in the nuclear engineering profession since January, 1957. I received a M.S. degree with nuclear physics major from Union College of Schenectady, N. Y., in 1961. I am a registered Professional Engineer, Certificate #E-026547, in the state of Ohio.

In my present work assignment at the NRC, I have supervisory responsibility for the review of the reactor core thermal-hydraulic design submitted in all reactor construction permit and operating license applications. In addition, my section participates in the review of analytical models used in the licensing evaluation of the core thermal-hydraulic behavior under various operating and postulated accident transient conditions. The latter responsibility includes technical review of the functional requirements for core monitoring systems to provide capability for detection and response to inadequate core cooling conditions.

Prior to joining the NRC staff in December, 1974, I was employed by NAI Corporation as a Senior Associate. In this capacity, I was responsible for the development and application of computer codes for analysis of nuclear reactor cores. I acted as a consultant to nuclear operating utilities in the use of these codes for analysis of their operation, and in the solution of general nuclear engineering problems. My tenure at NAI was from 1967 through 1974.

From 1962 to 1967, I was employed by Allis Chalmers Mfg. Co. My assignments during that period included supervisory responsibility for the safety analyses and licensing of the LaCrosse Boiling Water Reactor.

From 1958 to 1962, I was employed by Alco Products where I was project manager for the design, development, and fabrication of heat exchange equipment for nuclear liquid metal projects. Prior to that I was with the Nuclear Division of the Martin Company.

UNITED STATES NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
Metropolitan Edison Company	)	Docket No. 50-289
(Three Mile Island Nuclear	)	
Station, Unit 1 Restart)	)	

NRC STAFF TESTIMONY OF LAURENCE E. PHILLIPS

SUPPLEMENTARY TESTIMONY TO THAT OF LAURENCE E. PHILLIPS

FILED DECEMBER 1, 1980

REGARDING REACTOR WATER LEVEL INSTRUMENTATION

(UCS Contention 7, Sholly Contention 6(b) and

ANGRY Contention 5(b))

Q.1s Metropolitan Edison in their October 2, 1980 letter (TLL 500) to the NRC takes the position that no additional instruments are necessary or desirable since the NRC has not identified any needed improvements in the inadequate core cooling guidelines. Has the staff responded to this position to make clear that additional instrumentation will be required?

A. The position stated in the reference correspondence is no different than the position previously taken and addressed by the staff in the TMI-1 Restart Evaluation, NUREG-0680, June, 1980, pp. C8-20 and 21. Additionally, the present staff position, including rejection of the cited position of the licensee, is clearly stated in the September 24, 1980 letter and SER from D. Eisenhut to R. C. Arnold of Metropolitan Edison. The staff requirements for Metropolitan Edison and all other licensees are further reinforced by the issuance of NUREG-0737 in

November, 1980. Metropolitan Edison, in their letter of November 20, 1980 (TLL 591), did acknowledge the staff requirements and promised to provide a response by January, 1981. The staff has provided no comment on the October 2 letter and will have nothing to add to the discussion and staff position already provided until we complete our review of the January submittal.

Q.2s Since the staff has not identified differences in operator actions if level information were available versus the actions that are prescribed by the existing Inadequate Core Cooling guidelines, what is the need for reactor vessel level indication?

A. The Met Ed/B&W procedures for detection of Inadequate Core Cooling rely primarily on the saturation meter and core exit thermocouples. The saturation meter, while providing a basis for initial actions, does not distinguish between anomalous transients which can drain the pressurizer and cause primary loop saturation due to cooling and shrinkage of primary coolant versus loss of coolant inventory which could lead to inadequate core cooling if it continues.

TMI-1 Emergency Procedure 1202-6B describes the different operator responses to small break LOCA versus overcooling events which cause automatic high pressure injection. These procedures now require the operator to distinguish between the transients based on indirect indicators from existing instrumentation. Vessel level instrumentation, if available, would permit a much quicker and more reliable diagnosis of the conditions. For small break LOCA, an orderly cooldown is required, but not necessarily for an overcooling transient. In both cases, a vessel level meter if available, would provide coordinating information to assist the operator in restoring the water solid primary system (possibly using the upper head vent) and the normal water level in the pressurizer.

