UCS 2/23/83

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

Union of Concerned Scientists'

REQUEST FOR SUBPOENAS

Pursuant to 10 CFR Section 2.720(h)(2)(i), the Union of Concerned Scientists requests that the Board issue subpoenas requiring the attendance and testimony of C. J. Heltemes, Jr. and Frank H. Rowsome of the NRC Staff at the reopened hearing in the above-captioned proceeding, commencing March 7, 1983, in Bethesda, Maryland. Exceptional circumstances warranting the issuance of these subpoenas are present, as discussed below.

In an April 29, 1982 memorandum entitled "Reliability and Effectiveness of 'Feed and Bleed' Core Cooling at TMI-1", Harold Denton, Director, Office of Nuclear Reactor Regulation requested the Staff to explain the technical basis for its position on 'feed and bleed' in the TMI-1 restart proceeding and to "clarify" the difference between the "feed and bleed" and "boiler-condenser" modes of core cooling. A copy is attached. The final Staff report responding to Mr. Denton's request is entitled "Report on NRC Staff Position on Feed and Bleed Cooling." A copy of that report, with a cover memorandum dated July 1, 1982, is also attached.

In the course of preparing its response to Mr. Denton's request, various members of the Staff commented on a draft report. C. J. Heltemes, Jr., Deputy Director, Office for Analysis and Evaluation of Operational Data, is the author of a memorandum dated June 10, 1982, entitled "Draft Report on NRC Staff Position on Feed and Bleed Cooling at TMI-1 Restart Hearing." A copy is attached. The Heltemes memo constitutes the comments of the Office for Analysis and Evaluation of Operational Data (OAEOD) on a draft of the July 1, 1982 final report to Mr. Denton.

Mr. Heltemes' memo contains material which contradicts the NRC Staff Testimony of Brian W. Sheron and Walton L. Jensen, Jr. filed February 16, 1983. In particular, Sheron and Jensen state at page 6:

> The Staff has concluded that the heat transfer mechanisms involved in the boiler-condenser process are adequate to remove decay heat from the reactor system and will prevent core uncovery if at least one train of ECCS is operable. This conclusion is based on both the B&W CRAFT-2 calculations and the RELAP-4 audit calculations, as well as our evaluations of the heat transfer mechanisms involved in the process and discussed in commonly available heat transfer texts. Although detailed reactor coolant system behavior during the period of natural circulation interruption in the analysis of certain small break sizes is not well understood, the system must eventually drain down and a steam condensing surface in the steam generator would be exposed before the core could begin to be uncovered. Once a steam condensing surface were uncovered, boiler-condenser natural circulation would commence and depressurize the system so that the decreased break flow, along with the increased HPI flow, would result in a net inventory increase in the primary system before the core could begin to uncover. The Staff has evaluated the mechanism involved in the boiler-condenser heat transfer process and has concluded that the condensing surface that would be available would be capable of removing all decay heat generated by the core if an adequate supply of feedwater were available.

In contrast, Heltemes states on page 2:

> We believe that the conclusion
>
> > "If the feed and bleed process discussed above was insufficient to remove decay heat, natural circulation would be established in the boiler/condenser mode"
>
> is not a certainty, especially in the absence of experimental data for B&W plants. In the event that, for any reason, natural circulation cannot be established and the primary coolant pumps are not available, the "feed and bleed" mode of decay heat removal would have to be used.

In addition, Mr. Heltemes points out in paragraph 5 that the emergency procedures "are not presently in place" and believes "it is important to provide a sense of timing regarding what is in place and available now (in terms of equipment, procedures, and training) and what is likely to be available at some specified time in the future." As the Board knows, it is UCS's position that adequate emergency procedures must be in place before it can be found that either boiler-condenser or bleed and feed are sufficient means of decay heat removal. The Staff testimony deals not at all with the subject of emergency procedures; the cooling modes are treated in the abstract. It is apparent that AEOD recognizes the significance of procedures to the question of decay heat removal reliability.

Finally, Mr. Heltemes states in paragraph 7:

> We agree with the need for obtaining experimental verification of the analytical code predictions. <u>We believe that this section of the report should be expanded to clarify the items for which verification is considered appropriate or necessary</u>. In this regard, consideration should be given to (a) natural circulation in B&W plants, including establishment of boiler/condenser operation, and elimination of steam formations in the hot legs; and (b) the ability of existing PORV and safety valves to perform reliably in a 'feed and bleed' mode. (emphasis added)

While the memo itself is terse, there is a clear indication here that experimental verification of the code predictions is necessary before they can be relied upon. This should be contrasted with the Staff's testimony which concludes that, despite the lack of verification (or "confirmation," as the Staff would have it), their conclusion that adequate core cooling will not be jeopardized is unchanged. (NRC Staff Testimony of Brian W. Sheron and Walton L. Jensen, Jr. at 17.)

It appears that AEOD did not concur with the NRC Staff position on feed and bleed for TMI-1. In the cover memo from Roger J. Mattson and Hugh L. Thompson to Harold Denton, enclosing the "Report on NRC Staff Position on Feed and Bleed Cooling," the authors note that the report was prepared by the Division of Systems Integration (the Division to which witnesses Sheron and Jensen are attached) that the Division of Human Factors Safety "concurs" in certain parts and that AEOD has "reviewed this response and their comments have been considered." UCS has reviewed the final report given to Mr. Denton together with the Heltemes memo. While the AEOD opinions may have been "considered", they were not incorporated. While we are not able to tell the Board exactly what Mr. Heltemes will testify since we do not have access to him, the memo confirms that AEOD has a different perspective and opinion on the certainty of boiler-condenser.

Frank H. Rowsome, Deputy Director, Division of Risk Analysis, RES[1/], has previously appeared as a witness for the Staff in this proceeding and is the author of a report entitled "Feed and Bleed Issue for CE Applicants," January

1/ We understand that Mr. Rowsome has been reassigned, but is still a member of the Staff.

29, 1982, a copy of which is attached. The material, which consists primarily of risk assessment, contains the following conclusions of greatest importance to this proceeding (at pages 7-8 of the report):

> The value of an assured feed and bleed capability here is to eliminate the need for feedwater. This would eliminate the smaller (10^{-6}/yr) path to core melt without affecting the more prominent path via HPI failure. Note that <u>small LOCA with total HPI failure is predicted to result in a core melt frequency above the Commission goal for all core melts</u>. The provision of feed and bleed capability or of an improved AFW system will not help this. <u>It is a problem generic to PRWs and not unique to the CE designs</u>. It <u>appears that the high frequency of very small LOCA revealed by historical experience and the marginal HPI system reliabilities revealed by many PWR PRAs are combining to yield unacceptable core melt frequencies through S_2D-type sequences</u>. We suggest that NRR tackle this problem in two ways: First, a serious effort should be made to reduce the frequency of S_2 LOCA's. Second, a broad-scale attack on HPI reliability problems comparable to that instituted for AFW systems after TMI should be initiated for all PWR's.
> (emphasis added)

These conclusions, which are said by Mr. Rowsome to apply to all PWRs, go to the heart of the issue in this proceeding: the adequacy of decay heat removal. Neither the existence of this report nor its conclusions have been brought to the Board's attention by the Staff.

10 CFR 2.720(h)(2)(i) provides that the attendance and testimony of named NRC employees may be ordered by the presiding office "upon a showing of exceptional circumstances, such as a case in which a particular named NRC employee has direct personal knowledge of a material fact not known to the witnesses made available" by the Staff.

