UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF RESPONSE TO UNION OF CONCERNED SCIENTISTS
SECOND SET OF INTERROGATORIES

Pursuant to 10 C.F.R. §2.720 and 10 C.F.R. §2.744, the NRC Staff has completed more of its responses to Union of Concerned Scientists (UCS) second set of interrogatories to the NRC Staff dated February 1, 1980. Interrogatories not answered today will be answered by March 31, 1980. Each interrogatory is restated and a response provided. Where appropriate, the NRC Staff has invoked that portion of the Commission's Order of August 9, 1979 (Slip Op. at 11) which allows as an adequate response to a discovery request a statement that information is available in the Local Public Document Rooms and guidance as to where the information can be found. Following the responses to the interrogatories are affidavits identifying the individuals who prepared the responses and verifying them.

Interrogatory 191

Q. With respect to the discussion of Item 1d, "Analysis of Small Breaks" on pp. C1-11 to C1-14, please answer the following:

a. Define "adequate core cooling" as used on line 13 of page C1-12.

A. For loss-of-coolant accidents, adequate core cooling is defined as cooling of the core such that the limits are set forth in 10 CFR 50.46 are not calculated to be exceeded during the course of the accident using an acceptably conservative evaluation model. The limits include a peak cladding temperature below 2200°F, a maximum hydrogen generation of 1% of that theoretically available, and a maximum cladding oxidation of less than 0.17 times the total cladding thickness before oxidation.

However, "adequate core cooling," as used on line 13 of page C1-12 means the existence of two-phase mixture level covering the entire core throughout the duration of a small break loss-of-coolant accident. This condition is more restrictive than the 10 CFR 50.46 limits since no cladding heatup would occur and both hydrogen generation and cladding oxidation would be negligible.

Q. b. Describe the operator actions needed during heat removal by the steam generators and high pressure injection system, as discussed on lines 9-13 of page C1-12.

A. Operator actions are required for very small breaks (area 0.01 sq. ft. or less) with a loss of all feedwater which do not result in automatic initiation of HPI. In this case the operator must manually initiate HPI. With feedwater available he must raise steam generator levels to 95% on the operating range which will

depressurize the system and automatically initiate HPI. For larger break sizes (greater than 0.01 sq. ft.) which result in automatic initiation of HPI, the operator would verify the initiation of the HPI system and manually raise steam generator levels to 95% on the operating range.

Q. c. What is the basis for expecting that the operator would terminate HPI before the PORV or safety valves lift? (line 34 on page C1-12)

A. The basis for this statement was the judgment that, given the control room indications of reactor coolant system temperature and pressure, procedural guidance to the operator utilizing these parameters (subcooling), and previous training, the operator would terminate HPI before the PORV or safety valves open.

Q. d. What are the consequences if the operator fails to terminate HPI before the PORV or safety valves lift?

A. If the operator fails to terminate HPI before the PORV or safety valves open, the PORV (if not blocked) and safety valves will discharge water to the reactor coolant drain tank which is inside the containment. A sustained discharge would cause opening of the tank rupture disc and discharge of water to the containment recirculation sump.

Q. e. Has the staff analyzed a small break with loss of all main feedwater, an isolated PORV <u>and</u> a safety valve stuck open to determine whether this would result in uncovering the reactor core? If so, what were the results? If not, why not?

A. The staff has not analyzed a small break with loss of all main feedwater, an isolated PORV and a safety valve stuck open. However, a similar analysis was performed recently by B&W for a 2772 MWt, lowered loop, 177 Fuel Assembly plant. Assumptions consist of a 0.01 ft^2 break in the cold leg pump discharge piping, no main or auxiliary feedwater to the steam generators, and the PORV actuates and is assumed to stick open. This analysis showed that the core remained covered. A safety valve, larger than a PORV, if stuck fully open, would cause a more rapid depressurization and thus higher HPI flow rates. Analysis of larger breaks up to 0.07 ft^2 (May 7, 1979 generic calculations) shows no core uncovery. Therefore, core uncovery is not expected for the 0.01 ft^2 break and a stuck open safety valve, with loss of all feedwater, assuming manual actuation of two high pressure injection trains or reestablishment of auxiliary feedwater by 20 minutes, if HPI has not been automatically initiated.

