UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMSSICN

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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

Friday, November 7, 1980

Evidentiary hearing in the above-entitled matter was resumed, pursuant to adjournment, at 9:00 a.m.

BEFORE:

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

DR. WALTER H. JORDAN, Member

DR. LINDA W. LITTLE, Member

Also present on behalf of the Board:

MS. DORIS MORAN,
Clerk to the Board

APPEARANCES:

On behalf of the Licensee, Metropolitan Edison Company:

GEORGE F. TROWBRIDGE, Esq.
THOMAS BAXTER, Esq.
Shaw, Pittman, Potts and Trowbridge,
1800 M Street, N.W.,
Washington, D. C.

8011170580

On behalf of the Commonwealth of Pennsylvania:

ROBERT ADLER, Esq.
Assistant Attorney General,
505 Executive House,
Harrisburg, Pennsylvania

On behalf of Union of Concerned Scientists:

ELLYN WEISS, Esq.,
ROBERT D. POLLARD
Harmond & Weiss,
1725 I Street, N.W.
Washington, D. C.

On behalf of the Regulatory Staff:

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

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EXHIBITS

P R O C E E D I N G S

CHAIRMAN SMITH: Are you ready, Mr. Baxter?

MR. BAXTER: We may have a preliminary matter.

MR. CUTCHIN: Mr. Chairman, may I make a few preliminary remarks?

CHAIRMAN SMITH: Yes, please.

MR. CUTCHIN: Mr. Chairman, the staff noticed yesterday that perhaps the Board had some concerns with respect to the scenario that took place at TMI 2, that we were not able to fully address, although we are satisfied that the evidence that we put in yesterday satisfactorily addresses the contention that was being addressed, and we feel that otherpieces of the puzzle, if you will, will be supplied in our later evidence.

If it would be of assistance to the Board, the staff would be happy to make available early next week or at the Board's convenience a staff witness who does have an in-depth understanding of the Three Mile Island 2 accident scenario, who could address whatever concerns the Board may wish to raise.

(Pause.)

DR. JORDAN: We are not so much concerned with picking up again on UCS 1 and UCS 2. However, the contention that we will be addressing today, UCS 8, and the Board questions concerning it, will require, we believe, expertise from the staff on the TMI 2 accident, since the TMI 2 accident was indeed a small break accident, and I was going to mention to the staff this morning that in my opinion the written testimony that has been submitted by the staff is inadequate, and I will be calling on the staff to address each one of the recommendations, as did the licensee, in responding to the Board question.

I will ask the Board to have someone who can, who has analyzed the licensee's response to each of the recommendations and either agrees or disagrees that the response from the licensee is adequate.

So, if the witness proposed is not adequate, or prepared to do that, then it would behoove the staff to have someone here who can indeed address the Board's question on UCS 8.

MR. CUTCHIN: Then we will make every effort to have the appropriate person here when our turn comes next week.

CHAIRMAN SMITH: Another observation. You correctly observed that the Board was having some difficulty yesterday. In retrospect, I think that part of the problem was that your witness was presented for a rather narrow point, which point was indeed covered in his testimony. However, he tried to place a narrow point into context, and it was that context effort that caused the difficulties.

I think in the future when this arises, I think we ought to take more pains at the beginning to separate, you know, what is naturall: told when you are trying to put your narrow story into context from the actual context itself. I think that might be helpful.

MR. CUTCHIN: We will make an attempt to do that in the future, Mr. Chairman.

MS. WEISS: Mr. Chairman, if the staff i going to put a new witness on to address these questions, and he is going to present new testimony, we will ask that we be given that in writing at least five days in advance as the rules provide. That is not just a technicality. That is for us to prepare.

CHAIRMAN SMITH: Well, Ms. Weiss, if you might recall -- yes, you certainly are not going to ever have to address any evidence in this case as to which you have not had full notice. However, in this particular case, you are going to have to follow the Board's wishes. We may give you five. We may not. This is a contention that UCS introduced, failed to pursue, the Board took up. We may ask you to work fast and hard on it to keep up.

MS. WEISS: If you are talking about one and two, we didn't drop those. Eight is correct. The Board is pursuing on our request. I would just like to know if we are going to have this testimony in writing in advance.

CHAIRMAN SMITH: You may or may not, depending on the circumstances. We will try to do it.

MS. WEISS: Could I ask that the Board inquire what staff's intention is?

CHAIRMAN SMITH: Concerning what?

MS. WEISS: Concerning production of this witness. I thought Mr. Cutchin said he was going to put him on the stand early next week.

CHAIRMAN SMITH: Mr. Cutchin offered for the Board's consideration, and I don't recall that he had a specific proposal.

MR. CUTCHIN: I understood Dr. Jordan to say that he didn't need any further witness on UCS 1 and 2, but he thought it would be important to have someone here for the next group of contentions that could indeed address the Three Mile Island 2 scenario, and it would be our intent to offer that individual as a live witness, and he would give his oral testimony in response to the concerns that Dr. Jordan raised, and that is to address -- try to address each of the applicant's responses to the specifics of the Board's questions on UCS 8.

I don't have a name for that person at this moment, so obviously I don't have anything in writing. But we will make every effort to accommodate the Board's wishes by next week, and it will have to be live.

CHAIRMAN SMITH: You don't believe you could have written testimony? What is there of that next week?

MR. CUTCHIN: Could I have a moment, please?

CHAIRMAN SMITH: All right.

(Pause.)

MR. CUTCHIN: We will be getting together, Mr. Chairman, with our witness back in Bethesda early Monday morning, and we will just have to see what we can do between then and the time the session takes up next week.

CHAIRMAN SMITH: Right. There is nothing about next week which is magic for addressing this. The Board would prefer having more deliberate consideration of the problem plus the opportunity for UCS to have some warning to meet the substantive requirements of advanced warning and what they have to confront, so that next week does not have to be the time for the testimony.

MR. CUTCHIN: I understand, sir, but we will make every attempt to be able to address each of the items in 0565 and 0623 that the Board's questions covered as early as we can. But if it is going to be immediately following the licensee's testimony, it may well be live, and if the Board has a different preference, of course, we will accommodate that.

CHAIRMAN SMITH: I am just trying to satisfy the substantive rights that intervenors and other parties may have to have some advance notice of it, whether it be live or written. That may require more time for that purpose alone.

MR. CUTCHIN: It may well be, Mr. Chairman, that we look at each of the licensee's responses and say we have no problem with it, in which case there is no substantive problem, and they have had warning, but we will just have to see how it develops.

CHAIRMAN SMITH: Mr. Baxter?

MR. BAXTER: Yes. Licensee calls to the stand Mr. T. Gary Broughton and recalls Mr. Robert C. Jones, Jr.

CHAIRMAN SMITH: Would you state yourname, please?

MR. BROUGHTON: My name is Thomas Gary Broughton.

Whereupon,

THOMAS GARY BROUGHTON

and

ROBERT C. JONES

were called as witnesses, and having been first duly sworn by the Chairman, were examined and testfied further as follows:

FURTHER DIRECT EXAMINATION

BY MR. BAXTER:

Q Gentlemen, I would like to call your attention to two documents, each of which bear the caption of this proceeding. The first is dated September 15, 1980, and it is entitled Licensee's Testimony of Robert C. Jones, Jr., and T. Gary Broughton in Response to UCS Contention Number 8 and ECNP Contention Number 1(a), Additional LOCA Analysis.

The second document, which is dated October 28th, 1980, is entitled Licensee's Testimony of Robert C. Jones, Jr., and T. Gary Broughton in Response to The Board Question on UCS Contention 8.

Do these documents include testimony that you prepared under your direct supervision for presentation in this hearing. Mr. Jones?

A (WITNESS JONES) No.

Q Mr. Broughton?

A (WITNESS BROUGHTON) No.

Q Do you have any changes or corrections to make to the testimony which is associated with your name in these documents? Mr. Jones?

A (WITNESS JONES) No.

Q Mr. Broughton?

A (WITNESS BROUGHTON) No.

Q Is the testimony associated with your name in these documents true and accurate to the best of your knowledge and belief? Mr. Jones?

A (WITNESS JONES) Yes.

Q Mr. Broughton?

A (WITNESS BROUGHTON) Yes.

MR. BAXTER: Mr. Chairman, I would ask that each document be received into evidence and incorporated into the transcript as if read.

CHAIRMAN SMITH: Without objections, we will receive the documents.

(The documents referred to follow:)

(1)

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- 89
	)	(Restart)
(Three Mile Island Nuclear Station, Unit No. 1)	)	
	)	

LICENSEE'S TESTIMONY OF

ROBERT C. JONES, JR., AND T. GARY BROU HTON

IN RESPONSE TO UCS CONTENTION NO. 8

AND ECNP CONTENTION NO. 1(e)

(ADDITIONAL LOCA ANALYSIS)

OUTLINE

The purposes and objectives of this testimony are to respond to UCS Contention 8, which asserts that adequate small-break loss of coolant accident (LOCA) analyses have not been performed, and to respond to related Board questions. The testimony addresses the small-break LOCA analyses performed prior to the TMI-2 accident and their conformance to 10 CFR Part 50, Section 50.46. The purpose, assumptions and results of small-break analyses subsequent to the TMI-2 accident are described. Operating guidelines and procedures for small-break LOCA mitigation are discussed. It is shown that adequate protection for small-break LOCA's is provided.

<u>INDEX</u>

INTRODUCTION

This testimony, by Mr. Robert C. Jones, Jr., Supervisory Engineer, ECCS Analysis Unit, Babcock & Wilcox Company, and Mr. T. Gary Broughton, GPU Control and Safety Analysis Manager, is addressed to the following contention:

UCS CONTENTION NO. 8

> 10 CFR 50.46 requires analysis of ECCS performance "for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered." For the spectrum of LOCA's, specific parameters are not to be exceeded. At TMI, certain of these were exceeded. For example, the peak cladding temperature exceeded 2200° fahrenheit (50.46(b)(1)), and more than 1% of the cladding reacted with water or steam to produce hydrogen (50.46(b)(3)). The measures proposed by the staff address primarily the very specific case of a stuck-open power operated relief valve. However, any other small LOCA could lead to the same consequences. Additional analyses to show that there is adequate protection for the entire spectrum of small break locations have not been performed. Therefore, there is no basis for finding compliance with 10 CFR 50.46 and GDC 35. None of the corrective actions to date have fully addressed the demonstrated inadequacy of protection against small LOCA's.

ECNP Contention 1(e) was accepted only to the extent that ECNP was permitted to adopt UCS Contention 8. Consequently, ECNP Contention 1(e) is not quoted here. (See, Board Memorandum and Order, September 8, 1980.) UCS withdrew its sponsorship of UCS Contention 8, which has been adopted as a Board Question (See, Board Memorandum and Order of Prehearing Conference of August 12-13, 1980, dated August 20, 1980).

RESPONSE TO UCS CONTENTION NO. 8

BY WITNESS JONES:

UCS Contention 8 asserts that analyses to demonstrate conformance with 10 CFR Part 50, Section 50.46 (10 CFR 50.46) for the entire spectrum of small-break loss of coolant accident (LOCA) locations have not been performed. Additionally, it is stated that none of the corrective measures being implemented for TMI-1 assure that adequate protection is provided for small-break LOCA's. Contrary to the contention, compliance with 10 CFR 50.46 has been demonstrated and adequate protection for small-break LOCA's is provided.

Prior to the TMI-2 accident, small-break LOCA evaluations had been performed to verify conformance of TMI-1 to 10 CFR 50.46. In order to perform these analyses, the break location which imposes the most severe requirements on the ECCS was identified. As a result of this identification, an analysis was performed of the core flood line break, which results in only one core flood tank and one high pressure injection train available to mitigate the accident under the worst single failure assumption. Also, an analysis of a spectrum of breaks in the reactor coolant pump discharge piping was performed, as this location results in the loss of a portion of the high pressure injection fluid. These analyses were performed using the B&W ECCS evaluation model which has been approved by the NRC as meeting the requirements of Appendix K to 10 CFR Part 50. The actual analyses which were performed are contained in References 1 and 2, and are summarized in Table 1. For the worst case break, the peak cladding temperature was less than 1100°F and no metal-water reaction nor cladding rupture were calculated to occur. Therefore, conformance to 10 CFR 50.46 was demonstrated.

The analysis performed prior to the TMI-2 accident assumed the use of only safety-grade equipment for accident mitigation, and assumed no mitigating operator actions within ten minutes of the initiating event, except as follows:

- Emergency feedwater was assumed to be available.
- Operator action to cross-connect the High Pressure Injection System (HPI) was determined to be required in the event of a small break in the reactor coolant pump discharge piping and the postulated failure of the HPI train which discharges into the unbroken coolant loop.

BY WITNESS BROUGHTON:

With regard to the first of the above items, Licensee's testimony on the Emergency Feedwater System (in response to Licensing Board Question No. 6) will address the reliability of the Emergency Feedwater System (EFW).

BY WITNESS JONES:

In the event of a loss of all feedwater following the EFW upgrade, the feed and bleed mode of emergency cooling is available for LOCA mitigation. See Licensee's testimony on Natural and Forced Circulation (in response to UCS Contentions Nos. 1 and 2).

BY WITNESS BROUGHTON:

With regard to the second of the above items, modifications to the high pressure injection lines have been made to add cross connections and flow limiting devices to ensure sufficient flow without operator action (See, TMI-1 Restart Report, Supplement 1, Part 3, responses to questions 1, 2 and 3).

BY WITNESS JONES:

Subsequent to the TMI-2 accident, additional small-break LOCA analyses were performed. In light of the fact that the severity of the TMI-2 accident was aggravated by operator actions, the purpose of these analyses was to provide an improved analytical basis for emergency operating procedures for small-break LOCA's, not to demonstrate compliance with 10 CFR 50.46. A description of the events analyzed, the key assumptions, and the results of the evaluations are provided in Tables 2 through 8. The analyses performed included an extension of the lower end of the break spectrum previously analyzed, an assessment of the effect of failures in the feedwater system, and an assessment of small-break LOCA's with delayed reactor coolant pump trip. From these analyses it was concluded that multiple failures must occur before a LOCA scenario can result in a challenge to 10 CFR 50.46 limits. A summary of the results of the analyses is also provided below:

- In the event of a loss of all feedwater, without a small-break LOCA, operator action within twenty minutes to either establish emergency feedwater or manually actuate high pressure injection assures that the core remains covered, thus assuring adequate core cooling. (See Table 2.)
- In the event of a small-break LOCA with loss of all feedwater, ECCS may not be automatically actuated. For this situation, operator action within twenty minutes to either establish emergency feedwater flow (which will in turn result

in automatic ECCS actuation), or to manually actuate high pressure injection, assures that the core remains covered, thus assuring adequate core cooling. (See Table 3.)