This has been held to authorize the subpoena of Staff personnel who hold differing opinions (as opposed simply to knowledge of particular "facts") on critical safety issues. <u>Pacific Gas and Electric Co</u>. (Diablo Canyon Nuclear Power Plant, Units 1 and 2), ALAB-519, 9 NRC 42 (1979). In that case, the Appeal Board stated:

> The ability of nuclear power plants to withstand earthquake damage is undeniably crucial in California, where seismic phenomena are not uncommon. The Board, the Staff, the applicant, and amicus curiae have all allowed the procedural undergrowth to obscure the substantive forest. This is more than a run-of-the-mill disagreement among experts. We have here a nuclear plant designed and largely built on one set of seismic assumptions, an intervening discovery that those assumptions underestimated the magnitude of potential earthquakes, a reanalysis of the plant to take the new estimates into account, and a post hoc conclusion that the plant is essentially satisfactory as is--but on theoretical bases partly untested and previously unused for these purposes. We do not have to reach the merits of those findings to conclude that the circumstances surrounding the need to make them are exceptional in every sense of that word. Subpoenas to compel the testimony of the two ACRS consultants whose views diverge from the consensus just described are therefore not only permissible under the Rules of Practice, but appropriate.
>
> 9 NRC at 46, emphasis added.

The situation here is markedly similar.[2/] This Board is addressing complex technical issues of first impression and, after the discovery of the EG&G test results, is now being presented with new analyses and opinion from the Staff, largely unverified by experimental data, which have never been presented to any NRC Board, so far as we are aware. Harold Denton's request that he be "informed" of the technical basis for Staff position on bleed and feed for TMI-1 is evidence of the ad hoc nature of the conclusions being

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2/ Mr. Rowsome's case is even clearer. He is possessed of "facts", i.e. risk assessment calculations, which are not incorporated or alluded to in the Sheron or Jensen testimony.

offered to the Board here. It is incumbent upon the Board to consider the divergent views of qualified personnel. This is particularly so because the Staff is at this point in the position of defending its previous judgments.

Respectfully submitted,

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Dated: February 23, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "UNION OF CONCERNED SCIENTISTS' REQUEST FOR SUBPOENAS" have been served on the following persons by deposit in the United States mail, first class postage prepaid, this 23rd day of February 1983.

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

April 29, 1982

MEMORANDUM FOR: Roger Mattson, Director, DSI
Hugh Thompson, Director, DHFS

FROM: Harold R. Denton, Director, ONRR

SUBJECT: RELIABILITY AND EFFECTIVENESS OF "FEED AND BLEED" CORE COOLING AT TMI-1

As we have discussed, questions have been raised recently which center around the staff's position on the reliability and effectiveness of "feed and bleed" as a core cooling technique following a SBLOCA. Specifically, the staff's technical basis for its position on "feed and bleed" at the TMI-1 restart hearing has been questioned. In order for me to be fully informed on this issue, I would like a report which includes the following:

(1) A description of the staff position at the TMI-1 restart hearing on the role of "feed and bleed" during a SBLOCA.

(2) An interpretation of the TMI-1 Licensing Board decision regarding the need for reliable and effective "feed and bleed" during SBLOCA.

(3) A detailed explanation of the staff's technical basis for its position on "feed and bleed" at TMI-1. Include an assessment of existing information and ongoing work, both within the staff and by the industry. Also, clarify the difference between the "feed and bleed" mode of cooling and the "boiler-condenser" mode of cooling.

(4) Recommendations for future NRC and/or industry actions needed to move towards a better understanding of the reliability and effectiveness of the "feed and bleed" technique.

Since the results of your work should be coordinated with RES, AEOD and OELD, I suggest we hold a meeting as soon as you can compile preliminary information on this subject. I expect a report prior to the date for filing our response to exception in the TMI proceeding.

HR Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

cc: W. Dircks
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8205190149

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

JUL 1 1982

MEMORANDUM FOR: Harold Denton, Director, Office of Nuclear Reactor Regulation

FROM: Roger Mattson, Director, Division of Systems Integration

Hugh Thompson, Acting Director, Division of Human Factors Safety

SUBJECT: NRC STAFF RELIANCE ON "FEED AND BLEED"

As requested in your memorandum of April 29, 1982, we have prepared the attached report addressing each of the four issues which you identified. To summarize, the NRC staff did not rely on "feed and bleed" cooling to protect the core at TMI-1. This position was made clear to the board. Babcock and Wilcox performed feed and bleed analyses for the development of inadequate core cooling procedures. Such procedures would be utilized as defense in depth for events beyond the design basis. These procedures instruct the operator to establish and maintain feed and bleed cooling following a complete loss of heat sink until feedwater can be restored.

This response was prepared by the Divison of Systems Integration. The Division of Human Factors Safety concurs in the statement regarding the reliance that we place on operator actions for initiation of emergency feedwater and on feed and bleed cooling in emergency operating procedures for accidents beyond the design basis. The offices of ELD and AEOD have reviewed this response and their comments have been considered.

We recommend you consider informing the TMI-1 Appeal Board of this staff analysis of the Licensing Board's decision along with our conclusion that our areas of disagreement are not material.

Roger J. Mattson
Roger J. Mattson, Director, DSI

Hugh L. Thompson Jr.
Hugh L. Thompson, Jr. Acting Director, DHFS

Enclosure: As stated

cc: see next page

JUL 1 1982

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REPORT ON NRC STAFF POSITION ON

FEED AND BLEED COOLING

Item 1 <u>A description of the staff position at the TMI-1 restart hearing on the role of "feed and bleed" during a SBLOCA</u>

RESPONSE

The staff's position at the hearing was that feed and bleed cooling is not relied on for heat removal. This position was made clear to the ASLB in the TMI-1 restart hearing in (1) written testmony by NRC staff witness J. Wermiel and (2) oral testimony of W. Jensen as follows.

(1) <u>Written Testimony of J. Wermiel in Response to Board Question 6:</u> Question 6i. Will the reliability of the emergency feedwater system be greatly improved upon conversion to safety-grade, and is it the licensee's and staff's position that the improvement is enough such that the feed-and-bleed backup is not required.

(Witness Wermiel)

Response: Based on knowledge of the improvement in reliability gained by eliminating first order failure sources, it is the staff's judgment that the reliability of the emergency feedwater system will be improved once the fully safety-grade system is installed. The single failure problem associated with integrated control system/non-nuclear

instrumentation described in the response to 6a and b above will be eliminated. In addition, various other hardware, procedural and administrative improvements as identified in the TMI-1 Restart SER, NUREG-0680 under Order Item 1a should enhance emergency feedwater system reliability. However, a quantitative reassessment of the reliability of the fully safety-grade EFW system has not been performed. The feed-and-bleed back-up is not required by the staff and, therefore, need not meet all requirements of a safety system. However, it is recognized as additional defense in depth for providing core cooling in the very unlikely event that emergency feedwater is lost, and the HPI pumps and primary safety valves which comprise the feed and bleed mode are required to be available by Technical Specifications.

(2) <u>Oral Testminony of W. Jensen Regarding UCS Contentions 1 and 2</u>

(Dr. Jordan) I would address the question then directly to Mr. Jensen. Did I misstate what you said? Do you believe that the high pressure injection system is important in that it not only supplies emergency cooling inventory but it also removes heat in the feed and bleed mode? That that is an important safety feature?