Q. f. Identify the specific operator actions that are required for a small break accident, distinguishing between the immediate and follow-up actions. (line 24 on page C1-13)

A. "Small Break Operating Guidelines" were developed by B&W to identify those operator actions that are required based upon the small break LOCA analyses. These guidelines offer general guidance upon which the plant-specific emergency procedures are developed. The TMI-1 procedures have recently been submitted to the staff for review. These guidelines outline the immediate actions as follows:

1. If ESFAS has been initiated due to low RCS pressure, immediately secure all RCPs.

2. Verify control room indications support the alarms received, verify automatic actions, and carry out standard post-trip actions.

3. Balance HPI flow between all injection lines when HPI is initiated. (This should not be required at TMI-1 due to a proposed design modification including cavitating venturis, which is under review.)

4. Verify appropriate steam generator level is maintained.

5. Monitor RCS pressure and temperature. If saturation conditions occur, initiate HPI.

Once the immediate actions have been accomplished to assure adequate core cooling (HPI initiated and steam generator cooling), the operator would complete the followup actions which are directed toward achieving stable cold shutdown conditions. In general, the guidelines outline the following procedure:

1. Maintain HPI unless termination criteria are satisfied.

2. Maintain steam generator levels at 95% on the operating range.

3. Monitor RCS conditions for adequate core cooling.

4. Commence an RCS cooldown by adjusting the secondary system relief valves.

5. Cool down and depressurize the RCS until long-term cooling can be established using the LPI system.

Q. g. For each of the actions described in the answer to subpart (f) above, identify the specific "circumstances" during which the action is required. (line 25 on page C1-13)

A. The guidelines developed for paragraph (f) were structured to include operator actions which would be applicable under all circumstances (i.e., all small break sizes). For breaks which actuate HPI, steps 1, 2, 3, and 4 are required. Smaller break sizes develop more slowly and do not result in reaching the low RCS pressure ESFAS set point. Therefore, irrelevant steps would not be performed. The followup actions would apply to all break sizes.

Q. h. The staff states on line 27 of page C1-13 that immediate operator action is required "as soon as the problem is diagnosed." Is it the staff's position that the amount of time required to diagnose the problem has no bearing on the consequences?

A. Each of the immediate actions in the emergency procedure for small breaks is confirmatory in nature, with the exception of manually tripping the reactor coolant pumps if reactor coolant system pressure decreases to the automatic actuation set point of high pressure injection, and manually increasing steam generator levels from 50% to 95% on the operating range. Since the tripping of the reactor coolant pumps may be required within the relatively short time frame of approximately 2 minutes (depending on break size, see NUREG-0623), a trained and dedicated operator to perform this function is required by IE Bulletin 79-05C to be in the control room at all times. In the longer term this action will be performed automatically. Consequently, it is the staff's position that the amount of time required to diagnose the problem does have a bearing on the consequences.

Raising steam generator level to 95% is performed to enhance natural circulation in the reactor coolant system (see NUREG-0623, Appendix A). Failure to provide any secondary cooling (loss of all feedwater) is analyzed in the May 7, 1979 generic B&W small break analyses. Adequate core cooling is provided by manually initiating two HPI trains or restoring emergency feedwater within 20 minutes. The staff evaluation of these analyses is provided in NUREG-0565.

Balancing HPI flow, listed as an immediate action in the B&W guidelines, should not have to be performed at TMI-1 due to the installation of cavitating venturis in each of the four injection lines. This design modification is discussed in the TMI-1 Restart Report and is under review.

Q. i. For each operator action identified in the answer to subpart (f), above, specify the earliest and latest times after the break during which the operator action must be performed. In other words, give the "window" of time after the break during which the operator action must be taken.

A. For the manual immediate action of tripping the RCPs the "window" of time after the break is 2 minutes to 2 hours, depending on the break size. The basis for this window is discussed in NUREG-0623. For a loss of all feedwater and a very small break (area 0.01 sq. ft. or less), two trains of HPI or emergency feedwater must be manually initiated within 20 minutes (see NUREG-0565).

Q. j. Of the time available to accomplish each necessary operator action, how much of the time is available for diagnosis and how much is available to actually accomplish the action?

A. The "windows" described above include the time for the operator to observe the plant status and to take action. A dedicated and trained operator is required to be present in the control room at all times to trip the reactor coolant pumps. The basis for these windows is discussed in NUREG-0623 and NUREG-0565.

Q. k. For each of the required operator actions, specify the information available to the operator to make the diagnosis and to confirm that the action has been accomplished.