- In the event of a loss of main feedwater followed by the pressurizer power operated relief valve (PORV) opening and failing to close, the automatic actuation of high pressure injection is sufficient to assure adequate core cooling. (See Table 4.)
- In the event of the pressurizer PORV opening and failing to close, followed by the loss of all feedwater, the automatic actuation of high pressure injection is sufficient to assure adequate core cooling. (See Table 5.)
- For certain very small breaks (between 0.005 ft^2 and 0.01 ft^2) which cause a loss of coolant inventory at a rate in excess of the capacity of high pressure injection, the steam generators would normally be utilized to remove a portion of the energy added to the primary system fluid by core decay heat. During the transition from natural circulation to the boiler-condenser

mode of cooling (see Licensee's testimony on Natural and Forced Circulation in response to UCS Contention No. 1), an interruption of the energy removal process from the primary system will occur due to void formation in the hot legs, and the primary system pressure will increase. However, the subsequent establishment of steam condensation by the steam generators as a heat removal mechanism controls the repressurization and assures that the core remains covered, thus assuring adequate core cooling. (See Table 6.)

- If the reactor coolant pumps operate continuously throughout the LOCA transient, or are tripped promptly upon receipt of a low reactor coolant pressure safety signal, adequate core cooling is provided for all break sizes. For certain break sizes (between 0.025 ft^2 and 0.2 ft^2), adequate core cooling has not been demonstrated if the reactor coolant pumps remain in operation and are subsequently tripped at certain times in the transient. Therefore, in order to assure adequate core cooling, the reactor coolant pumps should be tripped promptly following automatic initiation of high pressure injection. (See Table 7.)

- In the event of a very small LOCA with loss of all feedwater, system repressurization may actuate the PORV which can subsequently stick open. For this situation, operator action within twenty minutes to either establish emergency feedwater flow (which will in turn result in automatic ECCS actuation) or to manually actuate high pressure injection assures that the core remains covered, thus assuring adequate core cooling. (See Table 8.)

Similar to the pre-TMI-2 analyses, the analyses performed after the accident assumed the use of only safety-grade equipment for accident mitigation, and assumed no mitigating operator actions within ten minutes of the initiating event, except for the two items previously identified (at page 3 above) and the manual action of tripping of the reactor coolant pumps following automatic initiation of high pressure injection.

The system behavior which results in the instruction for pump trip involves an extended loss of inventory due to continuous operation of the reactor coolant pumps. While continued pump operation provides forced circulation cooling of the core, it also causes more fluid inventory to be discharged out the break than would otherwise occur for certain break sizes. As a result of this increased loss of inventory, the fluid in the Reactor Coolant System will evolve to a high void fraction. If the pumps are tripped after a high void fraction is reached, the available water in the Reactor Coolant System would not be sufficient to keep the core covered. If the core is significantly uncovered, the cladding temperature would begin to increase and the ECCS may not provide, under these conditions, reflooding of the core at a rate which assures that cladding temperatures are maintained within the criteria of 10 CFR 50.46. Since all analyses have confirmed that the plant can be maintained in a safe condition (as defined by 10 CFR 50.46) during a small-break LOCA without the reactor coolant pumps operating during the transient, provision for prompt tripping of the pumps upon indication of a LOCA (receipt of a low reactor coolant pressure safety injection signal) assures that adequate core cooling is provided. While other, non-LOCA events may lead to a low pressure safety signal, tripping of the reactor coolant pumps for these events still allows adequate core cooling to be provided.

BY WITNESS BROUGHTON:

The generic analyses performed by B&W are applicable to TMI-1. The low pressure reactor trip setpoint has been raised to 1900 psig and the Engineered Safety Features Actuation

System (ESFAS) setpoint has been raised to 1600 psig, the values assumed in the generic analyses. (See TMI-1 Restart Report sections 11.2.11 and 11.2.12).

BY WITNESS JONES:

Based upon the analyses described above, B&W has also developed operator guidelines for managing small-break LOCA's. These guidelines contain two parts: Part I provides the guidelines which define operator actions during a small-break LOCA; Part II provides a description of plant behavior during a small-break LOCA and discusses the effect of the operator actions given in Part I.

BY WITNESS BROUGHTON:

TMI-1 procedures have subsequently been developed to implement these guidelines. The TMI-1 Emergency Procedures which implement the B&W Loss of Coolant Accident guidelines place strong emphasis on maintaining reactor coolant system pressure-temperature relationships to assure that a subcooling condition of at least 50°F exists. Specifically, procedures require that upon automatic initiation of high pressure injection, flow shall not be reduced unless: (1) low pressure injection pumps are in operation and flowing at a rate of not less than one thousand gallons per minute each and the situation has been stable for 20 minutes; or (2) all hot and cold leg temperatures are at least 50°F below the saturation temperature for the existing reactor coolant system pressure and the flow reduction is necessary either to prevent pressurizer level from going off scale high or to avoid excessive reactor vessel pressure/downcomer temperature limits. If 50°F subcooling cannot be maintained, the procedure requires the high pressure injection system to be reactivated. In situations where high pressure injection is manually initiated, flow reductions are permitted only if reactor coolant system pressure is above 1600 psig and the 50°F subcooling margin exists and can be maintained, or if the criteria for flow reductions following automatic initiation are satisfied.

BY WITNESS JONES:

In summary, extensive small-break analyses have been performed for the TMI-1 facility. These analyses demonstrate that small LOCA's can be mitigated within the criteria of 10 CFR 50.46. Also, additional small-break analyses have been performed in order to develop improved emergency procedures. Thus, contrary to the contention adequate protection for small LOCA's has been demonstrated and is provided.

References

1. Topical Report, BAW-10103A, Rev. 3, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," July 1977.
2. Letter, J. H. Taylor (B&W) to S. A. Varga (NRC), July 18, 1978.

TABLE 1

PRE-TMI-2 LOCA EVALUATIONS

Topical Report BAW-10103A, Rev. 3

- Core Flood Tank Line Break
- 0.5 ft^2 Reactor Coolant Pump Discharge Piping Break
- 0.04 ft^2 Reactor Coolant Pump Suction Piping Break

Letter Report, J. H. Taylor (B&W) to S. A. Varga (NRC), July 18, 1978

- 0.15 ft^2 Reactor Coolant Pump Discharge Piping Break
- 0.10 ft^2 Reactor Coolant Pump Discharge Piping Break
- 0.085 ft^2 Reactor Coolant Pump Discharge Piping Break
- 0.07 ft^2 Reactor Coolant Pump Discharge Piping Break
- 0.055 ft^2 Reactor Coolant Pump Discharge Piping Break
- 0.04 ft^2 Reactor Coolant Pump Discharge Piping Break

TABLE 2

LOSS OF ALL FEEDWATER WITHOUT SMALL-BREAK LOCA

Sequence of Events and Assumptions

- Loss of main feedwater occurs.
- Direct trip on loss of feedwater fails and reactor trips on high reactor coolant pressure.
- Loss of offsite power occurs coincident with reactor trip.
- Emergency feedwater is not provided to steam generators.
- Reactor coolant pressure continues to increase.
- Pressurizer PORV does not open.
- Pressurizer safety valves open.
- Core decay heat is 1.0 times the ANS standard value.
- Single failure occurs in the High Pressure Injection System.

Summary of Results

- Operator action within twenty minutes to initiate emergency feedwater will lower reactor coolant pressure and terminate loss of reactor coolant inventory, assuring adequate core cooling; or
- Operator action within twenty minutes to activate high pressure injection provides sufficient reactor coolant inventory to assure adequate core cooling.

TABLE 3

SMALL-BREAK LOCA WITH LOSS OF ALL FEEDWATER

Sequence of Events and Assumptions

- Small-break LOCA occurs.
- Reactor trip occurs on low reactor coolant pressure.
- Loss of offsite power and loss of main feedwater occur coincident with reactor trip.
- Emergency feedwater is not provided to steam generators.
- Core decay heat is 1.2 times the ANS standard value.
- Both high pressure injection trains function.

Summary of Results

- For break sizes greater than 0.01 ft^2 emergency core cooling is automatically initiated and no operator action is required to assure adequate core cooling.
- For break sizes equal to or less than 0.01 ft^2 the setpoint for initiation of high pressure injection is not reached. Operator action within twenty minutes to initiate emergency feedwater (which will subsequently result in high pressure injection) or to actuate high pressure injection will assure adequate core cooling.

TABLE 4

LOSS OF MAIN FEEDWATER WITH PORV FAILURE

Sequence of Events and Assumptions

- Loss of main feedwater occurs.
- Direct trip on loss of feedwater fails and reactor coolant pressure increases.
- Pressurizer PORV opens and does not close.
- Reactor trip occurs on high reactor coolant pressure.
- Emergency feedwater is provided to steam generators.
- Core decay heat is 1.2 times the ANS standard value.
- Single failure occurs in the High Pressure Injection System.

Summary of Results

- Automatic actuation of high pressure injection provides sufficient reactor coolant inventory to assure adequate core cooling.

TABLE 5

PORV FAILURES FOLLOWED BY LOSS OF ALL FEEDWATER

Sequence of Events and Assumptions

- o Pressurizer PORV fails open and does not close.
- o Reactor trip occurs on low reactor coolant pressure.
- o Loss of offsite power and loss of main feedwater occur coincident with reactor trip.
- o Emergency feedwater is not provided to steam generators.
- o Core decay heat is 1.0 times the ANS standard value.
- o Single failure occurs in the High Pressure Injection System.

Summary of Results

- o Automatic actuation of high pressure injection provides sufficient reactor coolant inventory to assure adequate core cooling.

TABLE 6

VERY SMALL LOCA WITH LOSS OF MAIN FEEDWATER

Sequence of Events and Assumptions

- Very small-break LOCA (0.005 - 0.01 ft^2) occurs.
- Reactor trips on low reactor coolant pressure.
- Loss of offsite power and loss of main feedwater occur coincident with reactor trip.
- Emergency feedwater is provided to steam generators.
- Core decay heat is 1.2 times the ANS standard value.
- Single failure occurs in the High Pressure Injection System.

Summary of Results

- Natural circulation initially removes core decay heat, then is interrupted as reactor coolant inventory decreases.
- Reactor coolant pressure increases when natural circulation is interrupted, then is stablized by steam condensation in the steam generators.
- Automatic actuation of high pressure injection provides sufficient reactor coolant inventory to assure adequate core cooling.

TABLE 7

SMALL-BREAK LOCA WITH DELAYED REACTOR COOLANT PUMP TRIP

Sequence of Events and Assumptions

- Small-break LOCA between 0.025 ft^2 and 0.2 ft^2 occurs.
- Reactor trips on low reactor coolant pressure.
- Reactor coolant pumps initially continue to operate, then are tripped at a later time during the accident.
- Core decay heat is 1.2 times the ANS standard value.
- Both high pressure injection trains function.

Summary of Results

- If the reactor coolant pumps continue to operate, adequate core cooling is assured.
- If the reactor coolant pumps trip after a high system void fraction is reached, adequate core cooling has not been demonstrated.
- If the reactor coolant pumps are tripped promptly upon automatic initiation of high pressure injection, adequate core cooling is assured.

TABLE 8

SMALL-BREAK LOCA WITH LOSS OF ALL FEEDWATER

AND SUBSEQUENT PORV FAILURE

Sequence of Events and Assumptions

- Very small-break LOCA (0.01 ft^2) occurs.
- Reactor trips on low reactor coolant pressure.
- Loss of offsite power and loss of main feedwater occur coincident with reactor trip.
- Emergency feedwater is not provided to steam generators.
- Core decay heat is 1.2 times the ANS standard value.
- Both high pressure injection trains function.
- Reactor Coolant System repressurization results in PORV opening and remaining open.

Summary of Results

- Operator action within twenty minutes to initiate emergency feedwater (which will subsequently result in high pressure injection) or to actuate high pressure injection provides sufficient reactor coolant inventory to assure adequate core cooling.

T. GARY BROUGHTON

Business Address:	GPU Service Corporation 100 Interpace Parkway Parsippany, New Jersey 07054
Education:	B.A., Mathematics, Dartmouth College, 1966.
Experience:	Control and Safety Analysis Manager, GPU Service Corporation, 1978 to present. Responsible for nuclear safety analysis and integrated thermal, hydraulic and control system analysis of nuclear and fossil plants. Supervised on-site technical support groups at Three Mile Island, Unit 2 during the post-accident period. Safety and Licensing Engineer; Safety and Licensing Manager, GPU Service Corporation, 1976 to 1978. Performed and supervised nuclear licensing, environmental licensing and safety analysis for Oyster Creek, Three Mile Island and Forked River plants. Served as Technical Secretary to Oyster Creek and Three Mile Island General Office Review Boards. Officer, U.S. Navy, 1966 to 1976. Trained at Naval Nuclear Power School, Prototype and Submarine School. Positions held include Nuclear Propulsion Plant Watch Supervisor, Instructor at D1G prototype plant and Engineering Officer aboard a fast-attack nuclear submarine.
Publications:	EPRI CCM-5, RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analyses of Complex Fluid Flow Systems, Volume 4: Applications, December, 1978, Section 6.1, "Analysis of Rapid Cooldown Transient - Three Mile Island Unit 2", with N.G. Trikouros and J. F. Harrison.

"The Use of RETRAN to Evaluate Alternate Accident Scenarios at TMI-2", with N. G. Trikouros. Proceedings of the ANS/ENS Topical Meeting on Thermal Reactor Safety, April 1980, CONF-800403.

"A Real-Time Method for Analyzing Nuclear Power Plant Transients", with P.S. Walsh. ANS Transactions, Volume 34 TANSAD 34 1-899 (1980).

ROBERT C. JONES, JR.

Business Address: Babcock & Wilcox Company
Nuclear Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

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.

(2)

LIC 10/28/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

LICENSEE'S TESTIMONY OF

ROBERT C. JONES, JR. AND T. GARY BROUGHTON

IN RESPONSE TO THE BOARD QUESTION ON UCS CONTENTION 8

OUTLINE

This testimon̈y ːpplements Licensee's Testimony of Robert C. Jones, Jr., and T. ᷾ary Broughton in Response to UCS Contention No. ᶠ ⁊nd ECNP Contention No. 1(e) (Additional LOCA Analysis), dated September 15, 1980. In particular, this testimony responds to the one aspect of the Board Question on UCS Contention 8 which was not addressed by the earlier testimony -- n̄mely, the recommendations made in NUREG-0565 and NUREG-0623.

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INTRODUCTION

This testimony, by Mr. Robert C. Jones, Jr., Supervisory Engineer, ECCS Analysis Unit, Babcock & Wilcox Company, and Mr. T. Gary Broughton, GPU Control and Safety Analysis Manager, is addressed to the following Board Question regarding UCS Contention 8:

BOARD QUESTION REGARDING UCS CONTENTION 8

> The board directs the staff and the licensee to present experts and the fundamental documents involved in the small break LOCA analysis, and to have very complete testimony on this subject. The recommendations of NUREG-0565 and NUREG-0623 should be addressed.
>
> It appears from the small break LOCA analysis that there is a large amount of reliance upon operator action and on non-safety grade equipment. The board wants that issue explored by testimony, including why such reliance is proper.

RESPONSE

BY WITNESSES JONES AND BROUGHTON:

Licensee's testimony in response to UCS Contention 8 addresses the small break loss of coolant accident (LOCA) analyses which have been performed to support the operation of TMI-1. The exhibits identified as items 3-13 in Licensee's Certificate of Service dated September 15, 1980, and provided to the parties pursuant thereto, present the fundamental results of these small break LOCA analyses.

The limited extent to which operator action and non-safety-grade equipment are utilized in the analyses for accident mitigation is discussed in the previously filed testimony (pages 3, 4, 8 and 9). Those discussions also address why such reliance is appropriate.