(The Witness) The high pressure injection system is an important safety feature for making up the coolant lost from a

small break LOCA. The NRC does not rely on this system for heat removal in the feed and bleed mode by which core decay heat would be forced through the safety valve or the PORV. Instead, we rely on the heat removal from the emergency feedwater system.

(Dr. Jordan) Okay. That's fine.

(Ms. Weiss) If I can refer, Dr. Jordan, I think the exact question you are asking is answered on page 9 of the staff testimony in response to Board question number 6. I was going to read the sentence to you. (Wermiel testimony above)

The feed and bleed back up is not required by the staff and therefore need not meet all the requirements of the safety system. It's just simply a direct quote.

(Dr. Jordan) Yes. I remember that and thank you for pointing that out. I think that clears up the matter."

Item 2 <u>An interpretation of the TMI-1 Licensing Board decision regarding the need for reliable and effective "feed and bleed" during SBLOCA</u>

RESPONSE

There is an interest in whether the ASLB accepted the staff position on the reliance to be placed on feed and bleed cooling. We believe that the ASLB did not accept our position, regarding emergency feedwater reliability, as shown in the following excerpts from its decision. We believe however that the board did not err in declining to find that additional modifications to the emergency feedwater system are necessary at TMI-1 prior to restart.*

Page 224 of the TMI-1 Licensing Board decision acknowledges the NRC Staff position (see Item 1 above) by noting that:

"The Staff's position is that the loss of emergency feedwater following a main feedwater transient is not an accident which must be protected against with safety-grade equipment."

To us, this observation by the ASLB says that our position in Item 1 above was understood by the Board. At Page 242 of the decision the Board goes on to point to a precedent ruling made by the St. Lucie-2 Appeal Board for requesting additional reliability numbers from the staff. The TMI-1 Board noted that:

*NRC response to UCS' exceptions to the PID, filed with the Appeal Board in the TMI-1 Restart proceeding May 20, 1982.

"The (St. Lucie) Appeals Board decided that measures were required to mitigate such an event** should it occur. We believe that similar measures are necessary at TMI-1; that the reliability of the EFW system has not been demonstrated to be adequate by itself. However, the EFW system is backed up by the high pressure injection system, so that in the event of failure of the EFW system the core can be cooled by feed and bleed while repairs are being made to the EFW system."

We conclude from this statement that the TMI-1 Board has relied upon the availability of feed and bleed in reaching its finding that the TMI-1 design is acceptable. The question then is how the Board reached this conclusion in light of the Staff position (Item 1 above). The answer is summarized on page 250 of the TMI-1 Board decision where the Board states:

"We have relied on the staff figures on reliability of the EFW system <u>and our own estimates</u> (emphasis added) of the adequacy of the feed-and-bleed backup to arrive at our conclusion that the core is adequately protected from a loss of main feedwater transient, the dominant challenge to the EFW system."

**Complete loss of all AC power including both diesel generators.

We conclude that the Licensing Board reached the same conclusion as the staff (the TMI-1 design satisfies the Commission's regulations), although the board's basis for the conclusion is different. The basis for the staff position is summarized in Question 3 below. We have studied the Licensing Board decision to understand the basis for its conclusion. At paragraph 1056 we find the following:

"Since the EFW System is backed up by a safety-grade HPI, designed to protect the core in the event of a small break LOCA, we believe we can conservatively assume an additional safety factor of 100, or an overall probability of failure to protect the core of about 10^{-6}/yr. Lacking any demonstration that the above failure probabilities are grossly in error, we conclude that the EFW system, as modified, will, with the HPI backup, adequately protect the health and safety of the public."

During the TMI-1 hearing, the NRC Staff did not provide any detailed discussion, for or against, the above Licensing Board assessment. We do not have sufficient information regarding the uncertainties associated with of feed and bleed cooling to credit it with a 100 fold reduction in the probability of core melt.

Item 3a <u>A detailed explanation of the staff's technical basis for its position on "feed and bleed" at TMI-1.</u>

RESPONSE

It was the Staff's position during the TMI-1 hearing that the emergency feedwater (EFW) system is required to be available for decay heat removal in feedwater transients and certain small break loss-of-coolant accidents without feedwater. We also noted that should EFW be initially unavailable, there is at least 20 minutes time available to take action to establish EFW flow prior to uncovering of the core following a loss of main feedwater or certain small break loss of coolant accidents. The TMI-1 EFW system will, at the time of restart, meet the Commission's requirements for safety related equipment, in the event of small break LOCA and/or loss of main feedwater if credit for operator action is given (to initiate the system) within 20 minutes. The TMI-1 EFW system will be fully automatic for these events by the first refueling outage after restart. The staff recognizes that a feed and bleed capability exists at TMI-1 to provide additional defense in depth for decay heat removal should EFW fail. The inadequate core cooling procedures at TMI incorporate the feed and bleed process. Operators are trained in the use of these procedures at TMI-1 and feed and bleed is covered in the scope of OLB examinations of the TMI operators. It is usually covered in the simulator portion of the examination. Safety grade equipment to accomplish feed and bleed backup to EFW in the event of a complete loss of all feedwater is not required to be included within the design

basis since the EFW system at the time of restart is sufficiently reliable to make a postulated loss of EFW system acceptably low.

Item 3b <u>Clarify the difference between the "feed and bleed" mode of cooling and the "boiler/condenser" mode of cooling</u>

RESPONSE

For small breaks below a certain size, the break area is not large enough to relieve all the energy generated by decay heat. For this condition, heat transfer through the steam generator is the preferred method of providing additional required energy removal capability. To accomplish this, emergency or auxiliary feedwater systems must be operating. Since the reactor coolant pumps are tripped for most small breaks, coolant flow through the core is by natural circulation. Feed & Bleed is a method by which decay heat is removed from the primary system if no feedwater were available so that natural circulation did not occur. The "boiler/condenser" mode of cooling is one of three modes of natural circulation cooling discussed below. Each mode represents a progressively degraded condition of the primary system in terms of system inventory. Thus it is possible for some small break scenarios to experience all three modes of natural circulation heat removal. In small break LOCA

calculations by B&W* temporary interruption of all modes of natural circulation was predicted however, inventory loss in these three modes is not sufficient to cause extended core uncovery and fuel damage. It is not necessary that the primary system be refilled folllowing a LOCA in order to adequately cool the core. Analyses by B&W indicate that adequate decay heat can be removed under any of the following three natural circulation modes.

1. Single phase - In this mode the entire primary system remains in a subcooled liquid state. Core flow is maintained solely by density differences between hot and cold liquid.

2. Two phase continuous - This mode is similar to mode 1 except that the hot side is at saturation and at low steam quality. Bubbles are formed in the upper portion of the core and are swept, as part of a continuous two phase mixture, into the steam generator and condensed. During this time, some of the steam generated in the core will rise into the upper head and accumulate there as a single large bubble. For B&W plants this heat removal mode will persist until the liquid level drops below the hot leg U-bend.

*B&W report "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177-FA Plants" May 7, 1979.

3. Boiler/Condenser - When the hot leg U-bend is voided, liquid will not be carried into the steam generator. However, when sufficient steam has accumulated from boiling in the core such that a condensing surface is exposed within the steam generator tubes, heat will be removed by steam condensation on the tube walls. This method of heat removal is referred to as boiler/condenser. Thus a period will exist between formation of a bubble in the hot leg U-bends when mode 2 natural circulation is lost, and the uncovering of the steam generator condensing surface, during which no natural circulation would exist in B&W plants. The condensing surface is at a higher elevation than the core so that boiler/condenser natural circulation will be established in the event of a small break LOCA before the core could be uncovered. Boiler condenser natural circulation was demonstrated to be effective in LOFT* and Semiscale** experiments for U-tube steam generators.