A. Using the same numbering system as in paragraph f, information for each action is noted; followed by the information which would confirm the action has been accomplished. For the immediate actions:

1.a. ESFAS initiated

ESFAS actuation annunciator, low RCS pressure on indicator; response of ESFAS components (e.g., HPI pumps start, discharge valves open, HPI flow established, etc.).

b. Secure RCPs

Turn RCP control switches off; RCP breaker position lights indicate breaker open, RCP motor current reads zero.

2. Verify control room indications

a. Reactor trip annunciator(s); control rod position indicators indicate rods inserted.

b. Turbine trip annunciator; turbine stop valves indicate closed.

c. Letdown isolated; letdown valves closed.

3. Balance HPI flow (see response to part f)

   Throttle HPI discharge valves; HPI flow indicators indicate flow change.

4. Verify steam generator level

   Main or emergency feedwater systems operating; steam generator level indicators showing proper level.

5. Monitor RCS conditions for subcooling

   RCS pressure indicator, cold and hot leg temperature indicators, saturation meter; confirm response as in 1.a if necessary.

For the basic followup actions:

1. HPI operation

   Monitor HPI component operation per #1 of immediate actions.

2. Steam generator level

   Same as #4 of immediate actions.

3. Monitor adequate RCS cooling

   Monitor RCS conditions per #5 of immediate actions; assure subcooling.

4. RCS cooldown

   Steam header pressure controller, turbine bypass valve position; monitor RCS conditions.

5. LPI system operation

LPI pumps start, valves open; confirm LPI flow, RCS conditions.

Q. 1. In assessing whether it is reasonable to expect the operators to take the actions identified and discussed above at the correct time, has the staff considered the other events occurring in the plant which could distract the operator or otherwise demand his attention?

A. Yes.

m. If the answer to subpart (1.) above is "yes," identify all other actions the operator may be required to take on other systems and the alarms that can reasonably be expected to occur during that time.

A. The actions for the operator contained in the procedural guidelines are specifically intended to focus his attention on satisfying RCS inventory and core cooling requirements. It is reasonable to expect that other alarms will occur during and after the transient. However, with the emphasis placed on completing the actions noted in paragraph (f), the operators present in the control room should not be unduly distracted by alarms on systems not required to accomplish the desired core cooling. Operator conditioning to such scenarios in the control room is emphasized as a part of his training program.

196 Describe the events that could lead to "RCS void formation that could interrupt natural circulation flow" (line 35 of page 2-9)

Voiding will occur in the primary coolant system whenever the system pressure drops to the saturation pressure of the hottest coolant in the system. Depressurizing events that are expected to produce primary system voiding are loss-of-coolant accidents and some secondary system overcooling events.

a) Loss-of-coolant accidents

Primary system voiding and the potential for the voids to interrupt natural circulation during loss-of-coolant (LOCA) accidents is only of concern when the steam generators are required to remove some fraction of the decay heat during the accident. For breaks in the primary system above a certain size, the break itself is capable of removing all of the decay heat and the steam generators are not required for decay heat removal (and, in fact, become a heat source to the primary system). For Three Mile Island Unit 1, which is of the 177FA lowered loop class of B&W plants, the break size above which the steam generators are not required to remove decay heat is approximately 0.01 ft^2. For breaks less than 0.01 ft^2, decay heat removal through the steam generators is required to prevent a rise in system energy. These breaks will result in the primary system pressure dropping first to a value slightly above the secondary side pressure (such that the difference between the saturation temperatures of the primary and secondary system is sufficient to remove that

fraction of the decay heat not being removed by the break). The system will then drain by liquid loss through the break, and after steam being generated by boiling in the core fills the upper portion of the reactor vessel, it will begin to travel out the hot leg piping and accumulate at the top of the hot leg U-bend. Once the system drains such that the U-bend is filled with steam and does not allow a continuous flow path to the steam generator, natural circulation and decay heat removal will be temporarily stopped. Since steam will continue to be generated in the core but not condensed by the steam generator, it will begin to pressurize the primary system. However, the system liquid inventory is still decreasing by liquid flow out the break. As such, the volume of steam at the top of the U-bend will continue to increase and the liquid levels in both the steam generator and hot leg vertical piping will continue to drop. Once the liquid level in the steam generator drops to the elevation of the auxiliary feedwater spray spargers, steam will come in thermal contact with the colder auxiliary feedwater and condense. After steam condensation begins, the system pressure begins to drop and two-phase natural circulation is established. In this mode of decay heat removal, the core is cooled by pool boiling with steam condensation in the steam generators.