The following is a response to each of the recommendations (applicable to licensees) presented in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," and in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

NUREG-0565, RECOMMENDATION 2.1.2.a

> Provide a system which will assure that the block valve protects against a stuck-open PORV. This system will cause the block valve to close when RCS pressure has decreased to some value below the pressure at which the PORV should have reseated. This system should incorporate an override feature. Each licensee should perform a confirmatory test of the automatic block valve closure system.

<u>RESPONSE</u>

<u>BY WITNESS BROUGHTON</u>:

Design and installation of an automatic PORV block valve closure system is not being pursued at this time. The need for such a system has not been determined by appropriate analysis, which is called for by Item II.K.3.7 of NUREG-0660. Furthermore, it is not obvious that the addition of a closure system would be a modification which would provide greater safety, since the system may result in an increased probability of challenge to the pressurizer safety valves. Until the evaluations in response to Item II.K.3.7 are completed, the need to design and install an automatic block valve closure system has not been established.

NUREG-0565, RECOMMENDATION 2.1.2.b

> Most overpressure transients should not result in the PORV opening. Therefore, licensees should document that the PORV will open in less than five percent of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.

RESPONSE

BY WITNESS JONES:

Anticipated transients which produce an increase in Reactor Coolant System (RCS) pressure and which might cause the pressurizer power operated relief valve (PORV) to open include loss of feedwater, loss of external electrical load, turbine trip, uncontrolled control rod withdrawal from startup conditions, inadvertent closure of main steam isolation valves (MSIV's), and inadvertent moderator boron dilution. For any of these events the greatest potential for opening the PORV exists at the beginning of the fuel cycle when there is the minimum beneficial reactivity feedback. As the fuel cycle progresses, the moderator and Doppler negative reactivity feedback increases, thereby diminishing the magnitude of overpressurization. Also, as shown below, not every overpressurization event results in opening the PORV.

Overpressurization due to a loss of main feedwater, loss of electrical load or turbine trip will not cause the PORV to open because of the anticipatory trip functions which have been installed at TMI-1 and because of the increased PORV opening set pressure. This is the case at any time in the fuel cycle.

Safety analyses performed for TMI-1 (Final Safety Analysis Report) of the moderator dilution event at full power indicate peak system pressures lower than the present PORV opening setpoint. The lowered high pressure trip setpoint provides further assurance that the PORV will not open.

Inadvertent closure of the MSIV's does not result in a direct reactor trip and will result in an increase in primary system pressure. The most severe results from this event would involve closure of all MSIV's in a short time (a few seconds). At TMI-1, however, the MSIV closure time is about 2 minutes and inadvertent closure of the MSIV's is not expected to result in PORV actuation. Also, no inadvertent closure of all MSIV's has been experienced on a B&W plant to date.

Inadvertent control rod withdrawal from startup conditions can result in primary system overpressurization for a narrow range of small reactivity insertion rates. These are events which result in a relatively slow overpressurization requiring actuation of the high reactor coolant pressure trip rather than a high flux trip. The lowered high pressure trip setpoint and increased PORV opening setpoint, however, reduce the potential for PORV opening. Also, an event of this nature has not happened at a B&W plant to date.

In summary, there are some overpressurization events which can lead to PORV opening. Anticipated transients which have occurred, however, will not now result in PORV actuation due to the addition of anticipatory trip functions and the revision of the high pressure trip and PORV opening set points. Other, less frequent events which can currently result in PORV opening have not occurred to date at a B&W plant. Therefore, while no quantitative assessment of PORV opening has been performed for overpressurization events, it is readily apparent that this fraction is less than 5%.

NUREG-0565, RECOMMENDATION 2.1.2.c

All failures of PORVs to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

NUREG-0565, RECOMMENDATION 2.1.2.e

All failures of safety valves to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

RESPONSE

BY WITNESS BROUGHTON:

Licensee will propose changes to the TMI-1 Technical Specifications that will require reporting of failures or challenges to the PORV and safety valves as recommended.

NUREG-0565, RECOMMENDATION 2.1.2.d

> Licensees should submit a report to the NRC which discusses the safety valve failure rate experienced in B&W operating plants.

RESPONSE

BY WITNESS BROUGHTON:

Licensee is unaware of any instances of failures of Reactor Coolant System safety valves at any B&W plant. See Licensee's testimony in response to the Board Question on UCS Contention 6.

NUREG-0565, RECOMMENDATION 2.2.2.a

> The analysis methods used for small break LOCA analysis by B&W should be revised, documented, and submitted for NRC approval.

NUREG-0565, RECOMMENDATION 2.2.2.b

> Plant-specific calculations using the NRC approved model for small breaks should be submitted by all licensees to show compliance with 10 CFR 50.46.

RESPONSE

BY WITNESS JONES:

The small break LOCA analyses which were performed after the TMI-2 accident were done to provide an improved analytical basis for emergency procedures for small break LOCA's. These analyses were not performed to demonstrate compliance with 10 CFR 50.46. NUREG-0565 states that the post-TMI-2 analyses are beyond those normally considered in small break analyses and that the NRC Staff has some concerns relative to the use of the currently approved small break model for these purposes. However, NUREG-0565 (Section 2.2.1) also contains the following conclusion: "The small break analysis methods used by B&W are satisfactory for the purpose of predicting trends in plant behavior following small break LOCAs and for training of reactor operators." NUREG-0565 does not state that the approved B&W small break evaluation is difficient for demonstrating compliance for TMI-1 with respect to 10 CFR 50.46 and Appendix K. While further code development may be performed and model modifications may be made, the changes are not expected to result in a substantial change to the Appendix K evaluations performed for TMI-1.

NUREG-0565, RECOMMENDATION 2.2.2.c

> The effects of core flood tank injection on small break LOCAs should be further investigated to determine the amount of condensation realistically expected and to determine its effect on heatup and core uncovering. The condensation model and modeling procedures (i.e., injection location used in the computer analyses) require further investigation to assure that the effects of CFT injection are biased in a conservative manner. Semiscale and LOFT test data should be used to verify the models.

RESPONSE

BY WITNESS JONES:

This Staff concern relates to the potential for a large underprediction of system pressure, due to the analytical assumption of instantaneous steam condensation on the cold Core Flood Tank (CFT) water delivered to the RCS during a small break. Contrary to this concern, the small break analyses performed for TMI-1 do not predict large pressure oscillations caused by core flood injection. Thus, while further examination of this phenomena may be performed, the small break predictions are not expected to be substantially altered.

NUREG-0565, RECOMMENDATION 2.3.2.a

Tripping of the RCPs in the event of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break problem, for example, an increase in the HPI flow rate. In the interim, until a better solution is found, the RCPs should be tripped automatically in the case of a small break LOCA. The signals designated to initiate the RCP trip should be carefully selected in order to differentiate between a small break LOCA and other events which do not require the RCPs to be tripped.

NUREG-0623, CONCLUSION 6.0(4)

From items (2) and (3), above, we find that tripping all of the reactor coolant pumps during small break LOCAs is required at this time, and that this pump trip should be automatically initiated from equipment that is safety-grade to the extent possible.

NUREG-0623, CONCLUSION 6.0(5)

The impact of an early pump trip on non-LOCA transients is not predicted to lead to unacceptable consequences. However, tripping the reactor coolant pumps for non-LOCA transients can aggravate the consequences of these transients and extend the time required to bring the plant into controlled shutdown condition. For B&W plants, tripping of the reactor coolant pumps during severe overcooling events increases the potential for interruption of natural circulation due to steam formation in the coolant loops.

Therefore, we conclude that the criteria and requirements for reactor coolant pump trip to be established from item (4), above, should minimize, to the extent practicable, the probability of initiating a reactor coolant pump trip for non-LOCA transients.

NUREG-0623, CONCLUSION 6.0(6)

The staff recognizes the potential desirability of running the reactor coolant pumps to provide forced circulation during small break LOCAs and we encourage the continued exploration by the industry of means by which this could be accomplished. For example, an increase in HPI capacity or two-pump operation as proposed by Combustion Engineering are a step in this direction.

RESPONSE

BY WITNESS BROUGHTON:

The TMI-1 Restart Report, Supplement 1, Part 3, response to question 11, presents the design characteristics of our proposed reactor coolant pump trip system. This system is based on a coincident loss of sub-cooling margin and high pressure injection actuation. The NRC Staff has accepted this approach as described in NUREG-0680 (SER at p. C2-18).

NUREG-0565, RECOMMENDATION 2.3.2.b

> The B&W small break LOCA analyses rely on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features. The equipment used to provide the necessary RCP trip, the pressurizer PORV and PORV block valve, and equipment used to actuate the PORV and PORV block valve fall into this category. The reliability and redundancy of these systems should be reviewed and upgraded, if needed, to comply with the requirement of Section 9 of NUREG-0585, regarding the interaction of non-safety and safety-grade system.

RESPONSE

BY WITNESS JONES:

The equipment used in the post TMI-2 accident small break LOCA analyses (the analyses addressed in NUREG-0565) which is not part of the Reactor Protection System or part of the engineered safety features is identified in Licensee's testimony in response to UCS Contention 8 and ECNP Contention 1 (Additional LOCA Analysis) (pages 3, 4, 8 and 9).

The specific items utilized in the analyses are the Emergency Feedwater System and the equipment used to provide reactor coolant pump trip. The pressurizer power operated relief valve (PORV) and PORV block valve have not been relied upon in the LOCA analyses.

NUREG-0565, RECOMMENDATION 2.3.2.c

Plant simulators used for operator training should offer, as a minimum, the following small break LOCA events:

(1) continuous depressurization;

(2) pressure stabilized at a value close to secondary system pressure;

(3) repressurization;

(4) stuck-open PORV; and

(5) stuck-open letdown valve.

Each of these cases should be simulated with RCPs running as well as tripped. The first three events should be simulated for both cold and hot leg breaks. In addition to the usual assumed single failures in the ECCS and feedwater systems, complete loss of feedwater should also be simulated in conjunction with the above events. It is important that training programs also expose the operators to various kinds of system transients on inadequate core cooling as discussed in Section 2.1.9 of NUREG-0578.

RESPONSE

BY WITNESS BROUGHTON:

Operator training, including the use of simulators, will be addressed in Licensee's testimony on management competence.

NUREG-0565, RECOMMENDATION 2.6.2.a

> The various modes of two-phase natural circulation, which are expected to play a significant role in plant response following a small break LOCA, should be demonstrated experimentally. In addition, the staff requires that the licensees provide verification of their analysis models to predict two-phase natural circulation by comparison of the analytical model results to appropriate integral systems tests.

RESPONSE

BY WITNESS JONES:

The B&W small break LOCA evaluation model includes appropriate consideration of the mechanisms responsible for natural circulation. The computer code utilized models both density changes and flow losses under single- and two-phase fluid conditions. Thus, the evaluation model should reasonably predict the various modes of two-phase natural circulation. Additionally, for small break LOCA's, the steam generators do not have an important influence on the transient except for those cases where the break size is insufficient to discharge energy at least equal to that added by the core decay heat. As noted in Licensee's testimony in response to UCS Contentions 1 and 2 (Natural and Forced Circulation) (pages 6 and 7), this break size would be less than approximately 0.02 ft^2. Breaks smaller than 0.02 ft^2 will retain substantially more system inventory than the design basis small break, which is approximately 0.07 ft^2, and have large margins relative to the potential for core uncovery. Therefore, while further examination of two-phase natural circulation phenomena may be performed, TMI-1 is still expected to conform to 10 CFR 50.46.

NUREG-0565, RECOMMENDATION 2.6.2.b

Appropriate means, including additional instrumentation, if necessary, should be provided in the control room to facilitate checking whether natural circulation has been established.

RESPONSE

BY WITNESS BROUGHTON:

Checks that natural circulation has been established are included in appropriate plant procedures and require observing primary system hot and cold leg temperatures for a constant differential an observing that cold leg temperature approaches secondary system saturation temperature. The instrumentation used in this determination are located in the control room.

NUREG-0565, RECOMMENDATION 2.6.2.c

Licensees should provide an analysis which shows the plant response to a small break which is isolated and the PORV fails-open upon repressurization of the reactor coolant system to the PORV setpoint.

RESPONSE

BY WITNESS JONES:

A specific analysis providing the plant response to a small break which is isolated and the PORV fails-open upon repressurization of the RCS to the PORV setpoint has not been performed. However, based on the analyses discussed in Licensee's testimony in response to UCS Contention 8 and ECNP Contention 1(e) (Additional LOCA Analysis), the response to this event can be described.

Initially, as a result of the small break, the system will depressurize. Actuation of the High Pressure Injection system (HPI) will automatically occur, assuming feedwater availability, prior to the loss of natural circulation. Should break isolation occur after natural circulation is lost and prior to the establishment of the boiler-condenser mode of steam generator heat removal, system repressurization would occur. Assuming that the repressurization reaches the PORV setpoint and that the PORV subsequently sticks open, a transient very similar to that calculated for a PORV initially stuck open would then occur. Adequate core cooling would be continuously maintained for this transient by the fluid provided by HPI.

NUREG-0565, RECOMMENDATION 2.6.2.d

Licensees should provide an analysis which shows the plant response to a small break in the pressurizer spray line with a failure of the spray isolation valve to close.

RESPONSE

BY WITNESS JONES:

A break in the pressurizer spray line along with a failure of the spray isolation valve to close results in inventory loss from both the RCS cold leg and the top of the pressurizer. The leak rates from the cold leg would be limited by the area of the spray line, 0.025 ft^2, and from the pressurizer the leak rate would be limited by the flow area of the spray nozzle in the pressurizer, 0.072 ft^2. The small break LOCA analyses performed for TMI-1 to demonstrate conformance to 10 CFR 50.46 envelope the total leak flow area for this case. Thus, system inventory losses similar to that which would occur for this scenario have already been considered in the LOCA analyses. However, for this accident, liquid inventory would remain in the pressurizer while the TMI-1 small break analyses empty the pressurizer. The effect of the stored inventory in the pressurizer for this event is expected to be offset by the increased availability of HPI for core cooling. In the analyses performed for TMI-1, less than 70% of the HPI was calculated to enter the core due to the direct bypass of the injected fluid out the break, which was assumed to be located in the bottom of the cold leg pump discharge piping between the HPI nozzle and the reactor vessel. For the spray line break, no HPI fluid would bypass out the break without first entering the vessel. The increased HPI flow for the spray line break would establish long term cooling earlier, relative to an equivalently sized pump discharge break, and is expected to offset the effect of the stored inventory in the pressurizer. Therefore, an analysis of this accident is not expected to provide results which are in excess of 10 CFR 50.46 limits.

NUREG-0565, RECOMMENDATION 2.6.2.e

Licensees should provide confirmatory information to show that HPI and CFT flows during small breaks are insufficient to form water slugs, or if they do, to show that the structural design bases of the primary system includes loads due to:

1. water slug intertial motion;
2. water slug impact; and
3. pressure oscillation due to steam condensation

RESPONSE

BY WITNESS JONES:

During small breaks, water slugs are not expected to be formed as a result of HPI and CFT flows. The HPI flows would be less than 140 ft^3/min during a small break transient. Since the piping volume from the HPI nozzle to the reactor vessel is 280 ft^3, it would take two minutes to fill the pipe. Also, the reactor vessel internals vent valves will continuously equalize pressures throughout the primary system. Therefore, the HPI water will drain into the vessel and there is no mechanism available to hold the HPI water in the cold leg pipe. Thus, slug flow as a result of the HPI will not occur.