*NUREG CR-1570 "Experimental Data Report for LOFT Nuclear Small Break Experiment L-3-7", August 1980.

**EGG-SEMI-5507 "Quick Look Report for Semiscale Mod-2A Test S-NC-2," July 1981.

If heat removal through the steam generator cannot be achieved due to loss of all feedwater (an event not required to be considered as a part of the design basis), "feed and bleed" can be used as an alternate heat removal method. The procedure involves energy removal by venting hot water and/or steam through the primary system PORVs and/or safety valves (bleeding), and replacing the vented coolant with cold HPI water (feeding).

Item 3c <u>Assessment of Current Status and Existing Information on "Feed and Bleed"</u>

RESPONSE

As you recall, in a recent communication to Dr. Henry Myers we noted that for a small break LOCA which is subsequently isolated, a phenomenon similar* to "feed and bleed" might ultimately occur as the means of decay heat removal if steam bubbles were trapped at the top of hot legs and did not rapidly condense even if emergency feedwater were available. This method of heat removal from the primary system might occur if the core were sufficiently cooled so that decay heat no longer boiled the incoming HPI water but forced it through

*The term "similar" is used, since in this case feedwater to the secondary side of the steam generator is assumed available, and no operator actions are assumed to initiate decay heat removal via the safety valves.

the safety valves as liquid. If boiling occurred in the core, the steam production would act to increase the bubble size in the hot-leg U-bends. If the hot leg bubble size increased sufficiently, a condensing surface on the steam generator tubes would be exposed. This would establish natural circulation in the boiler/condenser mode.

The bubbles could not expand sufficiently to uncover the core or to exhaust steam out of the pressurizer since the secondary system water level in the steam generators would be above the core and the pressurizer surge line entry elevation. Although our study of this scenario is recent and was not discussed during the TMI-1 hearing, no additional staff reliance on feed and bleed should be implied since if the feed and bleed process discussed above were insufficient to remove decay heat, sufficient coolant loss through the safety and relief valves would eventually reestablish natural circulation in the boiler/condenser mode. The letter to Dr. Myers is attached for further information on these recent developments.

All three PWR suppliers are developing emergency procedure guidance to licensees on how to use equipment to perform "feed and bleed" operations as a backup method of heat removal if all measures for feeding steam generators are lost. It is important to stress that at this time "feed and bleed" is not a preferred method of decay heat removal. The equipment used for feed and bleed operation was not designed for that purpose. Feed and bleed is only one possible emergency alternative for primary system heat removal for events beyond the design basis. All PWRs have in their proposed emergency guidelines, methods for use of

decay heat removal schemes other than the design basis equipment. In particular, guidance is given to provide alternate sources of secondary cooling if main and auxiliary feedwater are unavailable (e.g., by depressurizing the secondary system and activating the condensate pumps). Operators would resort to feed and bleed only if no source of water is available to feed the steam generators. The NRC has no design requirements for these other alternate schemes, just as we have none for the "feed and bleed" capability. What is required for the design basis is a reliable auxiliary feedwater system to remove decay heat until the RHR system can be activated to ultimately achieve cold shutdown. However, to provide defense in depth, feed and bleed procedural instructions should be available to operators because the capability to feed and bleed exists.

As to the technical performance of "feed and bleed," we know it depends on the HPI pump performance characteristics, the PORV relieving capacity, and the plant power to volume ratio. Analyses have been conducted by all three PWR suppliers to examine "feed and bleed" capability for their designs. Also, NRC contractors at LANL and INEL have analyzed "feed and bleed" with the computer codes TRAC and RELAP. As noted previously, a B&W calculation for a TMI class plant showed that "feed and bleed" was an effective heat removal method even if no credit is taken for PORV actuation. This is because most B&W plants have HPI pumps with a very high shutoff head, and enough energy can be relieved at high pressure through the safety valves. It is important to note that the assessment of "feed and bleed" rests almost exclusively on <u>analysis</u>.

Analytical uncertainties related to such phenomena as non-equilibrium thermodynamics, bubble formation and repressurization caution against taking too much credit for analytical predictions of system behavior.

One LOFT experiment (L9-1/L3-3) explored "feed and bleed" in a limited way. After a simulated loss of feedwater, the PORV was latched open to allow depressurization. The results showed that depressurization to the HPI actuation point did indeed occur. However, HPI actuation was purposely not allowed to occur so that other accident mitigation schemes could be explored.

Item 4 Recommendations for Future Action

It is desirable to improve the experimental basis for understanding system behavior during "feed and bleed." This should improve the guidance in emergency procedures and training that is being developed under Task I.C.1 of NUREG-0737. To accomplish this, we are exploring ways to expand the current Semiscale test series to include "feed and bleed" experimental data. We expect shortly to issue a request to RES which will include these proposals.

The current Semiscale configuration cannot simulate the unique features of the B&W NSSS. You know from previous discussions that we have been trying to resolve the problem of uncertainties for the B&W analytical methods in predicting long term LOCA recovery under Task II.K.3.30 NUREG-0737. We

are investigating the unique features of the B&W design and the lack of integral systems data (see attached letter to B&W owners). We will shortly transmit to all B&W owners our conclusion that such data are required. The basis for this conclusion is the need for additional verification of some aspects of the thermal-hydraulic behavior during natural circulation cooling of the B&W design with feedwater available during small break LOCAs, as well as uncertainty in the feed and bleed process. You will also recall that the ACRS letter of June, 1982 highlighted this problem for resolution prior to its concurrence on full power operation of Midland, a B&W reactor.

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

JUN 10 1982

MEMORANDUM FOR: Gerry Mazetis, Section Leader
Section C
Reactor Systems Branch, NRR

FROM: C. J. Heltemes, Jr., Deputy Director
Office for Analysis and Evaluation of
Operational Data

SUBJECT: DRAFT REPORT ON NRC STAFF POSITION ON FEED AND BLEED COOLING AT TMI-1 RESTART HEARING

In accordance with your June 2, 1982 request, AEOD has reviewed the subject draft report. Enclosed is a copy of AEOD's comments on the report (which have been provided to you informally during June 7, 1982 telecons between Hal Ornstein, Walt Jensen, and yourself). If you have any questions concerning this matter, please contact Hal Ornstein on extension 24439.

C. J. Heltemes, Jr., Deputy Director
Office for Analysis and Evaluation of
Operational Data

Enclosure:
As Stated

cc w/enclosure:
R. Mattson
H. Thompson
T. Speis
B. Sheron
W. Jensen

AEOD Comments on the draft "Report on NRC Staff Position on Feed and Bleed Cooling at TMI-1 Restart Hearing"

1. It is our understanding that the report is in response to Harold Denton's April 29, 1982 memorandum, "Reliability and Effectiveness of Feed and Bleed Core Cooling at TMI-1." In this regard, AEOD believes that the clarity of the report would be enhanced if the scope were limited to B&W plants (if possible). However, if it is deemed necessary to discuss other vendor designs, it is suggested that such discussions be placed in separate sections (or appendices) of the report, rather than having such discussions intermingled with the discussions of B&W plants.