In summary, void formation during small break loss-of-coolant accidents will result in a temporary interruption of natural circulation in B&W Lowered-loop plants until a condensing surface can be exposed in the steam generator. Once this occurs, two-phase natural circulation will be established.

b) Depressurizing events

For certain events which produce primary system depressurization by overcooling, there is the potential to produce steam voids in the primary system. If these steam voids accumulated at the top of the hot leg U-bends and were of sufficient volume to fill the U-bends, natural circulation could be interrupted.

To bound events which could overcool the primary system, B&W analyzed a 12.2 ft^2 double-ended steamline rupture. This event is beyond the present design bases of the plant since it requires the postulated double failure of both steam generator isolation valves.

The results of this analysis showed that natural circulation flow in the pressurizer loop was temporarily reduced until the core flood tanks injected. However, in the loop without the pressurizer, steam voids were not calculated to occur and decay heat removal by natural circulation was maintained throughout the event.

Based on this analysis, it is not anticipated that less severe overcooling events of more moderate frequency will produce

sufficient steam voids to interrupt natural circulation. Moreover, any reduction in natural circulation flow that might occur for more severe overcooling events should only occur in the pressurizer loop, with natural circulation maintained in the loop without the pressurizer.

Interrogatory 197a & 197c (UCS)

With respect to the discussion on page C2-10 (Status Report on the Evaluation of Licensee's Compliance with the Order, Dated August 9, 1979) please answer the following:

a. Describe the circumstances under which continued operation of engineered safety features would threaten reactor vessel integrity.

b. Does the staff take the position that reliance on operator action to prevent loss of reactor vessel integrity meets the Commission's regulations? If so, reference the specific regulations that permit this.

Response

a. Operation of the HPI system can potentially maintain the reactor coolant system at a pressure equal to the shutoff head of the pumps or the set point of the safety/relief valves. If a temperature reduction has taken place in the RCS, it is possible that the reactor vessel will be in a temperature/pressure regime where a nonductile failure (brittle fracture) could occur. Therefore, it is necessary that the reactor coolant system have its pressure maintained at a value where this is not of concern. These temperature/pressure limitations are described in Appendix G to Section 50 of 10 CFR and the ASME Code Appendix G, "Protection Against Non-Ductile Failure." These limitations are also described specifically in the Technical Specifications for each operating reactor.

b. There is no requirement in the Commission regulations which prohibits the use of the operator in monitoring RCS temperature and pressure, or taking actions, to assure that vessel integrity is not compromised. The conditions which might compromise vessel integrity would result from off-normal conditions and would not be expected to occur early in the event. It has generally been staff policy

to allow intervention of the plant operating staff during an off-normal event if, in studying the effects of an operator error of "omission", there is time to take remedial action. In cases where rapid actuation of engineered safety systems is required, the actuation is required to be automatic and operator independent. Due to the time frame in which operator action is required to assure vessel integrity, the staff feels that operator intervention is acceptable.

198 With respect to NUREG-0623, please answer the following:

a) Identify the non-LOCA transients for which the consequences are aggravated by reactor coolant pump trip and describe the extent of the aggravation? (NUREG-0623, p.1)

As stated in NUREG-0623, the non-LOCA events which could be aggravated are those which produce a primary system depressurization sufficient to require pump trip based on the present pump trip criteria. These are identified as:

- increased secondary system heat removal
- reactor coolant system inventory decrease
- reactor coolant system pressure control malfunction

The event of most concern for B&W plants is the steam line break. For this accident, tripping the reactor coolant pumps could increase the potential for steam voids to form in loops and interrupt natural circulation. To examine this, B&W performed analyses for a 12.2 $ft.^2$ double-ended steam line break (exceeds design basis) and showed that decay heat removal could still be performed. This is described in more detail in the response to interrogatory no. 196.

Another event of concern is the steam generator tube rupture event. The concern is that tripping of the reactor coolant pumps could slow down the rate at which the plant could be depressurized and hence increase the total amount of radioactive water which leaks to the secondary side.

This event however, (steam generator tube rupture) is analyzed in the Safety Analysis Report assuming the pumps are tripped and shows the consequences acceptable.

In general, tripping the reactor coolant pumps produces a loss of the pressurizer sprays, since the sprays are driven by the pressure in the cold leg pipe. During recovery from depressurizing events when the pressure is rising, the availability of the sprays is desirable for pressure control. Under these conditions, the restart of the pumps could be attempted. We have requested in short term item 5 of IE Bulletin 79-05C that the licensee should develop pump restart criteria.