The water injected from the CFT's also is not expected to produce slug flow since the fluid is directly injected into the reactor vessel downcomer. Also, the internals vent valves minimize pressure gradients within the vessel such that no holdup of injected CFT water will occur. Thus, no water slugs will occur as a result of CFT injection.

NUREG-0565, RECOMMENDATION 2.6.2.f

> Licensees should provide an analysis of the possibility and impact of RCP seal damage and leakage due to loss of seal cooling on loss of offsite power. If damage cannot be precluded, licensees should provide an analysis of the limiting small break LOCA with subsequent RCP seal failure.

RESPONSE

BY WITNESS BROUGHTON:

This recommendation was addressed in Licensee's response to R.W. Reid's letter of November 21, 1979, which was provided by letter No. TLL-285, dated June 30, 1980. In this response, a description of the RCP seal system and its cooling was provided along with a discussion of the probable degradation mechanism, the time and methods available to restore seal cooling, and the result of loss of cooling for up to 60 minutes. The results of that analysis did not indicate that excessive seal leakage would occur within 60 minutes.

NUREG-0565, RECOMMENDATION 2.6.2.g

Licensees shall provide pretest predictions of LOFT Test L3-6 (Reactor Coolant Pumps Running).

NUREG-0623, CONCLUSION 6.0(7)

We will require verification of small break models with the pumps running against appropriate integral systems experimental tests. In particular, we will require that the PWR vendors and fuel suppliers perform pretest predictions of the LOFT SBLOCA test with pumps running scheduled to be performed in March of 1980.

RESPONSE

BY WITNESS BROUGHTON:

GPU is a participant in the B&W owners' group program to predict LOFT L3-6. This analysis will be performed by B&W and provided to the NRC.

NUREG-0565, RECOMMENDATION 2.6.2.h

With regard to the effects of noncondensible gases during a small break LOCA, the licensees should provide the following information:

1. The technical justification for omitting the radiolytic decomposition of injected ECC water as a source of noncondensible gas; and

2. Confirmatory information to verify the predicted condensation heat transfer degradation in the presence of noncondensible gases.

RESPONSE

BY WITNESS JONES:

Analyses of the effect of noncondensibles on the condensation heat transfer process in the steam generator during a small break LOCA have been performed. These analyses, which included the effects of radiolytic decomposition, determined that sufficient condensation surface would remain within the steam generator and that the boiler-condenser mode would not be prohibited. Additionally, even under a postulated condition that the noncondensible gases prohibited condensation, HPI can be operated in a feed and bleed mode to supply adequate core cooling - see Licensee's testimony in response to UCS Contentions 1 and 2 (Natural and Forced Circulation). Thus, while further examination of the effect of noncondensibles on the condensing heat transfer process within the steam generator may be performed, provisions are available at TMI-1 to assure adequate core cooling.

NUREG-0565, RECOMMENDATION 2.6.2.i

By use of analysis and/or experiment, address the mechanical effects of induced slug flow on steam generator tubes.

RESPONSE

BY WITNESS JONES:

Analysis of the effect of induced slug flow on the steam generator has been performed. The analysis assumed that a sudden front of water impacted the tube sheet with a flow equivalent to that of normal operation. It was assumed that this load was suddenly applied and that the entire load was absorbed by the tubes directly under the inlet nozzle of the steam generator. The loading on a steam generator tube was calculated to be 21.5 lbf, in comparison to the theoretical buckling load of approximately 700 lbf. Thus, induced slug flow will not affect the integrity of the steam generator tubes.

T. GARY BROUGHTON

Business Address: GPU Service Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054

Education: B.A., Mathematics, Dartmouth College, 1966.

Experience: Control and Safety Analysis Manager, GPU Service Corporation, 1978 to present. Responsible for nuclear safety analysis and integrated thermal, hydraulic and control system analysis of nuclear and fossil plants. Supervised on-site technical support groups at Three Mile Island, Unit 2 during the post-accident period.

Safety and Licensing Engineer; Safety and Licensing Manager, GPU Service Corporation, 1976 to 1978. Performed and supervised nuclear licensing, environmental licensing and safety analysis for Oyster Creek, Three Mile Island and Forked River plants. Served as Technical Secretary to Oyster Creek and Three Mile Island General Office Review Boards.

Officer, U.S. Navy, 1966 to 1976. Trained at Naval Nuclear Power School, Prototype and Submarine School. Positions held include Nuclear Propulsion Plant Watch Supervisor, Instructor at D1G prototype plant and Engineering Officer aboard a fast-attack nuclear submarine.

Publications: EPRI CCM-5, RETRAN - A Program for One-Dimensional Transient Thermal-Hydraulic Analyses of Complex Fluid Flow Systems, Volume 4: Applications, December, 1978, Section 6.1, "Analysis of Rapid Cooldown Transient - Three Mile Island Unit 2", with N.G. Trikouros and J. F. Harrison.

"The Use of RETRAN to Evaluate Alternate Accident Scenarios at TMI-2", with N. G. Trikouros. Proceedings of the ANS/ENS Topical Meeting on Thermal Reactor Safety, April 1980, CONF-800403.

"A Real-Time Method for Analyzing Nuclear Power Plant Transients", with P.S. Walsh. ANS Transactions, Volume 34 TANSAD 34 1-899 (1980).

ROBERT C. JONES, JR.

Business Address:	Babcock & Wilcox Company Nuclear Power Generation Division P.O. Box 1260 Lynchburg, Virginia 24505
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.

MR. BAXTER: Mr. Chairman, the second document we just received into evidence responds to only one aspect of the Board's question regarding UCS Contention Number 8, which was withdrawn by the Union of Concerned Scientists and picked up by the Board, and that is the Regulations NUREG 0565 and NUREG 0623. It is our view that the remainder of that Board question is addressed in the September 15 filing.

Now, the Board question regarding UCS Contention Number 8 also directed licensee to present the fundamental documents involved in the small break LOCA analysis, and on September 15, we attempted to serve or we did serve those documents crucial to the small brea' LOCA analysis, and I would now like to undertake to have each of these documents identified as exhibits.

(Pause.)

MR. BAXTER: I would like to have first marked for identification Licensee's Exhibit Number 3, the document entitled Topical Report BAW-10103A, Revision 3, entitled, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, dated July, 1977.

(The document referred to was marked for iden^ification as Licensee Exhibit Number 3.)

MR. BAXTER: As Licensee's Exhibit Number 4, I would like to have marked for identification a document which is a letter dated July 18, 1978, from J. H. Taylor of Babcock and Wilcox to S. A. Varga, of the NRC, with attached Additional ECCS Small Break Analysis.

(The document referred to was marked for identification as Licensee Exhibit Number 4.)

MR. BAXTER: As Licensee's Exhibit Number 5, document entitled Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant, Volume I, Section 6, "Small Break Analyses," May 7, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 5.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 6 a document entitled Small Break in the Pressurizer (PORV) With No Auxiliary Feedwater and Single Failure of the ECCS, Supplement 1 to the May 7, 1979, Small Break Analysis, dated May 12, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 6.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 7 a document entitled Small Break in the Pressurizer (PORV) with no Auxiliary Feedwater and Single Failure of the ECCS, with Realistic Decay Heat, Supplement 2 to the May 7, 1979, Small Break Analyses, dated May 12, 1979.

(The document referred to was marked for identification as LIcensee Exhibit Number 7.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 8 a document entilted Auxiliary Feedwater Flow Required for LOCA, Supplement 3 to the May 7, 1979, Small Break Analyses, and dated May 24, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 8.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 9 B&W Document 86-1103585-00, entitled System Response to Total Loss of SG Heat Synch, dated August 7, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 9.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 10 a report entitled Analysis Summary in Support of an Early RC Pump

Trip, dated August 21, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 10.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 11 a document entitled Supplemental Small Break Analysis, Supplement to the August 21, 1979, Analysis in Support of An Early RC Pump Trip, and dated September 2, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 11.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 12 B&W Document 69-1106001-00, entitled Small Break Operating Guidelines and dated November, 1979.

(The document referred to was marked for identification as Licensee Exhibit Number 12.)

MR. BAXTER: I would like marked for identification as Licensee's Exhibit Number 13 B&W document 86-1117679-000, entitled Small Break With Failed PORV, dated February 11, 1980.

(The document referred to was marked for identification as

Licensee Exhibit Number 13.)

MR. BAXTER: We have provided the Board and the parties with our September 15 filing, an outline which summarizes briefly or identifies in more description what these additional LOCA analysis exhibits include, but I would like to rely on Mr. Jones briefly to amplify or explain for the record right at this point where we have identified them, what these documents are, without summarizing where they are.

CHAIRMAN SMITH: Would it also be appropriate to place the outline into the transcript at this point?

MR. BAXTER: Yes, that would be helpful, I believe, and we can provide the reporter with a copy.

CHAIRMAN SMITH: If you will make the outline available to the reporter, we will have that bound into the transcript at this point.

(The material referred to follows:)

(3)

OUTLINE
ADDITIONAL LOCA ANALYSIS EXHIBITS

The exhibits listed as Items 3 through 13 on Licensee's Certificate of Service, September 15, 1980, are submitted in response to the Board Question regarding UCS Contention 8, and provide a documentary history of the small break LOCA analyses performed by the Babcock & Wilcox Company which are applicable to TMI-1. The results of these analyses are presented in Licensee's Testimony of Robert C. Jones, Jr., and T. Gary Broughton in response to UCS Contention No. 8 and ECNP Contention No. 1(e) (Additional LOCA Analysis).

Report BAW-10103A, Rev. 3 (Item 3 on Licensee's Certificate of Service), and the July 18, 1978 supplemental analysis (Item 4) constitute a complete spectrum of small break analyses which show conformance to the requirements 10 CFR 50.46 and Appendix K to 10 CFR Part 50.

Section 6 of the May 7, 1979 report, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" (Item 5), Supplements 1, 2 and 3 (Items 6, 7 and 8) thereto, and B&W Documents 86-1103585-00 (System Response to Total Loss of SG Heat Sink,) (Item 9) and 86-1117679-000 (Small Break with failed PORV) (Item 13) consist of additional analyses of plant response to various small break scenarios, which were performed in response to specific NRC requests following the TMI-2 accident. The results of these analyses demonstrate that, with appropriate operator action, the emergency core cooling system is capable of controlling the consequences of these scenarios. B&W Document 69-1106001-00, "Small Break Operating Guidelines" (Item 12), provides guidance for operator action based upon the results of the small break analyses.

The evaluations contained in the B&W report, "Analysis Summary in Support of an Early RC Pump Trip," (Item 10) and its "Supplemental Small Break Analysis," (Item 11) were performed pursuant to NRC IE Bulletin 79-05C. These evaluations demonstrate that, under highly voided reaction coolant conditions, a delayed trip of the reactor coolant pumps will result in unacceptable consequences when Appendix K evaluation techniques are used. The analysis further shows that the prompt reactor coolant pump trip upon receipt of a low pressure ESFAS signal (as required by these results) will provide acceptable LOCA consequences.

MR. BAXTER: With the outline in the record, I don't believe it would be necessary unless the Board feels further amplification is necessary for Mr. Jones to explain further what the documents are.

CHAIRMAN SMITH: While Dr. Jordan is reviewing the outline for a moment, I want to observe that the testimony scheduled for today was UCS Contention 8, which was also the surviving form of ECNP Contention 1E, and there is no representative from ECNP present today, nor any other intervenor, as far as that is concerned.

CHAIRMAN SMITH: The outline is adequate for your suggested purpose.

MR. BAXTER: The witnesses are available for cross examination.

DR. JORDAN: I would like to suggest this morning that instead of going immediately into the cross examination, that it would save time, I believe, if the witnesses would summarize their testimony and particularly address the tables in the back of the first document, dated 9/15/80.

I realize this would be moderately time consuming, but I think it would be more straightforward to do it now, and I may have questions -- in fact, I do have questions on understanding the tables, and I think it would help to clear them up early rather than wait until the cross examination starts. If anyone else has differences of opinion, I would be glad to hear them.

MR. BAXTER: I would be happy to ask the witnesses to do that. My only slight puzzlement is that we attempted to make the tables summaries themselves. I am not quite sure what your questions are on the tables.

DR. JORDAN: Well, I do need to understand just what goes on in each case, what are the actions, both of the equipment, the automatic actions, the response of the systems, and the operator response with respect to each one of the scenarios. Some of them may go very rapidly, and I would hope that they can, but nevertheless, for my understanding, I believe I need it.

MR. BAXTER: Fine. Let's try it with respect to the September 15 filing first.

DR. JORDAN: That is what I meant.

MR. BAXTER: Not the October 21. That is a series of little ones.

DR. JORDAN: Not the 21. The tables in the back of there is a good summary, as you pointed out, of the scenario,with response to each one of the transients, and I think it would be helpful to hear that.

MS. WEISS: When you refer to October 21st, do you mean the October 28th filing?

MR. BAXTER: Yes, I did. I am sorry.

BY MR. BAXTER: (Resuming)

Q Mr. Jones, do you understand the Board's inquiry, and could you summarize the processes and phenomena that are taking place in the plant for each of those scenarios, including any operator actions that are going on?

DR. JORDAN: Mr. Jones, is that to big an order on short notice? Would you like a little time?

WITNESS JONES: I will give it a try.

DR. JORDAN: I rather suspect that you are unable to do it, because from my experience yesterday, you seem to be on top of that.

WITNESS JONES: It is a fairly extensive set of analyses here, so it is going to take some time.

DR. JORDAN: I realize it will. I realize it will. But I think it is worthwhile.

WITNESS JONES: Let me --

DR. JORDAN: We may indeed spend the rest of the morning on this, if necessary, because I won't understand it completely on the first go-around, but I will ask questions as you go.

MS. WEISS: Before you even start, since we've got this funny noise going upstairs, I will have to ask you to talk really loudly.

MR. BAXTER: Mr. Jones, you understand that the Board is referring to the tables attached at the back of the testimony.

WITNESS JONES: Yes, I do, and I guess we can start with Table 1, which does not provide a summary of what is happening or of the results for the pre-TMI 2 LOCA analyses, and I guess I would like to start there.

The first case on the list from Topical Report BAW-10103 is the core flood tank line break. The analysis assumes that the line, the core flooding line attached to the reactor vessel breaks one of them.

DR. JORDAN: Is that break ahead of the check valve?

WITNESS JONES: It is between the check valve and the vessel.

DR. JORDAN: Yes.

WITNESS JONES: And the single failure assumption that is typically utilized, we assume that one of the low pressure injection trains is lost from the loss of the diesel, one of the high pressure injection trains is lost because of the loss of the diesel, and the other available low-pressure injection pump is assumed to be discharging into the broken core flooding line, so that to mitigate this transient, you have only one core flood tank and one high pressure injection pump.

DR. JORDAN: I see.

Does the core flood tank line break? Does that present unique problems that you don't get with all of the other coolant pump discharge breaks?

WITNESS JONES: Well, it does in the sense that you lose one of your core flooding tanks, and all of your low pressure injection system for that case.

DR. JORDAN: I see.