For a small break LOCA, the primary system will continue to lose coolant inventory, at a rate and duration dependent on the size and location of the break, until the safety injection make-up flow exceeds the rate of coolant loss. For some conditions, the time interval from the instant of primary system saturation conditions until the occurrence of superheat indication on the core exit thermocouples or hot leg RTDs is in excess of 30 minutes, and possibly up to three hours or more. The superheat condition does not occur until the core is partially uncovered and fuel heat up has begun.

If level instrumentation were available, the effectiveness of HPI in recovering the system and the trend of level indication (continuing to lose coolant or refilling the system) would provide valuable diagnostic information on the nature of the transient before the level drops into the core. The level indication would also provide evidence that the core is covered during recovery from a TMI-2 type flow blockage condition, even though superheat may persist at the core exit thermocouples. None of the process parameters monitored by existing instrumentation provide equivalent information on a continuous basis.

The NRC staff has not evaluated the possible actions the operator would take based on core level instrumentation but has required that the licensees describe how emergency procedures should be modified after level instrumentation is incorporated (Item (7) and (8) of Documentation Required in II.F.2 of NUREG-0737). Met Ed has not yet submitted any analyses or evaluation addressing additional operator actions which could be taken to prevent core uncovery for small break LOCAs if level information were available for prompt diagnosis of the condition.

A recent event at St. Lucie (June 11, 1980) provides a classic example of an anomalous event leading to steam bubble formation in the reactor vessel head. The condition was not recognized by plant operators since instrumentation was not available to detect the low-level condition. The post-event evaluation concluded that unsafe operator actions could have been taken. ("Report on the St. Lucie I Natural Circulation Cooldown on June 11, 1980," by E. V. Imbro, Office for Analysis and Evaluation of Operational Data, USNRC.) Vessel level information would have indicated the void formation in the upper head and thus prevented operator confusion which contributes to the likelihood of unsafe actions. Additionally, vessel level indication would provide a safe bases for manual shut off of HPI to avoid overflow to containment.

Finally, vessel level information is important and possibly essential to proper emergency procedures relating to use of the reactor vessel head vent required by the TMI Action Plan. Vessel level information would indicate the existence of a void in the upper head so that the need for vessel venting could be evaluated. The NRC staff has not evaluated the conditions for which the head vent should be opened and has requested that procedures for use of the vent be provided by the licensee. However, the staff is aware of at least one vent design for which the designer insists on the coupling of level information to the safe operation of the vent.

In summary, the staff believes that reactor vessel level information will enhance the operating safety of PWRs.

CHAIRMAN SMITH: We will take a break for just a moment.

(Board conferring.)

CHAIRMAN SMITH: We are just addressing our travel schedule here while there is still opportunity. We will be checking tomorrow and concluding tomorrow by noon.

MR. CUTCHIN: I have only a couple of rebuttal questions, so how long it takes will depend upon how much cross examination and how many Board questions there are.

CHAIRMAN SMITH: Okay. Mr. Dornsife, do you think that is reasonable that we will be --

MR. DORNSIFE: Yes.

CHAIRMAN SMITH: Also, before you begin, may we throw away Mr. Rubin's testimony?

MR. CUTCHIN: Yes, you may.

BY MR. CUTCHIN:

Q Mr. Phillips, the licensee's witnesses, which consisted of both representatives of Met Ed and the reactor vendor, BMW, have testified that instrumentation to measure reactor vessel water level is unnecessary.

Can you identify whether the position taken by the licensee and BMW is consistent with that taken by other TWR vendors?

A It is not consistent with that taken by other vendors.

Q Would you explain, please?

A Other vendors -- by other vendors I am speaking of the PWR vendors, Westinghouse and Combustion Engineering. They have proposed water level measurement systems which they are recommending to their -- to their customers.