2. Some of the scenarios discussed in the report assume multiple failure events of safety grade systems. Usually the staff considers multiple active failures of safety grade systems not to be sufficiently credible that such failures need to be considered in the plants' design bases. Consequently, the reason for considering complete failure of the auxiliary feedwater system (if safety grade) or the high pressure injection system should be presented in the report; i.e., some discussion is warranted on NUREG-0737, item I.C.1 - Guidance for the Evaluation and Development of Procedures for Transients and Accidents, which requires guideline and procedure development to consider occurrences of multiple and consequential failures.

3. To improve the reader's understanding of several technical issues, it is suggested that some additional information be included on the following items:

   (a) Page 8 - The discussion on the different modes of natural circulation should include what assumptions are made regarding secondary side conditions and details of what conditions lead up to entering each mode, and what may be involved or necessary to recover.

   (b) Page 10 - The report should note that the scenario discussed assumes that emergency feedwater is available.

4. Page 10, item 3c - Assessment of Current Status and Existing Information on Feed and Bleed Response:

   We believe that the conclusion

   > "If the feed and bleed process discussed above was insufficient to remove decay heat, natural circulation would be established in the boiler/condenser mode"

   is not a certainty, especially in the absence of experimental data for B&W plants. In the event that, for any reason, natural circulation cannot be established and the primary coolant pumps are not available, the "feed and bleed" mode of decay heat removal would have to be used.

5. Page 11 - It is our understanding that the emergency guidelines (or emergency procedures) discussed in this section are not presently in place. Thus, it is important to provide a sense of timing regarding what is in place and available now (in terms of equipment, procedures, and training) and what is likely to be available at some specified time in the future.

6. Page 11, lines 16-19 - It is our understanding that the RHR system would be activated before achieving cold shutdown. The AFW system does not usually bring the plant to cold shutdown ($\leq 200^{o}F$).

7. Page 13, item 4 - Recommendation for the Future: We agree with the need for obtaining experimental verification of the analytical code predictions. We believe that this section of the report should be expanded to clarify the items for which verification is considered appropriate or necessary. In this regard, consideration should be given to (a) natural circulation in B&W plants, including establishment of boiler/condenser operation, and elimination of steam formations in the hot legs; and (b) the ability of existing PORV and safety valves to perform reliably in a "feed and bleed" mode.

ATTACHMENT 3

RECEIVED

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

'82 MAR 22 AM 8:59

OFFICE OF THE SECRETARY
D.C.

JAN 29 1982

MEMORANDUM FOR: Bob Tedesco, Assistant Director for Licensing
Division of Licensing, NRR

Themis Speis, Assistant Director for Reactor Safety
Division of Systems Integration, NRR

FROM: Frank H. Rowsome, Deputy Director
Division of Risk Analysis, RES

Joseph A. Murphy
Reactor Risk Branch
Division of Risk Analysis, RES

SUBJECT: FEED AND BLEED ISSUE FOR CE APPLICANTS

We have performed a quick and dirty analysis of the risk implications of CE designs that lack a capability for core cooling via HPI injection and deliberate venting of the reactor coolant system, in the absence of feedwater replenishment.

We conclude that three classes of accidents may each be more frequent than the Commission's safety goal of 10^{-4} core melts per reactor year or less, and that the total core melt frequency for such plants could be of the order of 10^{-3} per year or more. The three sequences are:

1. Transient and failure of all feedwater (not associated with loss of AC power) (TML).

2. Loss of offsite power, one diesel failure disabling the motor driven AFW train, and failure of the turbine-driven AFW train.

3. Very small LOCA and failure of HPI (S_2D).

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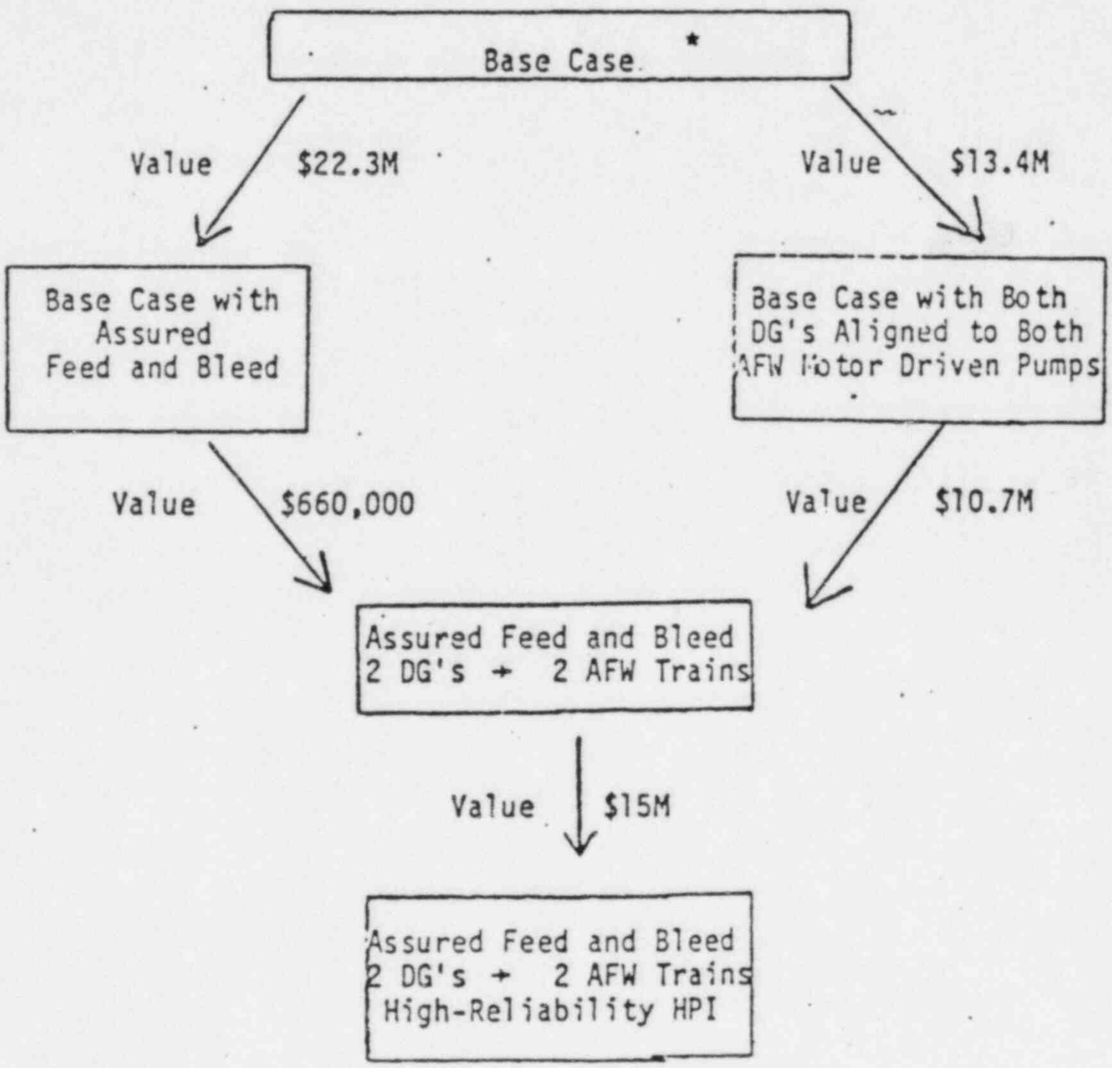
We recommend the following upgrades to these designs:

1. Provide an assured "feed and bleed" capability.

2. Provide that either diesel generator can energize a motor driven AFW train.

3. Examine carefully and perhaps upgrade HPI reliability and/or reduce the frequency of very small LOCA's.

The economic incentives to make these improvements, derived from reduced risk of economic losses associated with core melts, are roughly:



*The base case plant is assumed to be incapable of feed and bleed cooling, only one diesel generator is assumed capable of energizing the safety related motor driven AFW train. The turbine driven AFW train is AC-independent, but the non-safety grade motor-driven AFW train requires offsite power. Industry average HPI reliability and S_2-LOCA frequency is assumed. The analysis that shows that S_2D may be too frequent applies to other PWRs as well.