WITNESS JONES: So you have a unique problem in the availability of the core coolant system equipment. The basic transient for the case is quite simple. It is a very rapid depressurization, and I think to help explain these I am going to have trouble doing it totally verbally, and I would like to refer you with the exhibits to discuss what happens during the transient. I believe that would be easier.

MS. WEISS: Can we take two minutes to go get ours? We didn't bring them. To get our exhibits, copies of them?

(Pause.)

MS. WEISS: We are ready.

MR. BAXTER: Mr. Chairman, I also distributed to the partaies during the break a schematic diagram which is entitled Simplified Schematic Diagram of Engineered Safeguards System for Core and Building Protection, Three Mile Island Nuclear Station Unit 1, which I would ask to have identified as Licensee's Exhibit Number 14.

DR. JORDAN: I notice this is Figure 6.1. From what document?

MR. BAXTER: The final safety analysis report for Unit 1.

(The document referred to was marked for identification as Exhibit Number 14)

BY MR. BAXTER: (Resuming)

Q Mr. Jones, would you proceed, and perhaps explain briefly what Licensee's Exhibit 14 represents?

A (WITNESS JONES) Yes, to try to help in the explanation of the emergency core cooling systems, how they are utilized in the analyses, Exhibit 14 will help provide a base for understanding. I would like to briefly explain the emergency core cooling systems which are provided at Unit 1 before I continue on with the rest of my explanation of what happened during all of the analyses we have performed.

As noted on the figure, there are two core flooding tanks inside the reactor building which discharge directly into the reactor vessel. These tanks contain roughly 1,000 cubic feet of water, and have a nitrogen overpressure of 650 psi.

When the primary system drops below that pressure, the check valves would swing open, and the water would discharge into the vessel at a rate dependent upon the line losses and the Delta P between the tank and the vessel.

Additionally, there are two sets of pumped injection systems provided on the plan. These are the high pressure injection pumps and the low pressure injection pumps. The high pressure injection pumps are outlined in yellow on the diagram, and are the basic system used to mitigate small break LOCA's.

There are basically two trains that are normally utilized as high pressure injection systems which can be actuated either manually or automatically by the engineered safety features actuation system. This diagram shows each pump split into two lines, and is incorrect in that representation. This was the representation of the TMI 1 system prior to the modifications being made at this time.

Upon restart, the TMI 1 HPI pump, each pump will be capable of feeding all four injection lines. They will be cross connected to each other, and each of these injection lines discharge into one of the cold legs in the primary system loops.

The low pressure injection pumps are called on the diagram DH pumps, decay heat pumps. They are the normal -- they are used at low pressures to provide large flow rates to the vessel, mainly for large break LOCA's.

The comparison between the two pumps as far as their capacities, the high pressure injection pumps provide roughly 500 gpi each at a system pressure of approximately 600 psig in the reactor vessel. The low pressure injection pumps provide roughly 3,000 gpm at a system pressure of 100 psi. The high pressure injection pumps provide fluid over a pressure range from roughly 2,700 psi down to essentially atmospheric.

The low pressure injection pumps will provide flow only after the primary system pressure has dropped below approximately 200 psi. The line-up of the low pressure injection pumps is to connect into the core flooding line as shown on this figure between the two check valves in the line. It does not inject into the cold leg, but rather injects into the core flooding line, which then connects into the reactor vessel for direct injection into the reactor vessel.

Initially, in the transient, the pumps will draw their suction off of the borated water storage tank, which contains roughly 300,000 gallons. Following -- the word isn't depletion, but when the tank reaches a low level, has discharged a substantial amount of its inventory, operator action is utilized to line up the low pressure injection pumps to the reactor building sump, and as outlined in this diagram, the blue line coming around connecting up into the suction of the high pressure injection pumps is the line-up which is called the piggy-back operation, where the low pressure injection pumps provide suction to the high pressure injection pumps.

That is basically how the system lines up and works.

MS. WEISS: Mr. Chairman, Mr. Pollard had a question about one thing that was just said in description of the diagram. Would it be appropriate to ask that now?

DR. JORDAN: Yes, I think anything that will clear up the explanation will be helpful at this time.

MR. POLLARD: To hopefully reduce the number of times I will have to interrupt on the explanation of the tables, perhaps I could just ask you a general question first.

DR. JORDAN: Exactly the reason for doing it.

VOIR DIRE EXAMINATION

BY MR. POLLARD:

Q Mr. Jones, in your explanation today of Exhibit 14 and your subsequent explanations, you will be talking in general and when you give a very specific number such as the cutoff header for the pump or the minimum pressure needed for the pump to operate, are we to understand that this is a general figure, or are you talking that you know specifically the values for the actual pumps at Three Mile Island Unit 1?

A (WITNESS JONES) The values I quoted for the high pressure injection pumps are typical of the TMI 1 pumps.

Q Do you know that the high pressure injection pumps can run at full flow when the pressure in the reactor coolan_ system is near atmospheric without either cavitating or running out?

MR. BAXTER: Excuse me, Mr. Chairman. It is not clear to me that this is by way of furthering the explanation as opposed to cross examination.

MR. POLLARD: All I am trying to do, Mr. Jordan, is to know later on in this hearing whether or not I have to rely upon the numbers that Mr. Jones is now giving with respect to the characteristics of the specific equipment in Three Mile Island Unit 1, or if these are just general figures for perhaps any B&W plant. I won't need to interrupt any further.

DR. JORDAN: Go ahead and answer. Are they meant to be specific to TMI 1 or not? I think maybe that will clear it up.

WITNESS JONES: Well, the numbers I am quoting are basically our general design requirements for these systems. I cannot attest to their absolute accuracy, which is why I kept saying generally or about, because I am not sure. I do know that the high pressure injection pumps for TMI 1 can pump two high system pressures, roughly 2,700 psi. It may be 29. I am not sure. But I know it can pump up to about that value.

I also know from the calculations that I have seen that an individual high pressure injection pump is capable of providing at least 500 gpm at 600 psi, and with the installation of the cavitating venturies, from what I have seen, the characteristics or the design that they are trying to set up is such that the cavitating venturies will be sized such that the pumps will not cavitate at low system pressures. They will not run out and destroy themselves.

DR. JORDAN: I think perhaps that answers your question, but I think that that type of question, for guidance to you, Mr. Pollard, was a little too specific. It would be just as well to reserve those until cross examination.

MR. POLLARD: Yes, sir. I understanding. This was just an example of the question, and I will not interrupt further.

DR. JORDAN: Fine. Now, whenever there is a matter of not understanding what he is saying, please interrupt.

WITNESS JONES: All right. Now, I believe we were last at a description of the phenomena that occurs during a core flooding tank line break. And I think now that we have introduced this exhibit, you can see where the single failure assumption can lead to for this specific case, only one high pressure injection pump being available, and one core flooding tank being available for mitigating the transient.

Now, basically, the system response for this case is a very rapid depressurization, and if you look at Appendix C of Exhibit 3, which is the small break analyses that were performed in the top of the report, and specifically Figure C-4, you will see the system --

CHAIRMAN SMITH: I see that people have not caught up with you yet.

(Pause.)

DR. JORDAN: We have it now.

WITNESS JONES: All right. You will see that the system undergoes a very rapid depressurization transient which results in the emergency safety features actuation signal being reached within the first roughly ten seconds. That signal is set at about 1600 psi. And it is very quickly reached in this accident, and will start the diesel open valves, start the high pressure injection pumps, actuating the emergency core cooling systems.

At approximately 150 seconds, you can see that the core flood tanks are actuated or the core flood than in this case, and this system continues to depressurize and stabilizes in a long-term mode at approximately 100 psi.

Now, for this transient and basically for all of the transients I will be discussing until I get to Table 8 -- no, excuse me, Table 7 -- the assumptions used in the analysis was a loss of off-site power at the time of reactor trip. And for this case, I did not mention this, the reactor trip occurs subsequent to the emergency safety features actuation signal. I mean, prior to. It is at about 1900 psi. That will trip the reactor.

That loss of off-site power results in the loss of the pumping of the reactor coolant pumps, and they coast down, and an indication of the system flow for this transient is provided in Figure C-3, the page just prior to the one we were just looking at. And as can be seen, the system flow rate rapidly decays for this transient.

There is no operator actions required for this analysis. The system has a very rapid depressurization, actuates all the equipment necessary, and there is no early operator actions required for this transient. In the long term, when the BWST is emptied, he does have to make a manual switchover to the sump and line up the HPI system in a piggyback mode.

Additionally, it is possible or it may be possible for the operator to open up the cross connects between the decay heat pumps such that these are indicated in Exhibit 14 as the valves between -- I believe the valves -- these would be the valves between the decay heat coolers, but basically he would open up a cross connect such that one pump could feed both lines and he could throttle the LPI flow to balance flows, thus providing additional emergency core cooling system flow to the system.

That action is not, however, necessary to assure adequate core cooling. That is an action he can take in the long term, and if he takes that action and balances the flows in each line to where he gets roughly 1,000 GPM in each leg, he can terminate the high pressure injection flow per the throttling criteria that he has.

As far as the consequences of this accident, figure C-7 shows the inner vessel fluid inventory for this accident. The inner vessel, as utilized in this evaluation, is the core in the upper plenum, and as you can see, the liquid level in the core does drop until approximately the time the core flooding tanks come on, until the one core tank comes on which slowly recovers the core from the liquid inventory standpoint, but it shown up around the 18 to 20 foot height on this graph. There is another graph labeled inner vessel mixture height. This is a two-phase mixture which would be within the inner vessel, and as seen from this case, the core which is at 12 feet -- it is labeled Top of Active Core. The core remains totally covered throughout the transient.

DR. JORDAN: With the two-phase mixture?

WITNESS JONES: With the two-phase mixture, and that will provide adequate core cooling, and the temperature response for the cladding for this case, which is not shown in this diagram, will just decay with time, and basically stay after the initial discharge of the energy in the fuel rods which occurs during flow coast down of the pumps that will stay within a few degrees, five, maybe ten degrees of the saturated fluid temperature within the core.

The second case that is covered in this topical I don't feel needs a lot of discussion.

DR. JORDAN: No, just if there are any differences now.

WITNESS JONES: The half a square foot break and the pump discharge break pipe is basically the same typical response, the rapid depressurization, the difference being you would then have two core flooding tanks available along with one LPI pump, and with the location of this break, a portion of the high pressure injection fluid would be lost, and I will discuss that in more detail when we talk about the next set of analyses.

The .04 square foot in the reactor coolant pump suction is a little more interesting. But rather than belabor that analysis, because it really is very similar to one which is shown in the next report, I would just rather move on to the next figure.

The only significant actor in that break, however, it is a break in the suction pipe, and is less severe than the breaks in the discharge pipe, which will be described in the next exhibit.

I would like to discuss this exhibit, which is Exhibit Number 4, and still on Table 1, which is the analysis of the spectrum of cold leg breaks in the pump discharge pipe, and I would like to discuss these in a very generic or whole fashion for convenience, and use some of the figures in Exhibit Number 4.

In this analysis, to describe the operator actions, and what equipment was utilized, in the analyses presented, in both of these reports, the one I just discussed and this one, we have assumed emergency feedwater system. As will be shown later on from other analyses we have performed, the emergency feedwater system is not a significant actor on the transient for these sized breaks.

These are breaks all greater than the .02 square foot size, which as they testified earlier do not need the steam generator for heat removal.

Now, in this analysis again we have assumed the loss of offsite power reactor trip, but we have assumed an operator action in the analysis being presented in this next exhibit. The analyses -- well, it was found back in this time frame and this analysis, which is 1978, that it was possible that we had not looked at the worst break location, and subsequent investigation after that concern was raised determined indeed that we had not done a sufficient job in examining all break locations in light of the fact that there were substantial model changes made over the life of the plant.

We then undertook an analysis to demonstrate that this break location would be handled and we identified that it would require an additional operator action. Under a single failure assumption where only one high pressure injection pump was operating, you would have one high pressure injection pump discharging into two injection lakes in 1978. This was the pre-modified HPI system.

If you postulated the break between that injection point and the reactor vessel located specifically at the bottom of the pipe, the high pressure injection water that entered the broken runup pipe would just spill on the floor, so basically you were left with only 50 percent of one high pressure injection pump to handle the event.

That was found not to be sufficient. But if the operator performed a manual action within ten minutes to open the cross connects at the discharge of the pump as shown on Exhibit 14, he opened those valves, the two that are closed there, he would then be able to get about 70 percent of one high pressure injection pump fluid into the system through the three lines not connected to the broken cold leg. So, he would increase essentially the fluid injection that would reach the reactor vessel for core cooling.

With that assumption in the analyses, we then performed a series of break size evaluations which are shown in Exhibit Number 4. As shown in Figure 3, which is the pressure versus time history for these cases --

CHAIRMAN SMITH: That is Figure 3 of Exhibit 4?

WITNESS JONES: Yes.

DR. JORDAN: Does that still represent the situation at TMI 1? Have there been any changes in additional flow restrictions that have changed it so this is really past history we are talking about?

WITNESS BROUGHTON: Yes, sir. There have been modifications made to the high pressure injection lines which provide cross-connects and flow limiting devices such that no operator action is required to provide the flow assumed in the analyses that Mr. Jones will discuss.

DR. JORDAN: Very well. Then I think we can probably skip over it fairly fast.

WITNESS JONES: Well, I want to discuss this from the standpoint of this evaluation and the evaluation that I just discussed on the 50.46 compliance evaluations for TMI 1.

DR. JORDAN: I see. Okay.

WITNESS JONES: All that the change in the system has done in the HPI system has basically been to make these analyses conservative relative to TMI 1, because they will provide in fact more flow than assumed in the evaluation, because they would not have lost 50 percent of the fluid for the first ten minutes.

DR. JORDAN: Very well.

WITNESS JONES: But just to quickly go over the spectrum, as you see, there is a generally smooth transition in the pressure as a function of time.

DR. JORDAN: Very well.

WITNESS JONES: But just to quickly go over the spectrum, as you see, there is a generally smooth transition in the pressure as a function of time for this range of break size. The largest break size, of course, depressurizes th astest, and the smallest the latest, and it is a relatively smooth transition throughout these cases.

In performing this evaluation, one of the keys that we used in performing this is the break size. The smallest break size that we looked at is a break size which is totally mitigated by the high pressure injection system. That is, the core flood tank plays no role in mitigation of the transient, and as shown on Figure 3, the .04 square foot break does not depressurize in the time period of the evaluation to the core flood tanks.

As far as the consequences of this system pressure trace and the HPI flow, figure 4 shows the mixture height for these cases. It gets a little jumbled in the two to 400 time frame because there are several cases there. But the significant point here is that there are only three cases which had any core uncovery. These were the .055 square foot break, the .07 square foot break, and the .085 square foot break.

And, of course, any breaks in between there and slightly to either sides of the break size would result in this evaluation to some small core uncovery, but you can see that it is a fairly smooth envelope around this point, as far as the minimum mixture level, and you can see it continually decreases and slowly is building back up, and the trend would be expected to be valid on both sides.

DR. JORDAN: When you speak of the mixture height, this is, as I understand it, a two-phase mixture with a significant amount of water. Is that true? And what is meant by significant? If it is true, what do you mean by significant?

WITNESS JONES: It is a two-phase mixture, and I believe its water content is on the order of 70 percent for these cases.