Westinghouse has proposed at Delta-T measurement system. Combustion Engineering has proposed the heated junction thermocouple system. The heated junction thermocouple system, to expand a little bit on the principle of it, is basically one heated junction thermocouple and one regular thermocouple with a reference which is used as a reference temperature.

When the thermocouples are immersed in fluid, the output -- the amount of cooling of the heated junction, the difference between the two temperatures is proportional to the voiding in that particular region. So it is really a direct measure of the cooling in the area of the thermocouple.

These thermocouples are inserted in the top of the reactor vessel and extended down to the top of the core. They will have to be supplemented by other means to measure to the bottom of the core.

Other means would likely for the most part be regular thermocouples inserted in instrument tubes, and the Delta-T measurement system, as proposed by Westinghouse, basically consists of three taps; one in a vessel head; one at the bottom of the vessel; one at the hot leg.

They have done rather extensive analyses of what the response would be to various situations, and they interpret the measurement in terms of measuring mass while the pumps are running.

This is based primarily on the two phase pressure drop, the change in two-phase pressure drop with voiding, and a direct measure of level when the pumps are cut off and not running. So it is a measurement of coolant inventory.

The heated junction thermocouples have been tested at Oak Ridge to a good deal, and Combustion Engineering is running their own test. The staff feels that both of these concepts show high promise and would appear to certainly be a valuable additional instrument to those which are already employed.

Our interpretion of much of the testimony this morning seems to be weighing the level instrumentation as opposed to the other instruments. The staff is looking at this as a complete -- a system of inadequate core cooling instrumentation, and are looking at the interpretation of all the instruments together.

And if sufficient analyses have been performed, say, to identify certain anomalous situations for one instrument which may exist for a short period of time, if that is combined with the output from other instruments which will allow recognition of the anomalous situation, we feel that the likelihood is very high indeed of an acceptable system can be developed.

We feel that the processing of the data and the display to the operators is a very important part of that system, and as such, until the systems are installed, and the operating methods have been identified and the calibration and test data is available and the staff is certain that these systems are indeed a plus as far as safety goes and are not providing information which would lead to unsafe actions, we cannot say in advance a system is acceptable.

That is essentially where we stand.

DR. JORDAN: I do not quite undertand now. Are you saying this is a diverse method of measuring inadequate core cooling, that the thermocouples or whatever provide a complete system which tells you -- gives you an approach to inadequate cooling quite apart from all the other systems? Is that what you meant? Is that what you were saying? I did not quite really understand.

WITNESS PHILLIPS: I am saying that when we -- let's say if there is a leak in the reactor vessel head and we have a Delta-t measurement system as an example, analyses have identified that there is a situation where for a brief period of time under that situation you can get a false indication on levels.

Using other instruments, other data which are available, you can identify these situations and negate the indication during that period as far as the operator is concerned, during the period when an ambiguity may exist.

So there is a matter of interpretation of data in the design of the system, as well as just putting in the system itself. When we look at the system we look at the core exit thermocouples. Certainly they are a valuable indicator, and we have never proposed that levels should replace them. They should be another confirmatory source of information, and we just cannot believe that if a good source of level measurement is available, that the operator will be told to ignore that information.

DR. JORDAN: Then I think I did misunderstand. I think I got it backwards. In fact, what you are saying is that the core level instrumentation supplements all the other information. It is not adequate in itself. It does not do the job all by itself.

WITNESS PHILLIPS: That is correct.

DR. JORDAN: Okay. Then I just misunderstood, and I wanted to get that straightened out. You go ahead.

BY MR. CUTCHIN:

Q Mr. Phillips, I believe you may have answered part of this, but for clarity in the record, licensee's witnesses also testified that reactor vessel water level information in accident or transient situations may in their view tend to confuse the operator.

Would you comment on that?

A I believe I did perhaps jump the gun in answer to that. This would be a situation which has been identified beforehand by rather extensive analyses, situations where a false indication may be available for a short period of time, and if other indicators are used in the interpretation of the information, that can be processed either automatically in a data processing system or, depending on the complexity, by instructions such that such confusing interpretation need not exist.