The attached paper describes the analysis.

Frank H. Rowsome, Deputy Director
Division of Risk Analysis
Office of Nuclear Regulatory Research

Joseph A. Murphy
Reactor Risk Branch
Division of Risk Analysis
Office of Nuclear Regulatory Research

Attachment: As Stated

cc: R. Bernero
G. Burdick
R. Mattson
S. Hanauer
M. Ernst
A. Thadani
RRB Staff
RAB Staff

Feed and Bleed Issue for CE Applicants

We understand that the current crop of CE license applicants are proposing that no pressurizer PORV's be installed, that the HPI shutoff head is to be well below the pressurizer safety valve setpoint (around 1400 psi), that high point vents provide no more than two 1" diameter remote-manual vents, and that the auxiliary feedwater systems will be composed of one AC-independent turbine driven pump, one AC-power train, and a third non-safety grade motor driven pump.

We have attempted a back-of-the-envelope PRA in order to evaluate the risk implications if these plants are incapable of "feed and bleed" cooling. The results suggest that they may fail to meet the Commission's safety goal of a core melt frequency less than 10^{-4}/year and the present worth of a fix to enable assured feed and bleed cooling is of the order of $10 million or more per plant, based upon reduced financial risk alone. We considered five groups of accident sequences: loss of main feedwater, loss of offsite power, very small LOCA, transient-induced small LOCA (late start of auxiliary feedwater allows a lift of a pressurizer code safety valve which may stick open), and station blackout with restoration of AC power just before the point-of-no-return. We did not consider main steam line breaks or ATWS, although in these sequences an assured feed and bleed capability could also enhance safety as well as in the sequences considered.

The simple loss of main feedwater appears to be the dominant concern. For this sequence in a plant incapable of feed and bleed cooling, the frequency of core melt, $\lambda_{cm} = \lambda_m P(L)$, where λ_m is the frequency of critical (sustained) failures of main feedwater, and $P(L)$ is the probability of a critical failure of the auxiliary feedwater system.

WASH-1400 took the frequency of feedwater transients to be 3 per year, with 99 out of one hundred such occurrences recoverable. There is reason to doubt both numbers. Complete interruptions of main feedwater are more frequent than 3 per year during the life of the first core, while the plant is still being debugged, although many take place at startup or at low power when the decay heat level is too low to pose much risk. A mature plant has complete interruptions of main feedwater about once a year or less. The non-recovery factor of 10^{-2} applies to plants with simple feedwater controls, motor driven main feedwater pumps, and no major obstacles to feedwater restart after a trip. In large, modern plants with turbine-driven main feedwater pumps problems with feedwater restart are common, so a non-recovery factor of .3 to .1 is more reasonable. I judge that the frequency of non-restorable failures of main feedwater occurring from substantial (risky) initial power levels is roughly:

$$\lambda_m = \begin{cases} 0.3 \times 10^{+1}_{-1}, & \text{first core} \\ 0.1 \times 10^{+1}_{-1}, & \text{at maturity} \end{cases}$$

Auxiliary feedwater reliability is also uncertain. Data from the precursor program suggests that the PWR average experience has been a failure probability of 10^{-3}/demand. This average includes early-in-life experience as well as mature plant experience and two train as well as three train experience. System reliability analyses have suggested that the best of the three train systems can approach - at maturity - 10^{-5} per demand. However, these analyses failed to consider some common mode failure mechanisms so they can be regarded as having an optimistic bias. It is not uncommon early in plant life to find instances of repeated, consistent, auxiliary feedwater pump failures while the system is being debugged in service. The record suggests that the failure probability of the AFWS is substantially higher during the first core than in maturity. A system with two diverse safety grade AFW trains and a third full capacity non-safety grade train will probably achieve failure probabilities of:

$$P(L) = \begin{array}{l} 3 \times 10^{-3 \pm 1}, \text{ first core} \\ 1 \times 10^{-4 \pm 1}, \text{ at maturity} \end{array}$$

These estimates result in loss-of-all-feedwater frequencies of:

$$\lambda_{cm} = \begin{array}{l} 0.9 \times 10^{-3 \pm 1.4}/\text{yr, first core} \\ 1 \times 10^{-5 \pm 1.4}/\text{yr, at maturity} \end{array}$$

The uncertainty range is thus:

$$2.3 \times 10^{-2} \gtrsim \lambda_{cm} \gtrsim 3.5 \times 10^{-5}, \text{ first core}$$
$$2.6 \times 10^{-4} \gtrsim \lambda_{cm} \gtrsim 3.9 \times 10^{-7}, \text{ at maturity}$$

Note that even at maturity this core melt sequence frequency may be higher than the Commission's criterion for all core melt frequencies combined: $\lambda_{cm} \lesssim 10^{-4}$/yr, and that the best estimate is that it will exceed the Commission's criterion during the first core. Note also that common-causation of main and auxiliary feedwater failure due to fires, floods, earthquakes, or sabotage has not been considered and might increase this sequence frequency. The Commission's guidelines on acceptable risk do not indicate how to treat uncertainties or higher-than-average estimates for the first core.. Nonetheless, I think it unwise to allow a single core melt accident sequence to be this probable. The provision of an assured feed and bleed capability would enable HPI to cool the core in these scenarios. Even with common mode and external hazards, this should be worth at least one decade, more likely two decades reduction. We recommend it.

Next let us consider loss of offsite power. The failure frequencies or probabilities are taken to be:

λ_{LOSP} = 0.2/yr

P non-recovery of offsite power within 30 min - 1 hr = 0.2/occurrence

Thus λ_{LOSP} without recovery = 0.04/yr

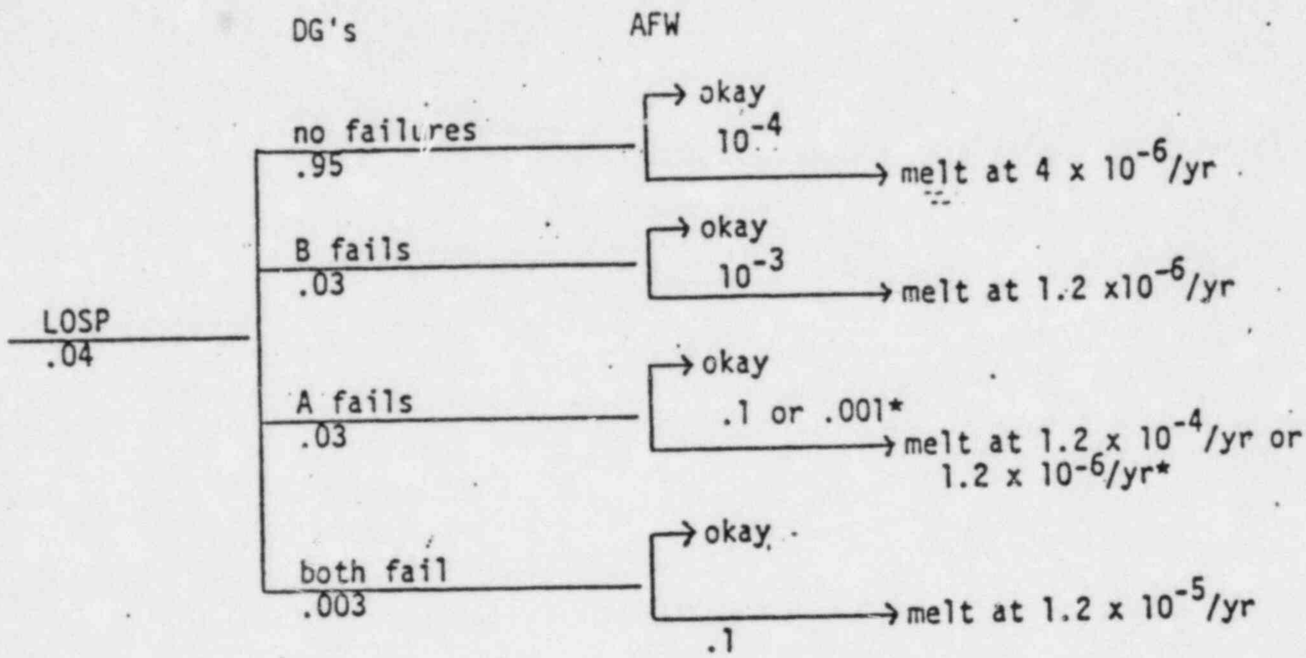
P_{DG} = 0.03/demand

P_{2DG} = 0.003/demand, including common mode

$P_{AFW\text{-turbine train}}$ = 0.1/demand

$P_{AFW\text{-motor train}}$ = 0.01/demand

Assume for convenience that diesel generator A is configured to energize the safety grade AFW motor driven train. As we shall see, the core melt frequency predictions are sensitive to whether or not diesel generator B can energize the non-safety grade AFW train or not. The event tree for loss of offsite power can be drawn:



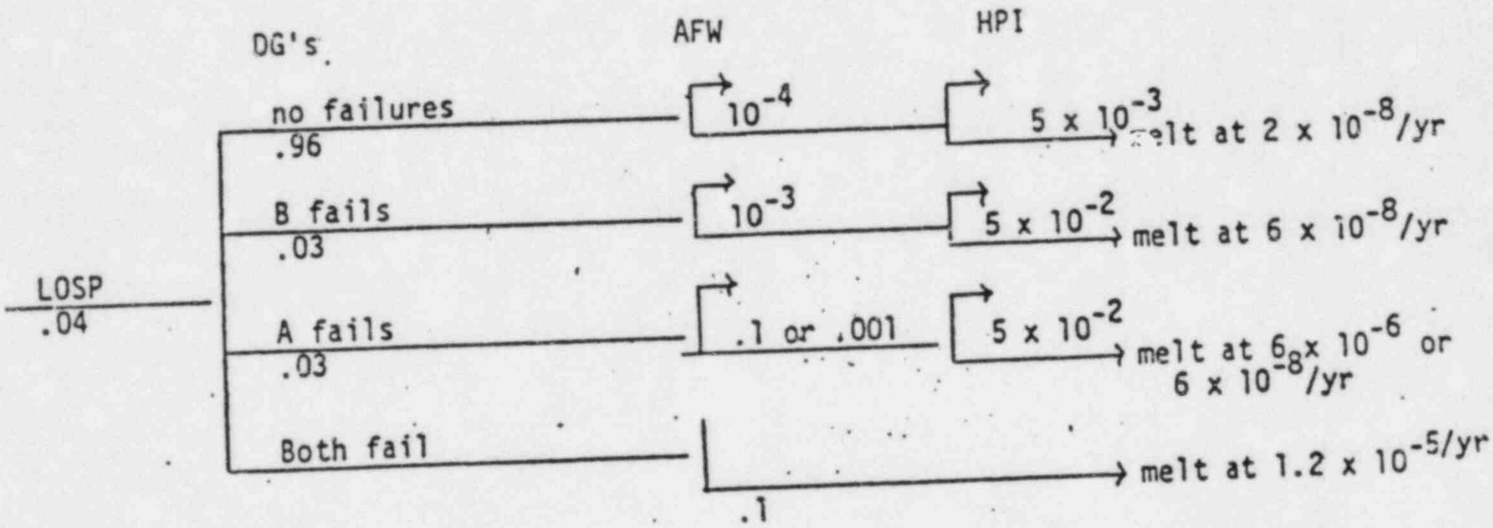
*The higher failure rate applies if one of the diesel generators (we have called it B) cannot power a motor driven AFW train; the lower failure rate applies if both diesel generators can power a motor driven AFW train.

Note that the Commission safety goal of 10^{-4}/yr for all core melt sequences may be violated by loss of offsite power and a single diesel generator failure if there is one diesel generator that cannot be aligned to energize a motor-driven AFW train. This high core melt frequency could be reduced to marginally acceptable value in either of two ways:

1. Insure that either diesel generator can be aligned to energize a motor-driven AFW train by (i) providing a swing bus for the safety grade AFW pump, or (ii) providing an essential (diesel-backed) power supply to the "non safety grade" AFW pump, or

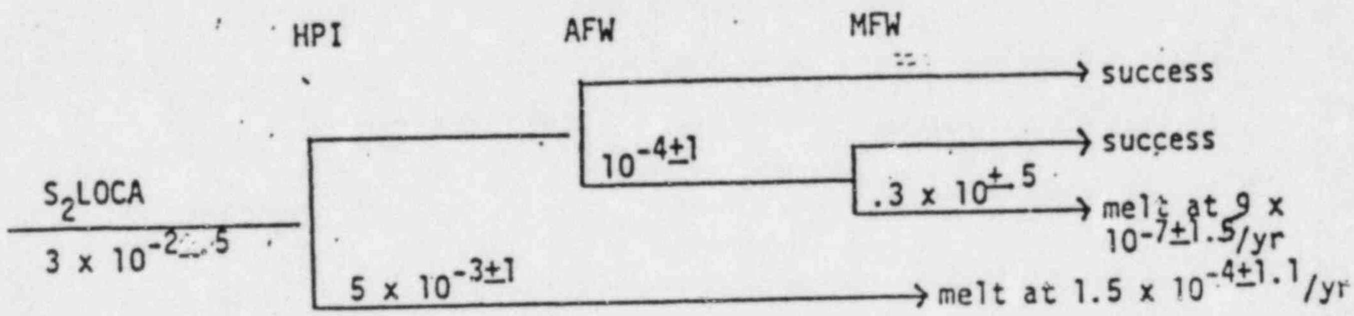
2. Provide an assured feed and bleed capability so that the one operable diesel generator and its associated HPI train can cool the core.

The case of full station blackout is considered later. The value of the feed-and-bleed fix can be inferred from the event tree for LOSP with this design:



Next let us consider very small (S_2) LOCA. Instrument line breaks, steam generator tube ruptures, charging pump line breaks, and gross reactor coolant pump seal failures have happened a dozen or so times in 500 LWR-years, suggesting a challenge frequency of $3 \times 10^{-2 \pm .5}$/yr for S_2LOCA excluding PORV LOCAs. They are less probable in the first year of service, so I will not single out first core numbers.