DR. JORDAN: Oh, yes, that is quite high. Thank you.

WITNESS JONES: And as stated, this evaluation uncovered basically what the worst case would be, the 07 square foot break. It has roughly one foot core uncovery for approximately a 400 second time frame, and figure 5, which is very difficult to see, is the peak clad temperature evaluations for these cases, and the 07 square foot case resulted in a peak clad temperature of less than 1,100 degrees, which is substantially below the criteria of 5046.

DR. JORDAN: These are, of course, Fahrenheit degrees we are speaking of entirely.

WITNESS JONES: Yes.

(Pause.)

WITNESS JONES: The analysis in support of Table 2, I believe that covers Table 1, unless there are any further questions.

DR. JORDAN: No, that is fine. Thank you.

WITNESS JONES: The evaluation in support of Table 2 is provided in Exhibit 9. The analysis that was performed assumed basically a transient in which all feedwater was lost. As listed in the sequence of events, basically, the transient we analyzed was loss of main feedwater. We assumed for conservatism that the anticipatory reactor trip on loss of the feed pumps did not work, and that the reactor would trip on the high pressure signal.

We assumed we had a loss of off-site power in the evaluation.

MS. WEISS: Excuse me. Could you repeat the first sentence again? We were shuffling our papers and did not hear. Maybe the reporter can read it back.

MR. BAXTER: The last sentence?

MS. WEISS: The first sentence.

MR. POLLARD: Mr. Chairman, if every time he changes to a new exhibit or new table, if he could just pause for a few seconds, it would be very helpful.

WITNESS JONES: I am sorry.

DR. JORDAN: Do you remember what you said?

WITNESS JONES: I can start it over, I think.

MS. WEISS: We got the loss of feedwater, and then we sort of drifted off after that.

WITNESS JONES: Basically, the analysis which was performed assumed, as the initiating event, first a loss of main feedwater. It was also assumed in the evaluation that the anticipatory reactor trip on loss of all main feedwater did not occur, and this resulted in a reactor trip on the high pressure set point. That is a very conservative assumption, and tends to minimize the fluid available in the steam generator for subsequent boiling as a heat synch.

It then went on with a typical assumption of the loss of offsite power, a very atypical assumption, which was that emergency feedwater was lost to the steam generators.

That is, it did not work. None of these systems. And basically the evaluation which was discussed, I believe, in some detail over the last few days is essentially a confirmation or demonstration of the feed and bleed mode of core cooling.

Now, to describe the system response, if you turn to Figure 2 of Exhibit Number 9 -- does everybody have that now? I think I will ask first before I go on. It will be easier that way.

You can see it is a fairly expanded scale. There is an error on the last two points on the graph for the time. That should be 10,000 and 12,000 seconds. But basically the response of the system is to initially repressurize the 2,300 psi, which you can barely see, to decrease to roughly 2,050 psi, and with they dryout of the steam generator, to repressurize to roughly 2,500 psi.

DR. JORDAN: That is the safety valve setting?

WITNESS JONES: That is correct. We did not use the PORV in this analysis, and used only the safety valves to relieve system pressure.

MS. WEISS: Is that one or two safety valves?

WITNESS JONES: The analysis used two safety valves, but it turns out that it actually only used the capacity of two safety valves for only a very short period of the transient, which I can probably point out on one of the subsequent figures to give you a handle on what happened.

Figure 3 presents the pressurizer mixture level in this accident. As would be expected, the pressurizer fills up as a result of the repressurization of the system due to the expansion of the primary system fluid as it heats up, because of the lack of heat synch.

DR. JORDAN: Forty feet, is that the top of the pressurizer, or is that the top of the gauge?

WITNESS JONES: Well, the models that are utilized, I am not sure whether it is the top of the gauge or what, but as far as the model, that is the top of the volume utilized. We would knock up the total volume, but the normal cross sectional area and the cylindrical section, and get a somewhat artificial height on the two ends, but it would track the proper volume, which is the significant point.

DR. JORDAN: I see. So, this filling up, then is the result of the high pressure injection?

WITNESS JONES: No, the filling up in this portion of the accident is simply the result of the heat being added to the fluid by the core decay heat, the expansion of the fluid as it heats up, and pushing liquid into the pressurizer while the system repressurizes.

DR. JORDAN: Does this mean the steam valves then would be -- I mean, the safety valves would be seeing liquid water at that time?

WITNESS JONES: Once the thing becomes totally solid, yes, it would be seeing liquid water, and then later in the transient, it is a little difficult to see, but the pressurizer level does decrease somewhat, and basically what happens at this point in time is, you are having phase separation in the pressurizer where you are discharging steam into the pressurizer through the surge line and it is bubbling up through the water that is in the pressurizer and is being bled out of the valves as steam.

To try to describe the discharge of the fluid going through the pressurizer safety valves, I would like you to just take a quick look at Figure 4. And as noted in Figure 4, there are basically three distinct regimes of flow through that valve. Between roughly 400 seconds and 1800 seconds, the flow through the valve is liquid. No steam is in the liquid, and the rate is being determined basically by the expansion of the fluid.

I would like to note that also at 20 minutes in this evaluation, we have assumed that the operator had initiated one high pressure injection pump. He is directed to actuate all, but we assumed a single failure also beyond that assumed for all the feedwater to be lost where he had had only one high pressure injection pump, and that was manually initiated in 20 minutes.

DR. JORDAN: I missed the point. Didn't the safety features actuation signal come on early and start the injection pumps?

WITNESS JONES: No. As discussed on the system pressure figure, and I would have brought it up there, the system pressure only reached a minimum of about 2,050 psi, which is well above the actuation set point of the high pressure injection system. And basically the failure to depressurize to the set point that actuates the system is a result of the total loss of the heat synch.

You will not for all breaks, and this case has no break in the system, you do not depressurize the system low enough.

DR. JORDAN: Okay. I have missed a point then. The safety features actuation signal on pressure is only on low pressure, not high pressure?

WITNESS JONES: That is right. The high pressure was a reactor trip signal.

DR. JORDAN: Okay. So therefore, during this first 20 minutes, there has been no high pressure injection of water. There has been some synch presumably by virtue of the fact that there has been water left in the heat exchangers. Is that correct?

WITNESS JONES: Yes, for roughly the first three minutes of the transient.

DR. JORDAN: So after the first three minutes, then, we are left with no synch, just the heating up of the water. And that continues then for the first 20 minutes. Is that correct?

WITNESS JONES: Yes, and in fact, the heating of the water continues longer than that.

DR. JORDAN: Yes, but at the end of 20 minutes, then the operator does actuate the high pressure injection system. Is that correct?

WITNESS JONES: Yes, that is what we have assumed in the evaluation, and we have assumed that only one functioned when in fact we would expect both.

DR. JORDAN: I see.

WITNESS JONES: Now, at the time we get to 1800 seconds, at this point in time we have basically finished the liquid discharge phase, pure liquid, and what is happening to the system at about this time is, the system is reaching -- well, the system is starting to undergo vigorous boiling. And that the valve that the fluid being discharged through the valve is being controlled by the volume of steam being created. It is trying to make room for itself, and is pushing water, steam, and pushing out of its way to find a pat[illegible]olume to sit in.

I said I did want to talk a little bit about the valve, just try to hit the point on how many safety valves are being utilized. Between zero and 1800 seconds, we are using roughly 40 percent of the capacity of one valve in the evaluation. That is what is being used. Not that is what is modeled. We have the valve modeled but we are only using roughly 40 percent of its capacity.

Now, during this period of time where we are vigorously boiling and shoving fluid out becase of the boiling process between 1800 seconds and roughly 2200 seconds --

DR. JORDAN: Two thousand?

WITNESS JONES: Well, 2,200, a little past that time. Where we are pushing out a fairly substantial amount of water and steam mixture. During that period of time, we are using near the capacity of both safety valves. They would both be utilized or have been utilized almost to full capacity. After 2,200 seconds, after we have discharged this much inventory in the primary system, we have lost inventory up to the surge line into the pressurizer. That is, the steam that had been created has discharged enough fluid so that there is a steam interface at the surge line, and at this point in time you basically have steam flow, only steam flow into the pressurizer, and your only loss of inventory is due to a boiling process, and you are trying to catch up to that boiling process with a high pressure injection system which you had actuated earlier.

And from 2,200 seconds to roughly 9,000 seconds, I believe it is -- yes, roughly 9,000 seconds, you have a mismatch between the boiloff of the core and the flow being injected by the high pressure injection system. And you continually lose inventory. After about 9,000 seconds, you would have a slow system refill.

Now, through this entire period of time, from 2,200 seconds to the 10,000 seconds analyzed, we are discharging steam through the valve, and we are again only using roughly 40 percent of one safety valve for the necessary capacity.

Now, a part of the system volume as a function of time for this transient is shown in Figure 5. I don't want to belabor that point, but the big actor there or the point to be made from that is simply that the core remains covered throughout the transient, and there is roughly an additional 1,000 cubic feet of water above the top of the core available for cooling.

DR. JORDAN: I see. And so you have reached a state of equilibrium.

WITNESS JONES: Yes, and we are slowing refilling. Not very fast. That is based again on only one pump. Not to mislead you, the analysis did assume a realistic core decay heat. We felt that the number of failures assumed in the evaluation were more than enough to require us to use the normally conservative 1.2 ANS.

DR. JORDAN: I understand.

That is listed as one of the assumptions.

WITNESS JONES: Yes, I just wanted to point it out because it is that.

DR. JORDAN: Thank you. I might have missed it.

WITNESS JONES: And I do want to make -- That is a reasonable value based on the knowledge today of the decay heat.

DR. JORDAN: We appreciate this very much, and we think you are deserving of a break for about ten minutes, and so we will come back, but you are doing just great.

WITNESS JONES: Thank you.

(Whereupon, the hearing was briefly recessed.)

BY MR. BAXTER: (Resuming)

Q Would you continue, Mr. Jones.

A I guess at this time we are on Table No. 3. And to get some of the system response, I will be using Exhibit 5. And in fact, Exhibit 5 will be used for several of the next few tables, I believe.

Q Excuse me, Mr. Jones. We're getting Exhibit 5.

When you referred earlier to moving to Table No. 3, you were referring to your written testimony of September 15th

A That is correct.

Table 3 is a summary of the analyses performed of a small break LOCA, with an assumed loss of all the steam generator water. The basic sequence of event and assumptions are listed on the table. And it's basically, we have assumed a small break LOCA, and we have looked at several sizes, specifically a .07 square foot break, which was the design basis break identified from the 10 CFR 50.46 analysis, a .02 square foot break, and a .01 square foot break.

The analyses, all the analyses, result in a depressurization transient initially, which actuates the reactor trip on low reactor coolant system pressure. As before, we have assumed that we lose off-site power at that time with the reactor trip, and we have assumed that the emergency feedwater pumps do not operate.

In performing the analyses, we have assumed that both high pressure injection trains function, and we have used the core decay heat of 1.2 times the ANS standard. Now, for specific information on the analyses, there are two figures I would like to first bring out, which are Figure 6.2.2 -- and I guess we can discuss this one first and then move on to the next figure.

This is the core pressure versus time for .07 square foot break. It is very similar to the system pressure trace for the 50.46 analysis. And what that indicates simply is that the steam generator does not play a significant role at all for the design basis small break LOCA's. As seen from the system pressure trace, there is a fairly rapid depressurization, which results in automatic actuation of the high pressure injection pumps. The system then decreases to approximately 1,000 psi and more or less stabilizes there, while the system inventory depletes, until you get to high-quality steam flow out the break, which results in a continued system depressurization and ultimately the actuation of the core flood tanks.

DR. JORDAN: What caused the change in slope at 500 seconds?

WITNESS JONES: Basically, the quality of the fluid going out of the break. There are figures in this whole section which will illustrate that.

I would now like to go -- I don't want to go through all the figures for this case. The basic conclusion of this analysis was that the core remained covered throughout the transient without feedwater and without any operator action during the short-term.

The next figure I would like to turn to is Figure 6.2.20.

DR. JORDAN: Okay.

WITNESS JONES: And this figure is the system pressure versus time for a .02 square foot break, and it is this analysis that forms the basis for the conclusions ion our testimony in response to UCS 1 and 2 that breaks larger than .02 square foot do not need the steam generator. As noted by the system depressurization, the system again depressurizes down to the emergency core cooling system actuation signal of approximately 1600 psi, though the actual analysis value was somewhat less.

At the actuation of the high pressure injection pumps, the energy that could be absorbed by the high pressure injection fluid plus the energy discharge through the break results in a fairly stable system pressure transient. And by the end of this analysis, we had matched the core decay heat.

DR. JORDAN: I am having a little trouble with the ordinate. I multiply by 10^{-2} the numbers there; is that it?

WITNESS JONES: No, you multiply by 10^{2}. Multiply by 100.

DR. JORDAN: Oh, that's a decimal point, 25.000.

WITNESS JONES: No.

DR. JORDAN: Oh, that's 25,000, 25.000.

All right. Now, the system pressure begins around 2200 psi, normal operating pressure, and falls, you say there, to 1400 psi about --

WITNESS JONES: Yes, about this pressure, and then it stabilizes.

DR. JORDAN: And at that time the high pressure injection system came on automatically, is that right?

WITNESS JONES: That is correct, the high pressure injection system for this case is automatically actuated.

DR. LITTLE: This is the computer-generated figure, isn't it? And you don't actually have five significant figures there, I don't think.

WITNESS JONES: It is a computer-generated figure, and we have a lot of significant figures in our computer model.

DR. LITTLE: Are they real?

WITNESS JONES: Some yes, some no. It depends on the state searches, for example. The significant figures can become important to carry them out very accurately.

DR. JORDAN: But are you again saying, then, that this reaches a state of equilibrium fairly rapidly?

WITNESS JONES: Yes, it does. And that the basic equilibrium or the energy process for the energy removal added to the primary system fluid by the core decay heat is via absorption on the cold high pressure injection water, energy absorbed by that, and the energy discharged through the break, which leads to a more or less stable system pressure trace.

And it is difficult to see on the figure, but generally you see a slowly depressurizing system as a result, as the decay heat decreases.

DR. JORDAN: Now, if the break is smaller than the .01, then do we go back to the case of essentially no break, zero break size, which you discussed previously?

WITNESS JONES: The next figure -- the next set I want to discuss is the next size.

DR. JORDAN: Okay. Thank you.

(Pause.)

DR. JORDAN: Take your time. Don't hurry.

CHAIRMAN SMITH: If you can tell us what you are looking for, maybe we can go to it at the same time you are.

WITNESS JONES: I've got it, I think. Yes, I've got it.

I would like now to refer to Figure 6.2.60.

DR. JORDAN: Does everyone have it? Okay.

WITNESS JONES: The analysis presented here is an analysis of the .01 square foot break, and this is the system pressure as a function of time. Now, as shown, the initial response of the system is to depressurize.

DR. JORDAN: I thought the previous figure we were looking at was the system pressure.

WITNESS JONES: That was a .02. This is a .01 square foot break.

DR. JORDAN: I see, okay.

WITNESS JONES: This is on the other side now.

DR. JORDAN: Good. I missed the point.

WITNESS JONES: And this is in fact where Mr. Johnson and I got two different numbers for where you need generator. It is interpretive as to which side it is on. It is between the two cases.