MR. CUTCHIN: Mr. Chairman, these are all of my rebuttal questions. However, in light of the hour and the fact that we have an outstanding question which Dr. Little had which appeared on page 7488 and 7489 of the transcript and had to do with her expression of some confusion in interpreting page 3-113 of new reg 0737, I thought that perhaps since it looks like we won't get into cross examination today, it may be appropriate to go ahead and have Mr. Phillips, who is the person managing the review of the requirements of that past action plan, try to remove, if he can, some of Dr. Little's confusion.

CHAIRMAN SMITH: Any objections to that?

(No response.)

CHAIRMAN SMITH: Let's review what the question is.

MR. CUTCHIN: I have the transcript pages that I would be happy to try to identify the question. The pages, I said, again are page 7488 and 7489.

Dr. Little said on 7488, I have been confused for some time now because I have read the section in new reg 0737 on instrumentation for protection of inadequate core cooling, and the new requirements for in-core thermocouples, and when I read that first page there it looks to me as though CRT and computer capability for reading those is being encouraged rather than discouraged.

Later on she says, let me tell you the page that is confusing me. If somebody has a chance to read it and come back sometime later in the hearing and explain it, I would be interested in hearing it. It is page 3-113 and is of new reg 0737, dealing with Roman numeral II.F.2.

WITNESS PHILLIPS: Yes. Could you be more specific please?

DR. LITTLE: Let me review it quickly.

(Pause.)

MR. CUTCHIN: I am sorry I caught you somewhat by surprise, but I thought with the short time left it might be a good time to take it up.

DR. LITTLE: I did have the page already opened there.

CHAIRMAN SMITH: I think you may have the witness by surprise.

MR. CUTCHIN: I hope not because we discussed it over the noon hour.

(Pause.)

WITNESS PHILLIPS: The statement to be read concerning our emerging computer system is a correct interpretation. That was an objective, and we feel that that will aid in the processing of a number of signals -- a number of indicators in interpreting and displaying to the operator such that he sees what he needs to know and doesn't have to interpret it on his own.

DR. LITTLE: I believe discussion arose when we were talking about use of CRT and the fact that totally qualified CRT displays were not available, but that the indication that was given in the section was that nevertheles, these would be encouraged -- CRT displays of in-core conditions would be encouraged, and that it felt that it was warranted because -- while this relaxation can be accomplished without compromise of ICC monitoring reliability by requiring 99 percent availability for the display systems.

That was the main thing, and by relying on diverse methods of ICC monitoring that include completely qualified display systems.

WITNESS PHILLIPS: Yes. That is referring to having a backup system available which is hard-wired. Let's see. I believe this is -- this is referring to the in-core thermocouple system here, and if you need the -- if you go over all of the requirements, you will find that where the CRT display is the primary system and the CRT is not seismically qualified, there is in all cases a hard-wired backup system which can be used to provide these same information in not such a good fashion, you might say.

But that is available in qualified systems.

DR. LITTLE: This leads us to the question of whether the operator is relying on something less reliable than what he has available already.

WITNESS PHILLIPS: No. He will still have available what he has available already. He is relying on a system which more completely processes the information available and displays it to him in a fashion which is much easier to understand, and it is a highly reliable system.

It is just that it does not meet seismic criteria, and some of the more stringent design criteria which would make, number one, a huge procurement problem in obtaining it on any sort of a reasonable schedule, if it could be obtained at all.

But in all cases we have looked at the information that is required, and in all cases we are requiring a backup hard-wired system which does meet seismic criteria which he would be able to make do on in the unlikely event that he has an earthquake, as he is making do on now without the additional instrumentation in the system.

For instance, the facts are determined to some degree; we agree with the licensee it is possible on the instrumentation that they have now to manage to detect inadequate core cooling.

What the staff and the Commission has determined is that an additional margin of safety can be obtained, and the techniques can be enhanced by providing additional instrumentation.

If we felt that what now exists, that there was not a way to do the job operating, say, somewhat in the dark, we would have recommended closing down all the plants. We do not think that is the case. We do think that the system can be improved, and we would look on the CRT system as an improvement.