With appropriate restart criteria any aggravation of non-LOCA transients due to pump trip should be minimal.

Interrogatory 198.b

Question

Has the staff evaluated the effect of reactor coolant pump trip during a large break LOCA with respect to the probability of pump (and flywheel) overspeed? If so, provide the analysis and conclusions.

Response

The staff has not completed the generic evaluation of the reactor coolant pump assembly (including the flywheel) overspeed analysis following a large break LOCA. The B&W Topical Report, BAW-10040, provides flywheel structural analysis and pump speed analyses for a range of pipe breaks including the large double ended 8.55 ft^2, cold leg break at the pump discharge. For this latter break size, a maximum pump speed of 276% of rated speed was predicted assuming an immediate pump trip. Prolonging the time for pump trip after the break occurs would reduce this peak overspeed.

The staff's review of this area is being performed in accordance with generic Task Action Plan B-68, "Pump Overspeed During a LOCA." This is an on-going program and an overall resolution of this issue has not been completed.

199 NUREG-0623 states that "small break LOCA's with the pumps operational or with delayed trip can result in more severe consequences than when the pumps are tripped early in the accident." (p. 1) In contrast, the staff states on page C2-18 of the Status Report that "the proposed logic [the automatic pump trip] is intended to preclude pump trip during ... very small breaks where maintenance of forced cooling is desirable." Please answer the following:

a. Specify the spectrum of break sizes and locations where maintenance of forced cooling is desirable and those where pump trip is desirable.

b. Discuss the means by which the operator will be able to determine whether the pump should be tripped or forced cooling should be maintained.

Response to (a):

The spectrum of break sizes where tripping of the pumps early in the accident is necessary to prevent unacceptable consequences was calculated by Babcock and Wilcox to range from about 0.25 ft^2 to 0.025 ft^2.

It should also be noted that for a given break size, a corresponding given period of time in the accident is also calculated to exist in which pump trip produces unacceptable consequences. This is shown in the accompanying figure.

The analyses performed by B&W, upon which the accompanying figure is based, assumed the break was located on the bottom of the cold leg pump discharge piping. Comparable analyses for a spectrum of break sizes assuming breaks at different locations on the circumference of the pipe (e.g., pump suction piping, hot leg piping) were not performed. One sample hot leg break analysis was performed and showed the results to be less severe than an identical analysis with the break in the cold leg.

It is concluded that for break locations other than the bottom of the cold leg discharge piping, the critical region shown on the accompanying figure would be smaller.

Response to (b):