DR. JORDAN: Yes, I understand.

WITNESS JONES: Now, as you can see, the system initially depressurizes because of the loss of fluid inventory and results in reactor trip on low pressure. The system continues to depressurize until approximately 200 seconds, at which time we have boiled off the inventory that was stored on the secondary side of the steam generator. And because of the assumptions used in the analysis, which was an ESFAS or emergency safety features actuation signal of 1350 psi, you can see that we do not result in automatic actuation.

In fact, we would expect for this case to get automatic actuation under normal circumstances. But we assumed, because it was a generic evaluation for all of our plants and some of them have lower set point, we assumed the lower set point minus its instrument error in the environment.

With the loss of heat sink, the system repressurizes slowly and by 20 minutes is up to approximately 2400 psi. We at this time assumed that the operator manually initiated the high pressure injection systems and took no further actions, just actuated those.

And as you can see from the figure, the system pressure flattens out due to the colder water being injected, which decreases core boiling rate and slowly depressurizes with time. And for this case again, no core uncovery was found. This is kind of the largest break that you would expect to ever get into a feed and bleed mode of cooling.

As you can see, it comes very close to the safety valve set points, and would bound basically the effects of any other sized break, because of the largest break area resulting in the largest loss of liquid inventory with time.

BY MR. POLLARD:

Q May I ask just one question. When you assume the operation of the high pressure injection pumps, you are assuming full flow at that pressure

A (WITNESS JONES) Yes.

DR. JORDAN: Now, wait just a minute. High pressure injection came on at what time in this figure?

WITNESS JONES: 20 minutes, 1200 seconds, maybe 1250. I'm not sure of the exact time.

DR. JORDAN: And that's where the curve peaks, is that it?

WITNESS JONES: That's right.

DR. JORDAN: Now, the high pressure injection system comes on at that time, and why does that stop the pressure rise?

WITNESS JONES: The pressure rise is occurring from the creation of steam via boiling and then, you know, its mismatch relative to the volume of fluid being discharged out the break. And up until this point in time you are generating a larger volume of steam than you are displacing a volume of liquid out the hole, and the system is repressurizing. Because that volume of steam, the mass of steam you are creating needs more volume, and the way it is doing it, really, is actually compressing itself so it needs less, which means the system pressure must rise. Now

--

DR. JORDAN: But it doesn't rise to the point where it hits the safety valve?

WITNESS JONES: It di' not quite make it. You will remember, we had a leak in the system which is slowing things down, the repressurization. As breaks get smaller from this, you would reach the safety valve.

DR. JORDAN: If you hadn't put on the high pressure injection system.

WITNESS JONES: If you hadn't, you would hit it, yes. And if you were at a smaller break and waited 20 minutes to put on, the high pressure injection system, you will also probably hit the safety valves, because the inventory loss through the break would have been decreased as the break size gets smaller.

DR. JORDAN: I am obviously missing something completely. Adding the high pressure -- water from the high pressure injection pumps can't be doing very much from the standpoint of cooling. There is already a large inventory of water, and the heat is being dissipated by generation of steam going from liquid to vapor. And I don't see what makes it turn around at 1200 seconds.

WITNESS JONES: The high pressure injection does provide a substantial amount of cooling. It is 90-degree water which you will have to raise to, for these type of pressures, 650 degrees.

DR. JORDAN: Okay, so that's what is doing it.

WITNESS JONES: And that takes some of the core decay heat away, which then decreases the amount of steam production occurring at this time. And that decreased steam production in combination with what you are losing out the break causes this curve to flatten out.

DR. JORDAN: I see. So that the change in slope, then, is really due to the cooling from the high pressure injection system?

WITNESS JONES: That is correct.

There is one other conclusion that we reach on Table 3, which is the effect of, instead of the operator actuating two high pressure injection systems at 20 minutes, what happens if he actuates the auxiliary feedwater system instead. And that analysis is provided in Licensee's Exhibit No. 8.

DR. JORDAN: Do I dare put this one away?

WITNESS JONES: No.

DR. JORDAN: We're having a little trouble finding two copies. But let's go ahead.

BY MR. BAXTER: (Resuming)

Q All right. Mr. Jones, one point of terminology. We have been referring to the TMI-1 system as an emergency feedwater system. Are you using auxiliary feedwater and emergency feedwater as interchangeable terms here

A (WITNESS JONES) Yes, I am.

I would just like to point to Figure 1 of this exhibit and just briefly discuss this analysis.

DR. JORDAN: This is now he has turned on the emergency feedwater; is that right?

WITNESS JONES: I was going to quickly explain that. This is the same case that we have just discussed, the .01 square foot break, except instead of turning on the high pressure injection systems manually at 20 minutes, he somehow manages to get back the emergency feedwater system at 20 minutes.

DR. JORDAN: Yes.

WITNESS JONES: And as you can see, there is a fairly rapid depressurization of the system as a result of cold auxiliary feedwater causing steam within the primary system to condense, which results in a depressurization transient.

DR. JORDAN: So this is the boiler condenser operation; is this right?

WITNESS JONES: That is correct. That is what you would be in at this time when you restored feedwater. The system rapidly depressurizes, automatically actuates the high pressure injection system around 2100 seconds, which then provides the cooling necessary to keep the core covered.

And that is all I have for that exhibit.

MR. POLLARD: May I just ask two clarifying questions.

BY MR. POLLARD:

Q In comparing Figure 6.2-60 in Exhibit 5 with Figure 1 in Exhibit 8, if I understand your explanation, in the high pressure injection case we have the pumps coming on at 1200 seconds, and in the feedwater case we have the pumps coming on at 1200 seconds. But it appears from these figures that the pressure gets higher. The peak pressure in one is higher than the peak pressure in the other. Could you explaion that?

A (WITNESS JONES) I think it is probably a result of slightly different times for actuating the systems. Let me check for some details.

DR. JORDAN: First, do you agree with the characterization that it's not all that obvious to some of us?

WITNESS JONES: It does appear, as best I can lay the computer plots together, that it is slightly higher. But let me -- as I said, I would like to look up a detail. It is difficult to tell you the exact, you know, unless the information in the exhibit has the exact proper time. I would have to go back and look at the computer run, because typically these are what you call restart computer runs. And we could set up one case with a specific time interval, and then to try to get the systems to actuate we may have to go slightly longer into the computer run in order to get the logic and the code to properly reflect it. But let me just look through that.

DR. JORDAN: I think that is a sufficient explanation at this time. If Mr. Pollard wants to pursue it later, we will let him do so.

DR. LITTLE: What is the precision on these numbers over here, plus or minus what? 2,000 psi plus or minus what?

WITNESS JONES: I don't really know. The computer evaluations that come out, of course, like any evaluation with these models, will probably have some uncertainty. I don't really know what it is. And part of the problem -- I won't say problem. Part of the reason that you do several of these evaluations for different break sizes is to assure yourself that there are not large pluses and minuses.

If the pressure is for this case, say, off by 100 psi, that is the same effect as having a slightly smaller break, and if it was lower it would have the same as having a slightly larger break. So by doing this spectrum analysis, you tend to wash out the types of uncertainties.

But as far as the statistical uncertainty of these codes, I have no idea.

DR. JORDAN: Well, certainly we're not going to depend on them for 5 percent numbers or something like that.

WITNESS JONES: Well, as I said, that is one of the reasons why we run spectrums, in order to wash out some of these effects.

DR. JORDAN: Okay.

WITNESS JONES: I guess I would now like to go on to Table 4 of my written testimony, and we will still be using Exhibit No. 5. And I would like to refer to Figure 6.2.62 of Exhibit 5.

Okay. This case is basically the TMI scenario, without throttling of the high pressure injection system. That is, the evaluation is a loss of main feedwater event which results in a system repressurization, which opens the PORV and it sticks open.

We have a reactor trip in this case on high pressure. We have assumed there is one difference relative to the TMI scenario, which is that we have assumed that the emergency feedwater works. And we have assumed also a single failure in the high pressure injection system. This is essentially an analysis of the transient-induced LOCA. A small break loss of coolant accident is similar to the TMI event.

DR. JORDAN: But now, this does assume that the set point for reactor trip is lower than the PORV as required by the new modification?

WITNESS JONES: No. This analysis was done assuming the old set points.

DR. JORDAN: So the first thing that happened was that the PORV tripped. But nevertheless the pressure continued to rise until the reactor tripped; is that right?

WITNESS JONES: Yes.

DR. JORDAN: Fine. I see that.

WITNESS JONES: And it would be very similar to a case with the PORV set points inverted, because the time frame that all these actions occur in are in the first ten seconds of the transient.

DR. JORDAN: Yes, I see that.

CHAIRMAN SMITH: Mr. Jones, on the second event you refer to repressurization. What repressurization over what pressurization are we referring to? Because you didn't have your LOCA at that point yet when you used the word "repressurization."

WITNESS JONES: On Table 4?

CHAIRMAN SMITH: Yes, sir.

WITNESS JONES: Well, yes. The repressurization, the reactor coolant pressure increase, I believe you're talking, from line 2?

CHAIRMAN SMITH: Yes.

WITNESS JONES: Okay. That repressurization is a result of the loss of main feedwater. The loss of main feedwater results in a boiling of the inventory on the secondary side, which decreases the heat transfer surface from the primary system, which allows the system fluid to heat up somewhat, which expands, compressing the steam space in the pressurizer, and causes the system pressure to increase.

CHAIRMAN SMITH: I understand that. What I did not understand was when you say "repressure," where did you explain your drop of pressure for it to repressurize?

WITNESS JONES: Well, it doesn't say it repressurizes.

CHAIRMAN SMITH: I know. You said it.

WITNESS JONES: I'm sorry, I must have just misspoke myself.

CHAIRMAN SMITH: That is what I thought.

WITNESS JONES: Basically, it is a fairly simple transient. We have assumed in this case that the reactor coolant pumps were running continuously. This analysis was done to more or less simulate a prediction of TMI with our evaluation, should -- if things had occurred correctly. And as seen, the system just depressurizes -- after the initial pressurization transient reactor trip, the system depressurizes, actuates the high pressure injection system automatically, and the system pressure stabilizes at roughly 1150 psi. And the forced circulation keeps very good heat transfer to the steam generator, and that controls the system pressure. And the accident is very easily handled.

That is basically everything on Table 4, on Table 5 of my written testimony.

DR. JORDAN: This is a case, of course -- the open PORV is equivalent, then, to a break larger than .01.

WITNESS JONES: No. The PORV is a break area of about .007 square feet, a 1.05 square inch orifice area.

DR. JORDAN: So slightly less than the case we were discussing a few minutes ago?

WITNESS JONES: Yes. But the case we were discussing a minute ago did not have feedwater delivered to the steam generator. That case does.

DR. JORDAN: Of course. Thank you.

WITNESS JONES: For Table 5, I will be using Exhibits 6 and 7.

(Pause.)

DR. JORDAN: Okay. I guess everyone has one.

WITNESS JONES: I would like to turn to Figure 2 in both of those documents. They are the core pressure traces for both cases.

DR. JORDAN: Okay.

WITNESS JONES: The basic event that was being analyzed in both of these analyses, they are the same event. And I did want to put the two side by side each other.

The testimony specifically is addressing Exhibit 7. Table 5 is basically off of Exhibit 7, but I would like to discuss both of these to get a complete record. The event is a failed open PORV with feedwater not delivered to the generators. Both main and emergency feedwater is lost.

DR. JORDAN: I see. So the transient is in this case a PORV opening by itself and staying open?

WITNESS JONES: Yes, with a total loss of feedwater.

DR. JORDAN: Yes.

WITNESS JONES: And additionally to that, we have assumed a single failure in the high pressure injection system. So it is a fairly severe set of failures.

The analysis in Exhibit No. 6 is based on 1.2 times the ANS decay heat curve, and the analysis in Exhibit 7 is 1.0 times the ANS decay heat curve. The analyses themselves are fairly similar in their response, and I will just generally characterize the system response for these events.

The system depressurizes in both events, resulting in an automatic reactor trip on low pressure and an automatic emergency safety features actuation signal being reached, which actuates the high pressure injection system.

Now, as discu-sed earlier, it was stated that a .01 square foot break would not result in actuation of the emergency safety features. And yet, for this case you can see that a .007 square foot break resulted in actuation of the syste. The main reason for that apparent inconsistency is the difference between the fluic)eing discharged in both cases for a break.

The .01 square foot break described previously was a break in the cold leg which discharged water. So its effect on the system is smaller than a break like in this case, which is a steam side break, which results in a much faster depressurization for any given leak flow.

DR. JORDAN: Are you going to make any attempt to tell me why steam comes out faster than water? Is it a matter of viscosity or what?

WITNESS JONES: It is not that it comes out faster. You put the system condition together, you have in these early portions of all of these transients basically a normally full reactor coolant system with a pressurizer with water in it and a steam space above it, with the pressurizer being the controlling pressure point.

Now, if you have a break in a ...er space its influence on the system pressure is to decrease the level in the pressurizer, and that expansion of steam is what determines the system pressure. When you pop the top of the pressurizer and draw off steam, its effect on the system pressure is much more dramatic.

DR. JORDAN: Okay. I see.

WITNESS JONES: Now, in both these cases at about 600 seconds or thereabouts the pressurizer essentially fills, and as a result of that the system pressure starts to increase, because instead of bleeding steam you are now taking out water.

In the case of the 1.2 ANS, you can see the system pressure is on a continuous rise, and the best way to characterize it is you are losing the race. As your pressure goes up, your HPI is going down, which means you're getting less inventory makeup, which means you get more boiling and the core decay heat is changing faster.

It's contribution is larger because of the 1.2 factor, and you are not catching up as your pressure is rising. It is not decaying rapidly enough for you. And this case, if continued, would not be coolable with one high pressure injection pumps.

If, however, you restore either a second pump -- this is 30 minutes -- or restore auxiliary feedwater to the system, you would be able to keep the core cooled. For the 1.0 ANS decay heat curve, because of the mismatch in the energy absorption capability for the high pressure injection system and the core decay heat is lessened relative to the other case, the 1.2 ANS, you can see that the repressurization is slower. And as the core decay heat decreases, you ultimately match it with the HPI, provide enough cooling, and start to decrease the system pressure. And that case would remain coolable with just one high pressure injection pump.

DR. JORDAN: In the case of the 1.2 ANS, if the curve were followed longer would it go up and hit the safety valves, or would you run out of inventory first?

WITNESS JONES: I really don't know.

DR. JORDAN: But in any event, you are not dissipating the heat as fast as it is being generated?

WITNESS JONES: That is correct, not with the one pump.

I guess now I would like to move on to Table 6.

(Pause.)

And once again, I will be using Exhibit No. 5, specifically Figure 6.2.93 and 6.2.148.

DR. JORDAN: How did you know what figures?

WITNESS JONES: I had to look for them, like everybody else.

DR. JORDAN: Okay.

WITNESS JONES: These analyses, the analyses that are discussed in Table 6 and shown on these two figures that

I just stated, are basically Appendix K analyses of very small break LOCA's. That is, as seen previously, we had performed analyses down to approximately .04 square feet. These are two other analyses which have been performed using the same Appendix K assumptions.

We have assumed that we have a very small break LOCA and in this case on the order of .005 square feet and .01 square feet.

