LIC 9/15/80

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)

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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.

INDEX

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.)

BY WITNESS KEATEN:

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 additiona. instrumentation is required to detect inadequate core cooling. We also disagree with UCS and ECNP 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.

- o Reactor Coolant System Elevations and Manometer Effects (4 hours) - theory of, and recognition and response to, manometric behavior during transients.
- o TMI-2 Transient (4 hours) - cause of and response to gas/steam binding affecting core cooling.
- o 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 ot. er 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.

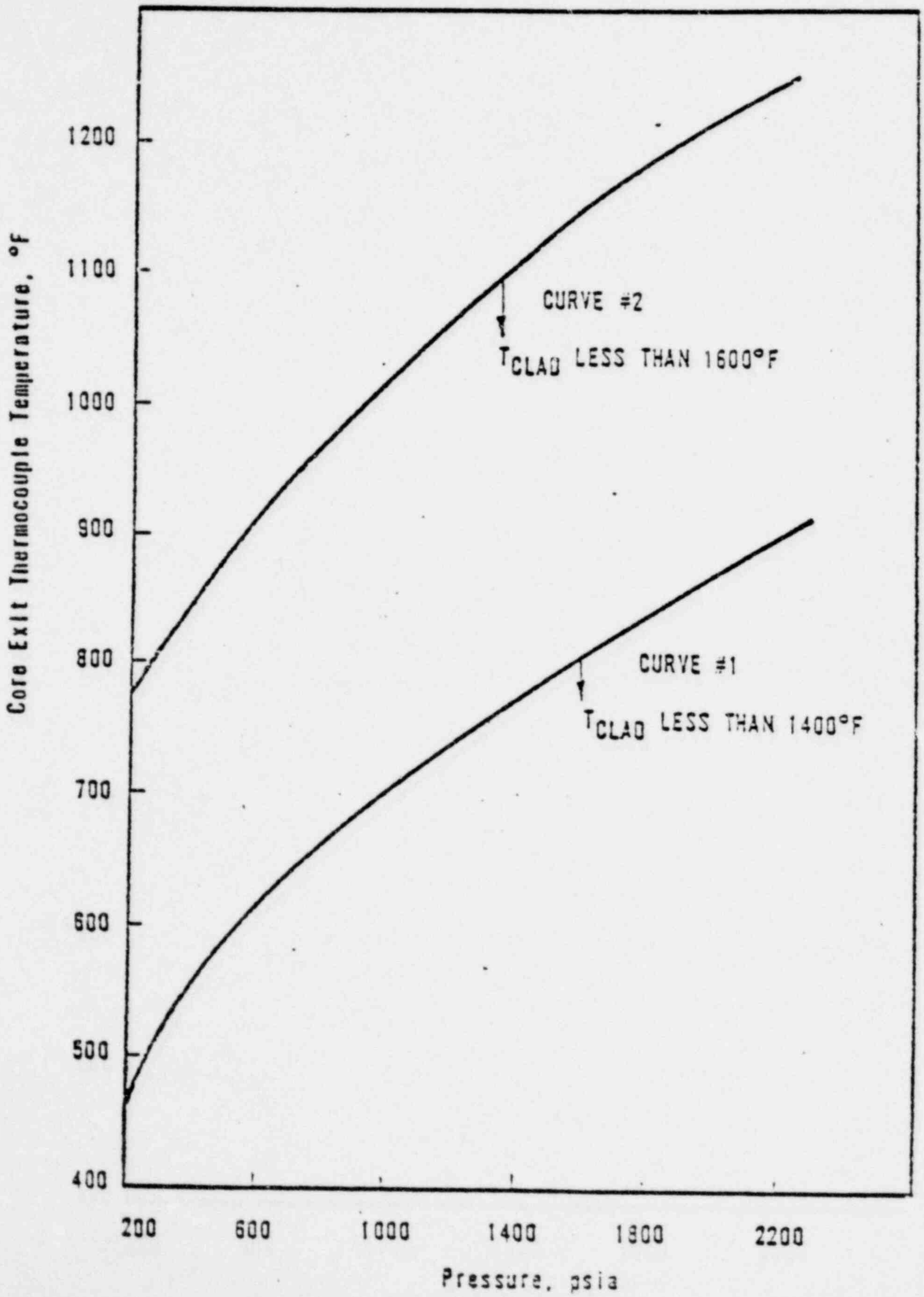
RESPONSE TO SHOLLY CONTENTION NO. 6(b)

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



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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.

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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

Professional <u>Affiliations</u>: Babcock & Wilcox Owner's Group, Fuel Handling Subcommittee.

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Education:	B.S., Nuclear Engineering, Pennsylvania State University, 1971. Post Graduate Courses in Physics, Lynchburg College.
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.