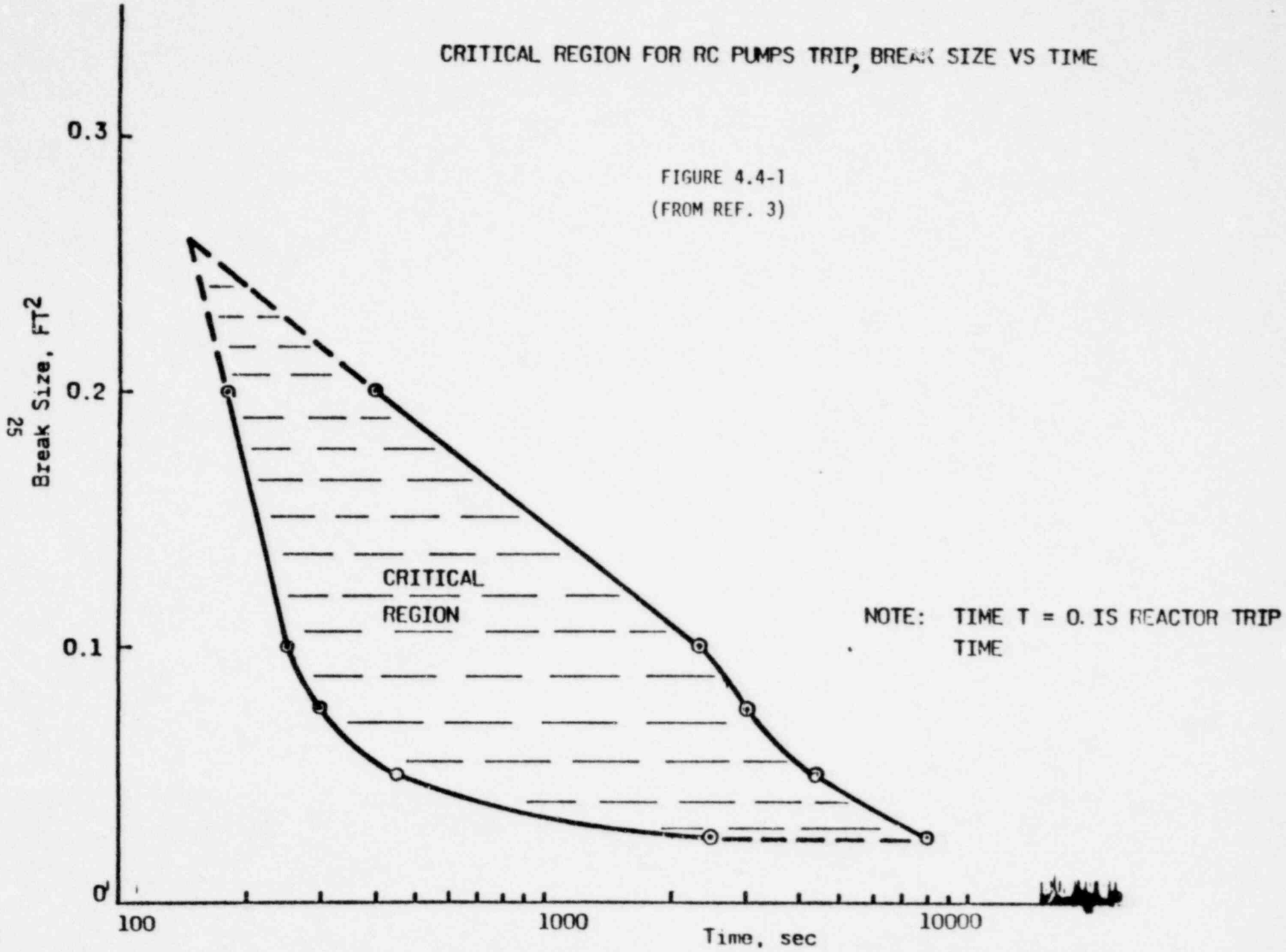
Until the recommendation requiring automatic pump trip is adapted by the Commission, the emergency procedures require the operator to manually trip all of the reactor coolant pumps upon reactor trip and actuation of the High Pressure Injection (HPI) System on low pressure. This criteria was chosen because HPI actuation on low pressure will occur for all breaks within the critical region and provide an easily recognizable signal for the operator. Moreover, the characteristics of small breaks are such that break sizes and locations which would permit continued pump operation cannot be distinguished by the operators from break sizes

and locations which do require pump trip. Finally, the critical region on the accompanying figure, within which early pump trip is required, is based only on analysis. The staff has pointed out in NUREG-0623 that the analysis models used to produce these results are presently unverified and could have a large uncertainty associated with them.

Consequently the critical region shown on the accompanying figure also has an uncertainty associated with both critical break size as well as the critical pump trip time.

CRITICAL REGION FOR RC PUMPS TRIP, BREAK SIZE VS TIME

FIGURE 4.4-1
(FROM REF. 3)



Interrogatory 204 (UCS)

On page C8-22 the staff states that it "has determined that post-accident operation of the reactor coolant pumps is highly desirable." Please answer the following:

a. Identify the particular accidents for which reactor coolant pump operation is highly desirable.

b. Describe in quantitative terms the difference in consequences for each of the accidents identified above assuming first that the reactor coolant pumps are operating and second that they are not operating. If no detailed evaluation has been done for the case where the pumps are operating, provide you (sic) estimate.

c. Has the staff evaluated whether classifying the reactor coolant pumps as safety related and providing an on-site power supply (or any other means of providing forced cooling of the core following an accident) would provide substantial, additional protection for the public? If so, what were the results of the evaluation? If not, why not?