DR. JORDAN: That is the difference between the two curves you are pointing out?

WITNESS JONES: Yes, that is the difference. It is strictly the break size.

We had used a core decay heat of 1.2 times the ANS standard as by Appendix K. We have assumed a single failure in the high pressure injection system. And we have assumed that the emergency feedwater is delivered to the steam generators.

These two analyses basically are the analyses which form the bases for the testimony in response to UCS 1 and 2, describing the system response during a small break LOCA and how the energy is removed via the steam generator for smaller break sizes.

What occurs in these evaluations in both cases initially is you get a system depressurization, resulting in an automatic actuation of the -- you're on automatic reactor trip -- an automatic actuation of the high pressure injection system. When the system depressurizes to approximately 1400 psi, you get some flashing within the primary system, which slows down the depressurization.

You have some steam production and that slows the depressurization rate. However, at this point in time you have a natural circulation process occurring. The system is basically liquid-full and you have a normal liquid natural circulation process occurring.

DR. JORDAN: Even though it is two-phase liquid in part?

WITNESS JONES: Even though there are some voids in the system.

DR. JORDAN: Okay.

WITNESS JONES: Now, for the .01 square foot break, at approximately 600 seconds the voids take up a significant -- well, take up a large enough volume in the primary system to block the path, the 180-degree U-bend in the hot legs. For the .005 square foot break, that does not occur until approximately 1200 to 1300 seconds.

As a result of the interruption of the natural circulation flow and because of the small size of these breaks, you get a system repressurization. At approximately 1500 seconds for the .01 square foot break and at around 2400, 2500 seconds for the .005 square foot break, you establish the boiler condenser mode of cooling. What happens is during this time period where the system is repressurizing, you are continuing to lose fluid in excess of the capacity of the HPI.

So you continue to lower the level in the primary system until such time that you expose the surface area upon which steam can condense on the cold tubes.

DR. JORDAN: I see. This is in the steam generator primary side.

WITNESS JONES: That is correct.

DR. JORDAN: I see. I had not really understood that until today.

WITNESS JONES: And you go right into the boiler condenser and you will slowly repressurize over time for both cases, as the core decay heat and the demand for energy removal via the steam generator decreases. Again, in both cases, like the others, adequate core cooling is continued to be maintained throughout the transient due to the high pressure injection system.

That finishes the use of Exhibit 5.

CHAIRMAN SMITH: Mr. Jones, I want to point out that your title of Table 6 omitted the word "break." You want very small break LOCA's, the way you identify it.

WITNESS JONES: Yes, I mean -- "LOCA" automatically means break to me. I missed that one. I'm sorry.

Table 7 of my testimony is based on the analyses presented in Exhibits Nos. 10 and 11.

DR. JORDAN: Okay.

WITNESS JONES: To start off, I guess, I would like to give a little narration before I get to the figures. But the figure that I will be ultimately examining will be Figure 2.5 of Exhibit No. 10.

Just to provide some background for this analysis, the NRC issued I&E Bulletin 79-05C in roughly August, late July and early August, requesting that analyses be performed assuming the reactor cooling pumps remain operative for some period of time during a small break LOCA, then are tripped subsequent to the accident at any possible time.

This was a departure from typical analyses assumptions, which were a loss of offsite power. Now, what these exhibits provide is the analyses in response to that specific -- the request of I&E Bulletin 79-05C, where we have looked at a spectrum of small break LOCA's which range from .025 square feet to up to a size of .2 square feet, and we have assumed that the reactor coolant pumps remain operative.

Now, Figure 2.5 shows the typical system pressure response for the spectrum of cases that we had analyzed, and is really not that much -- is really not that surprising relative to the type of system pressure traces we have seen from, you know, the reactor coolant pumps being off. But if you turn to Figure 2.6 --

DR. JORDAN: Also of Exhibit 10?

WITNESS JONES: Yes. What was found was the void fraction in the reactor coolant system could reach extremely high levels, 90, 99 percent, for certain of these size breaks. N w, basically what is happening for these transients with the pumps on is the pumps tend to keep the system homogeneous. When you do not have the reactor coolant pumps, the process is basically a very slow draining down of fluid through the leak and a slow loss of the liquid inventory in the system.

But once the inventory in the primary system has fallen below the nozzles that connect up to the reactor vessel, the only way to lose inventory is through boiling, so that you tend to collect or trap water without the pump running in the low point of the reactor coolant system, specifically the vessel and the loop seals in the pump suction piping and some in the steam generator.

DR. JORDAN: And the position of the break?

WITNESS JONES: Well, the position of the break that we had assumed in these analyses were in the cold leg pump discharge piping, and we did an evaluation of a hot leg break also. You are correct, if the break is in the pump suction you will not collect or maintain inventory in the pump suction for the side which is broken.

But that water, because of the geometrical arrangement of the system, is effectively lost water, anyway. Under a gravity draining situation, that water cannot get to the vessel. It sits there. It has no flow mechanism. It can't go uphill.

DR. JORDAN: Okay. But I guess I'm having a little trouble understanding the difference with the pumps running and the pumps not running. Is it because you have two-phased liquid at the break in one place and water essentially at the other place, and therefore that changes the rate at which you're losing inventory?

WITNESS JONES: That is basically correct. What happens is, without the pumps you lose water for a period of time and then you lose steam, basically. With the pumps running, you lose water continuously, irregardless of the void fraction, because of the assumed homogeneous nature of the primary system.

And what the system is trying to do is reach a void fraction whereby the HPI flow in equals the liquid mass out. And that void fraction is basically determined by the break size and how the system pressure changes, because that changes the leak rate. And as you can see, for a large fraction of these break sizes that we looked at, you reach very, very high void fractions, on the order of 95 percent.

Now, that is not a problem, provided that the reactor coolant pumps continue to run. The analysis we performed lemonstrated that keeping the reactor coolant running through this phase of the transient would provide adequate core cooling. There are some questions as to whether the mechanical integrity of the pump was such that it could continue to run in these void fractions, and as a result of this to postulate that the reactor coolant pumps may be lost at these high void fractions is not extremely unreasonable.

If you lose the reactor coolant pumps at these high void fractions, you will have very little inventory remaining in the system. You have only five percent of the system inventory left to you, and it will collect in the reactor vessel and in the loop seals when the pumps come off, because the mixing force is essentially lost.

Now, you have to refill the reactor coolant system and specifically the reactor vessel and core, try to recover the core. But these are high pressure transients and you do not have a pump which is capable of refilling the system rapidly, as you do at low pressures for large breaks. And for these cases it could not be demonstrated absolutely with Appendix K assumptions that you can provide adequate core cooling. And as a result of this, it was recommended that the reactor coolant pumps be tripped upon receipt of a low pressure ESFAS signal in the interim.

And there are other acceptable schemes that could be used, but at the time that we did the analysis that was the recommended trip signal for the plants that were operating.

Now, if you go to Figure 11 of Exhibit No. 11.

DR. JORDAN: Does everybody have it?

WITNESS JONES: This is essentially a synopsis of a study that we did, and what it shows basically is, if you trip the reactor coolant pumps early -- that is, to the left of the outline figure called "critical region" -- you would be able to assure adequate core cooling following a small break LOCA. If you tripped the pumps to the right of the figure, it is the same case. But that there is a region which we call the critical region under which we could not absolutely guarantee that adequate core cooling would be provided.

Though I would like to point out, one of the evaluations that we did within these two documents -- and I'm not sure which one it is in -- was a best estimate analysis of the consequences of losing the pump at the worst time. That analysis showed peak clad temperatures of approximately 2,000 degrees, so that this is not all -- this is a product of both the Appendix K analysis rules and under realistic circumstances this may not be a problem.

But it was determined that the pumps should be tripped promptly upon receipt of an appropriate signal, without diagnostics or anything like that. Basically, if you get the signal, go punch a button to trip the reactor coolant pumps.

DR. JORDAN: Thank you. I never understood that one before.

WITNESS JONES: The last table, Table No. 8, utilizes Licensee's Exhibit No. 13.

That is the analysis we had discussed earlier on the very small breaks, where no feedwater was available at all. We saw that the system could repressurize. And as I stated previously, smaller breaks would go up and actuate either the PORV or the safety valve.

This case is essentially an analysis of the potential consequential failure in that mode. It is where you have a -- the same case we were talking about, the .01 square foot break with no action for 20 minutes, and for the analysis what we did was we just simulated a stuck-open PORV, even though the signal had not quite been reached, because we recognized that a slightly smaller break would give us essentially the same results.

So we opened up the PORV at 20 minutes and also actuated the high pressure injection pumps at 20 minutes.

And what that analysis showed -- and if you look at, I guess, specifically Figure 1 of Exhibit No. 13, the system depressurizes as a result of the additional break in the system, and after two HPI's -- and that these two HPI's could handle not only the break itself, but the potential consequential failure that could occur under this scenario.

DR. JORDAN: Yes.

WITNESS JONES: That was the basic result and the reason that the analysis was done.

That finishes all my tables.

DR. JORDAN: Some time -- and it doesn't need to be today -- I will ask you to analyze those cases which required two high pressure injection pumps and the assumptions for heat rates or whatever else was connected with those. But I think that we will surely get into that when we get into cross-examination, anyhow. So it does not need to be done today, but it is going to come up. So get ready over the weekend, I guess is my thing.

Now, Mr. Chairman, it's almost the right time for -- it seems hardly worthwhile to start cross-examination.

WITNESS JONES: Dr. Jordan, could I ask one question, please? You said you want to go into some heat rates and stuff, and that may mean I've got to do some looking up the runs, and I'd like a little more direction.

DR. JORDAN: What I'd like for you to do is, to get ready for either my question or Mr. Pollard's: Are there not a number of situations where you require two high-pressure injection systems, and therefore don't you have a problem with the single-failure criteria? And I don't want to ask that question today, but I will ask it some time or other. I don't think I will, because Mr. Pollard will beat me to it.

WITNESS JONES: All right. I just wanted to understand if I needed to do a lot of analysis to look at it. Thank you.

MR. BAXTER: Mr. Chairman, at this point I would like to move into evidence Licensee's Exhibits 3 through 14.

MS. WEISS: No objection.

CHAIRMAN SMITH: Any objection, Mr. Cutchin?

MR. CUTCHIN: No objection.

CHAIRMAN SMITH: Mr. Adler?

MR. THEODORE ADLER: No objection.

CHAIRMAN SMITH: Licensee's Exhibits 3 through 14 are received.

(The documents referred to, previously marked as Licensee's Exhibits Nos. 3 through 14 for identification, were received in evidence.)

CHAIRMAN SMITH: I would guess you would want the weekend to prepare for your cross-examination?

MS. WEISS: I think we're going to want to start with what the witness has just said, rather than what we had planned to start with on the cross-examination plan.

CHAIRMAN SMITH: All right. We have provided for the delivery of the Intervenor's transcripts to your office in Washington on Monday. Has that been worked out satisfactorily?

MS. WEISS: Yes, it has been, yes.

CHAIRMAN SMITH: Is there anything further?

MS. WEISS: Could I bring up just a couple of scheduling matters as an alternative to moving the hearings to Washington, which I don't think really can be done, and we would not request that.

CHAIRMAN SMITH: We have never suggested moving the hearings to Washington. We said we would consider it on a particular circumstance, if you had a requirement.

MS. WEISS: In any case, we would like to request that we delay the starting point of Tuesday's hearings until perhaps 10:00 o'clock in the morning and go an additional hour that evening to accommodate some of us who are driving up. I think the Chair is himself.

CHAIRMAN SMITH: Always? I mean, week-in and week-out?

MS. WEISS: Well, when we are here I'd request it for our benefit. I don't know if the other parties need it or not. But at least next weekend we can't.

MR. BAXTER: We don't need it.

CHAIRMAN SMITH: The only thing you're suggesting is that we just shift the hearing day, for people who want to drive up on the same day the hearing begins?

MS. WEISS: Yes. If we could start on Tuesday morning, instead of 9:00 a.m., at 10:00 a.m., and perhaps go an extra hour that day, and make up for that hour.

CHAIRMAN SMITH: Does that create a problem for the reporting service?

THE REPORTER: No.

CHAIRMAN SMITH: Does anybody object to that procedure?

MR. CUTCHIN: It creates no problem for the staff, Mr. Chairman.

CHAIRMAN SMITH: Dr. Little suggested, too, that we might consider the possibility of starting earlier on Fridays and allot a bigger segment of time for that day. But we can take that up at another time.

MS. WEISS: We're dragging pretty hard by Friday.

CHAIRMAN SMITH: All right. We will meet, then, Tuesday at 10:00 a.m.

MR. BAXTER: Mr. Chairman, I discussed at least with Ms. Weiss the Board's latest determination that at least tentatively we would try to file cross-examination plans on agenda items 4, 5 and 6 on November 12th, and given the pace of the hearing we think it would be sufficient and we would propose just to file plans on agenda item number 4 on the 12th, with the Board's permission.

CHAIRMAN SMITH: All right.

MS. WEISS: I would just want to add one thing to that. We have had discussions with the Licensee and the staff about this potential problem of reaching item number 5 before the staff's interrogatories are completed. I just wanted to report to you that the staff has told us that they're going to make every effort to get those to us by Thursday of next week.

If that happens and if the hearing proceeds as we think it will, then that would obviate the need for rescheduling. But if that doesn't happen, we would need -- I just want to put the Board on notice that we might need to do some rescheduling.

CHAIRMAN SMITH: Item number 4 promises to be one of the longest segments.

MS. WEISS: I think it is highly unlikely that we will get to 5 next week. And assuming that the staff can get the answers to the interrogatories in by next Thursday, that ought to obviate the problem.

CHAIRMAN SMITH: All right.

MR. CUTCHIN: Mr. Chairman, just so the record is straight, they would be to me on Thursday and I will do my best to get it to Ms. Weiss soon thereafter. I said I would get them to her by Friday if at all possible. But either way, it should not be a problem.

CHAIRMAN SMITH: Okay.

Mr. Baxter, do you have something?

MR. BAXTER: No.

CHAIRMAN SMITH: If there's nothing further, then.

MS. WEISS:. One more thing. The Board asked us to make those markings on UCS Exhibit 1, the mockup of the matrix. Those have been done. We have supplied the reporter with the marked copies. We have asked the Applicant, the Licensee and the staff to look those over and to see if they are accurate. And so I would ask them now if they have any problmms with the marked-up copies.

MR. BAXTER: One moment.

MR. CUTCHIN: The staff has not seen it yet, to my knowledge.

CHAIRMAN SMITH: Well, if you haven't seen it, let's take it up -- although we indicated that it's desirable that we have the exhibits in the reporter's hand at week's end, it doesn't have to be.

MS. WEISS: Okay.

CHAIRMAN SMITH: So we'll take it up Monday, because you heard Mr. Cutchin say that he has not seen the comparison -- or Tuesday.

All right, then. We will adjourn until 10:00 a.m. Tuesday.

(Whereupon, at 12:04 p.m., the hearing was adjourned, to reconvene at 10:00 a.m. on Tuesday, November 10, 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: November 7, 1980

Docket Number: 50-289

Place of Proceeding: Harrisburg, Pa.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Alfred H. Ward

Official Reporter (Typed)

Alfred H Ward

Official Reporter (Signature)