This will not take away anything that already exists. It will still be there.

DR. LITTLE: I would assume then that the CRT display would be something that would, if in the unlikely event that there were seismic problems, that the CRT would not give information, rather than to continue to give information, but inaccurate information.

WITNESS PHILLIPS: I would say it would either give -- it is likely it would either give adequate informatin or it would not give it at all.

DR. LITTLE: So the operator would know immediately that he should use the backup. He would not be misled by getting inaccurate information over the CRT.

WITNESS PHILLIPS: Yes. I would say that that certainly should be a design consideration and would be something that would be looked at as part of the design.

DR. LITTLE: Could you elaborate a little on this sentence earlier in that paragraph, speaking to the feasibility of use of CRT displays for proper interpretation of reactor water level systems under development?

WITNESS PHILLIPS: I am sorry. I am having trouble finding that.

DR. LITTLE: It is about the section we are talking about. It is about six or seven lines above the bottom of the page, 3-113.

WITNESS PHILLIPS: Where it says, and make feasible the use?

DR. LITTLE: Of CRT displays that may be needed for proper interpretation of some reactor water level systems under development.

WITNESS PHILLIPS: Yes. That gets at the situation I was talking about earlier where if you have an anomalous situation which may occur on one indicator. If that indicator were looked at alone, it could be an ambiguous interpretation.

You may want to feed in additional figures which would allow automatic interpretation, and either make a signal as far as the operator was concerned at the time that it is unreliable, or process it in a manner that properly interprets it for feed display to the operator.

DR. LITTLE: So you are going to have some program that will tell which of the signals coming in is anomalous and which is not?

WITNESS PHILLIPS: Yes.

DR. LITTLE: I guess I would like to hear Mr. Jones and Mr. Keaten's opinion some time before we finish, the feasibility of doing that.

MR. CUTCHIN: I would like to point out, just for additional information in the record at this point, Dr. Jordan had some questions yesterday on this same general area in connection with one of the areas of this agreement between staff and licensee on human factors engineering.

That appeared start on page 10,523.

DR. JORDAN: Before you get to that, stay with the same page that you are on. It is just a matter strictly of clarification.

Two or three sentences above where you were talking about, you say, the pertinent portions of draft regulatory guide 1.97 have now been included as Appendix A. What is Appendix A?

WITNESS PHILLIPS: That is an error. It should have been Appendix B. It was Appendix A at the time this appeared. It later became Appendix B.

Appendix B is an extract from regulatory guide 1.97 in the draft form. We did not want to refer to a draft regulatory guide as a basis for the design of the systems.

DR. JORDAN: Appendix B to what?

WITNESS PHILLIPS: This document.

MR. CUTCHIN: New reg 0737.

DR. JORDAN: That will help. I looked at Appendix A and did not find it. Now if I look to Appendix B, I will find something.

WITNESS PHILLIPS: Right, I hope.

DR. JORDAN: That is all I needed to know, was where to look. I have not read it yet. I don't know what it says.

WITNESS PHILLIPS: It has the seismic defined criteria and all that other good stuff in it.

DR. JORDAN: Yes, all that other good stuff. I understand.

MR. CUTCHIN: That was all I had. I just wanted to tie the two pieces of the transcript together by citing the other page number. The discussion of the backup display for in-core thermocouples independent of the process computer took place yesterday at page -- starting at page, transcript 10,523.

I just wanted it to appear here for a reference back.

DR. JORDAN: All right. I see.

DR. LITTLE: Do you know of any vendor or operating plant or any entity which is close to having a program of the type which you just described which would display on the CRT how do you determine which of the readings is correct and which is not?

WITNESS PHILLIPS: We have some systems. This required some design descriptions of the system on January 1, 1981 recently, so those documents are just coming into the house, and as yet we have not -- by the house, I mean NRC.

DR. JORDAN: These are Westinghouse?

WITNESS PHILLIPS: Westinghouse documents.

DR. JORDAN: We will get into those tomorrow.