Response

a. & b. From a heat transfer perspective, forced circulation cooling by the reactor coolant pumps is the preferred means of removing heat from the reactor core in all situations. This is because the high flow rates allow decay heat to be removed with a smaller primary/secondary temperature difference that would occur from natural circulation. The quoted statement above is referring to the reinitiation of seal injection flow to the pumps so that they are available if wished after an accident. Though pump availability is desirable natural circulation has been shown to be efficient for normal and accident core heat removal. However, even though forced circulation cooling by the reactor

coolant pumps provides a more direct heat transfer [illegible] operation of the pumps during a small break LOCA results in greater inventory loss and a high void fraction early in the event. Adequate cooling of the core can be maintained if the pumps continue to operate. However, analysis indicates that if the pumps are stopped after a high void fraction is obtained, liquid that was previously dispersed around the primary system through pumping action would then collapse down to the low points of the primary system. This could result in uncovery of the reactor core and insufficient core cooling. Since continuous pump operation could not be assured, it was deemed prudent to trip the reactor coolant pumps whenever a small break LOCA is suspected. Analysis indicates that this action assures that decay heat can be removed from the reactor core by natural circulation following a LOCA.

The quantitative results of analysis performed to assess the effects of reactor coolant pump trips during a small break LOCA are presented in NUREG-0623.

c. Since analysis has shown that the reactor coolant pumps are not needed to assure adequate core heat removal, for any normal or accident situation, there are currently no plans to designate them safety related or require onsite power sources. However, the staff has acted to require that automatic systems be provided for tripping the coolant pumps upon indication of small break LOCA's. Until these systems are installed, the staff has required changes to operating plant emergency procedures which call for manual operator tripping of the pumps upon low reactor coolant system pressure.

Interrogatory 205

With respect to the discussion of isolation of the reactor coolant pump seal injection lines on pages C8-22 and C8-23, please answer the following:

a) Describe the evaluations the staff has done to determine whether the health and safety of the public is better protected by not automatically isolating the seal injection lines or by isolating them.

b) What is the staff's judgment concerning the probability of a loss-of-reactor-coolant if the seal injection lines are isolated? Consider in your answer the probability of loss of off-site power, the new procedure for the operators to trip the reactor coolant pumps and the addition of a means to automatically trip the reactor coolant pumps.

c) What information is available to the operator to indicate the need to manually isolate the seal injection lines?

d) In approving a design which does not provide for automatic isolation of the seal injection lines, did the staff consider the financial consequences of damges to the reactor coolant pumps?

Answer

b) TMI Unit 1 has Westinghouse designed reactor coolant pumps. The seals of the Westinghouse design are cooled by injection of coolant from the charging system. A second source for cooling the seals is a thermal barrier heat exchanger which utilizes the nuclear services cooling water system to cool the primary reactor coolant which in turn provides cooling to the seals if seal injection flow is not available. If isolation or failure of the seal injection system should occur, Westinghouse has stated that the thermal barrier cooling sytem will adequately cool the seals to prevent a loss of coolant accident. Since both the nuclear services cooling water system and the charging system are powered from the emergency diesels, if a loss of off-site power were to occur, seal cooling should not be affected. The staff has not accepted the Westinghouse assessment that availability of either system will adequately cool the seals. As a result Met-Ed will be required to evaluate the possibility and impact of seal damage and leakage assuming only one of these cooling systems is available. If seal damage cannot be precluded, the licensee will have to provide an analysis of the limiting small break LOCA with subsequent pump seal failure.

The new procedure for the operators to trip the reactor coolant pumps and the addition of a means to automatically trip the reactor coolant pumps (RCP) will not increase the probability of a loss of coolant accident. Cooling to the seals is maintained on a RCP trip and since the pump shaft is no longer rotating, the potential for damage to the seal faces and for a LOCA is actually reduced.

c) There is no information directly available in the control room to indicate the need to manually isolate the seal injection lines. Only if a break occurred in a seal injection line would isolation be desirable. If such a break occurred in this line, the containment building sump monitoring system would detect the leak and if it exceeded 1 gpm (which would be the case for any break or any break or crack of significance), the plant would have to be shutdown and the inside of the containment inspected for the source of the leakage.

d) Isolation of the seal injection lines will cause damage to the RCP seals, not vice versa, as stated in the question. Prevention of a LOCA due to seal failure is the staff's prime concern and maintaining seal integrity through cooling is the most acceptable way of achieving this goal. This also prevents expensive damage to the RCP's from occurring.

Interrogatory 212

Question

With reference to the staff's answer to UCS Interrogatory #46, does the staff agree that a break in the top of the pressurizer (or the pipes connected thereto) could have the same effect as an unisolated, stuck-open PORV? If the answer is "yes," identify which of the short- and long-term measures are addressed to mitigation of that postulated event.

Response

The staff believes that a break in the top of the pressurizer or in the pipes connected thereto would have essentially the same effect as an unisolated, stuck-open PORV if the break size and shape were similar to that of a PORV. The only short- or long-term measures required by the staff which should have a significant impact on a break in the top of the pressurizer are procedure changes and operator training. LOCA procedures have been modified to require RCP trip on a reactor trip in conjunction with an SIS due to low pressure. A discussion of indication of inadequate core cooling and steps to mitigate inadequate core cooling are to be included in applicable emergency procedures. High pressure injection pump termination criteria have been included in LOCA procedures. Guidelines and procedures have been developed and implemented for instituting and maintaining natural circulation cooling of the reactor core. TMI-1 operators must receive additional training on the B&W simulator on TMI-2 type accidents and all TMI-1 operators will be fully reexamined and qualified by the staff prior to TMI-1 startup. The operators are to be trained to follow operator actions designed to prevent formation of voids in the primary system which are large enough to

compromise adequate core cooling capability. The following documents were used in preparing this response: Commission Order concerning Met-Ed dated August 9, 1979; IE Bulletins 79-05A and 79-05B; and NUREG-0578.

Respectfully submitted,

Lucinda Low Swartz

Lucinda Low Swartz
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 17th day of March, 1980

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF BRUCE BOGER

I, Bruce Boger, being duly sworn, do depose and state:

1. I am employed by the Operator Licensing Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.

2. The answer to UCS Interrogatory 191 was partially prepared by me. I certify that the answer given is true and accurate to the best of my knowledge.

Bruce A. Boger
Bruce Boger

Subscribed and sworn to before me this , day of March 1980.

Notary Public

My Commission expires: July 1, 1982

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(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF SCOTT F. NEWBERRY

I, Scott F. Newberry, being duly sworn, do depose and state:

1. I am employed by the Reactor Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.

2. The answer to UCS Interrogatory 191 was partially prepared by me. I certify that the answer given is true and accurate to the best of my knowledge.

Scott F. Newberry
Scott F. Newberry

Subscribed and sworn to before me this 17 day of March 1980.

Milam M. Groff
Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF BRIAN W. SHERON

I, Brian W. Sheron, being duly sworn, do depose and state:

1. I am a Principal Nuclear Engineer in the Division of Operating Reactors, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for performing the safety review of assigned nuclear power plants, including Three Mile Island, Unit 1 Restart Program.
2. The answers to USC' Interrogatories 191a, 196, 198a, and 199 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Brian W Sheron
Brian W. Sheron

Subscribed and sworn to
before me this 1 day of
March, 1982.
Marian M Groff
Notary Public
My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, *et al.*	)	Docket No. 50-289
	)	
(Three Mile Island, Unit 1)	)	

AFFIDAVIT OF MARK P. RUBIN

I, Mark P. Rubin, being duly sworn, do depose and state:

1. I am employed by the Reactor Systems Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.
2. The answers to UCS Interrogatories 197a, 197c, 204 and Sholly 6-2 were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Mark P. Rubin

Mark P. Rubin

Subscribed and sworn to before me this 14 day of March 1980.

Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF ALGIS J. IGNATONIS

I, Algis J. Ignatonis, being duly sworn, do depose and state:

1. I am employed by the Reactor Systems Branch, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission.
2. The answer to UCS Interrogatory 198b was prepared by me. I certify that the answer given is true and accurate to the best of my knowledge.

Algis J. Ignatonis
Algis J. Ignatonis

Subscribed and sworn to before
me this 4 day of March 1980.

Micah M. Groff
Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF BYRON L. SIEGEL

I, Byron L. Siegel, being duly sworn, do depose and state:

1. I am a senior reactor engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing accident analyses and ECC and RHR systems of assigned nuclear plants.

2. The answers to Union of Concerned Scientists Interrogatory 205, parts b-d were prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Byron L. Siegel

Byron L. Siegel

Subscribed and sworn to

before me this 7 day of

March, 1980

Notary Public

My Commission expires: July 1, 1982

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF GLENN B. KELLY

I, Glenn B. Kelly, being duly sworn, do depose and state:

1. I am a reactor engineer in the Division of Systems Safety, Office of Nuclear Reactor Regulation of the United States Nuclear Regulatory Commission. I am responsible for reviewing accident analyses, and ECC and RHR systems of assigned nuclear power plants.
2. The answers to UCS' Interrogatory 212 was prepared by me. I certify that the answers given are true and accurate to the best of my knowledge.

Glenn B. Kelly

Subscribed and sworn to
before me this 13 day of
March, 1981

Notary Public

My Commission expires: July 1, 1982