WITNESS PHILLIPS: And we have not actually commenced our review of them, but they have -- I would say that the Westinghouse generic documents have three different levels of data processing on the systems.

We have not looked at them. I do not know what they are, but they are the -- the more sophisticated one would incorporate, I am sure, quite a bit of assessing data processing.

DR. LITTLE: What is your opinion about in the instrumentation for measuring water level without having a CRT display with the appropriate program?

WITNESS PHILLIPS: I am sorry. I did not catch that.

DR. LITTLE: I am asking you what your opinion is about having the instrumentation for measuring the reactor water level, but not having an accompanying program to interpret its significance.

WITNESS PHILLPS: It would depend on the system and whether there -- how complicated the interpretation is and where ambiguous connotations might occur.

So we would have to look at the analyses of how the system would perform under various circumstances. We would have to consider when and error might exist and what danger there is of that error contributing to operator confusion; whether just by simple training procedures and in the recognition and interpretation that he could operate satisfactorily in the manner that the information is displayed.

So I would say we are not in a position to say -- it would depend very much on our review of the specific system. It is quite possible that a system can be proposed and will be proposed without complicated data processing, which we may very well find acceptable.

DR. LITTLE: Would you care to make sort of a broad time range estimate of how long it would take to do a review of this type?

WITNESS PHILLIPS: Yes.

DR. LITTLE: Days, month, years, whatever.

WITNESS PHILLIPS: If you will bear with me until I find the right piece of paper.

(Pause.)

We have established a review schedule based on licensee submittals on January 1, 1981. We have a national lab contact to assist us in these reviews. We are currently in the process of developing generic positions and criteria for acceptance of the individual systems, and we expect that to be complete about April 1.

We accept that our questions and positions on the generic systems will be transmitted to the licensees adopting those particular systems on July 1. The licensee submittals and in responses to our questions and positions would expect on September 1. We would expect to have generic SER's and model tech specs issued on September 1, 1981.

The installation is scheduled for January 1, 1982. We expect that that will vary from plant to plant. It is not likely a shutdown will be required to put it in at that time. It will depend on their normal fueling outage, ad it would be next to impossible to do all of these at once anyway.

The licensee submittal would then have to make another submittal after installation giving the details of his installed system and the calibration and the tested data, and we would expect that about March 1, predicated on a January 1 installation.

We would then issue tech specs and planned specific approvals for implementation on May 1, 1982. The tech specs and plant-specific approvals indicating that these are -- the staff has accepted the systems and the way they operate for a specific plant would be required before they were actually implemented into the operating procedures, into the emergency procedures.

So our review would be complete, and the plant-specific SER's issued by July 1, 1982. That is our current review schedule.

If I may just highlight that for you, I think the key is that generic SER's and model tech specs which approve systems and in principle we would expect to be towards the end of the year, around December 1, just about the time the installations of the systems are starting; then based on the test data and the calibration of individual systems, we would approve the plant-specific approvals and that would be going on until about mid-1982.

(Board conferring.)

DR. LITTLE: Sometime tomorrow morning I would like to hear from the licensee as to whether or not having a program in place which would help interpret the information from the reactor level instrumentation would alleviate some of their concerns they expressed today.

MR. BAXTER: Both Mr. Jones and Mr. Keaten will be here tomorrow.

CHAIRMAN SMITH: Mr. Cutchin?

MR. CUTCHIN: Nothing further. As far as I am concerned, the witness will be available for cross as soon as you need him in the morning.

CHAIRMAN SMITH: Are there any questions on the special question that cannot be dealt with tomorrow? We will be here. Everyone will be here.

All right. Shall we meet at 9:00 o'clock?

(Whereupon, at 5:06 p.m., the hearing was adjourned, to reconvene at 9:00 a.m. on Thursday, January 22, 1980.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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in the matter of: METROPOLITAN EDISON COMPANY (TMI Unit 1)

Date of Proceeding: January 21, 1981

Docket Number: 50-289 (Restart)

Place of Proceeding: Harrisburg, Pa.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

David S. Parker

Official Reporter (Typed)

[signature]

(SIGNATURE OF REPORTER)