In the CE plants, both feedwater and ECCS (HPI) are required for successful core cooling. Main feedwater may remain operable or be restartable in some of these. The probability of HPI failure on demand was found to be 8.6 x $10^{-3\pm.5}$ in Surry (WASH-1400). Most PWR PRAs are finding a failure probability for the whole multi-train HPI between 10^{-2} and 10^{-3}/demand. We shall assume that the probability of HPI failure on demand is 5 x $10^{-3\pm1}$/demand for the CE plants. A rough cut at frequency estimation suggests:



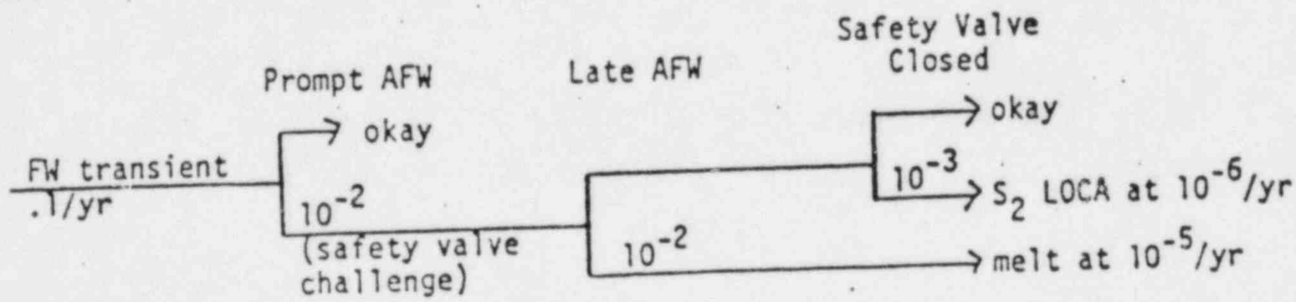
The value of an assured feed and bleed capability here is to eliminate the need for feedwater. This would eliminate the smaller (10^{-6}/yr) path to core melt without affecting the more prominent path via HPI failure. Note that small LOCA with total HPI failure is predicted to result in a core melt frequency above the Commission goal for all core melts. The provision of feed and bleed capability or of an improved AFW system will not help this. It is a problem generic to PWRs and not unique to the CE designs. It appears that the high frequency of very small LOCA revealed by historical experience and the marginal HPI system reliabilities revealed by many PWR PRAs are combining to yield unacceptable core melt frequencies through S_2D-type sequences. We suggest that NRR tackle this

problem in two ways: First, a serious effort should be made to reduce the frequency of S_2 LOCA's. Second, a broad-scale attack on HPI reliability problems comparable to that instituted for AFW systems after TMI should be initiated for all PWR's.

Next let us consider the transient-induced small LOCA's, with and without a PORV. A feedwater transient with a prompt autostart of auxiliary feedwater is assumed not to lift a pressurizer relief valve. However, a delayed start of AFW, which may be roughly one hundred times as likely as a sustained AFW failure, may lift a pressurizer valve (PORV or code safety) and the valve may stick open.

LER data suggest that PORV's stick open roughly once in one hundred challenges and code safety valves once in a thousand challenges. Neither type of valve have failed open spontaneously, to my knowledge, although there was one instance (Crystal River NNI bus fault) of a command fault leading to an open PORV. Since TMI I think it safe to assume that operators would successfully close the PORV block valve in at least 99 out of 100 instances of a PORV-LOCA.

Without a PORV we have (at maturity):



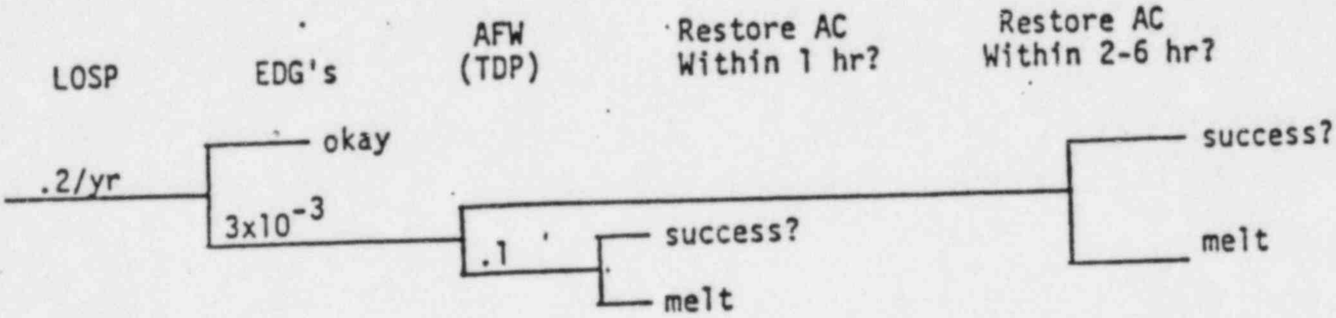
The core melt outcome from loss of all feedwater has already been considered. The increment in the likelihood of S_2 LOCA is negligible at 10^{-6}/yr. It can still be mitigated by HPI, if HPI works, as it will do in the vast majority of cases.

With a PORV we will get transient-induced LOCA ten times as often (10^{-5}/yr) but the block valve can be expected to terminate all but 1 percent of these for a frequency of transient-induced and unisolated LOCA of 10^{-7}/yr. If anything, the PORV helps rather than aggravates what is a negligible contributor to the overall S_2 frequency via transient-induced LOCA.

We should also consider the command fault LOCA's due to spurious "open" commands to a PORV. The frequency of occurrence is a sensitive function of the valve control logic design. It could be made as small as we wish by suitable reliability engineering. If we consider the Crystal River experience as one failure in 300 PWR-years, we get an industry average of 3×10^{-3}/yr for PORV command fault LOCA. Clearly, B&W did not do so well, but the combined experience of the three PWR vendors suggests that this frequency can easily be made much less than the overall S_2 frequency of $3\times10^{-2\pm.5}$/yr. I conclude that having a PORV or not having a PORV has a negligible effect on the likelihood of S_2 LOCA or of the likelihood that S_2 LOCA may lead to core melt, provided that system or component functional reliability is the only consideration. It goes without saying that this analysis is predicated upon a design with anticipatory trips so that routine transients do not lift pressurizer relief valves, and that the operators are trained to close the PORV block valve when appropriate.

There may also be a design adequacy issue. I feel uncomfortable with 1400 psi HPI pumps in plants without PORV's, even if the HPI and the AFW systems are highly reliable. Careful thermal hydraulic analyses together with thorough studies of plausible operator responses are necessary to verify that some S_2 LOCA's will not lead to degraded steam generator heat transfer and RCS pressures over 1400 psi while the core uncovers, even with operable HPI and AFW trains. The high point vents and reactor coolant pumps may help here even though these plants do not have full feed and bleed capability. However, these design adequacy issues are beyond the capability of this simplistic system reliability analysis.

Last, consider station blackout with AC recovery near the point of no return. The event tree may be drawn as follows:



Blackout with successful auxiliary feedwater (turbine driven pump) can be expected at a frequency of roughly $6x10^{-4}$/yr. The turbine driven AF pump has a finite success window, however. One of several factors will lead to core melt if AC power is not ultimately restored. These factors include: (a) loss of reactor coolant inventory (blown RCP seals, etc.); (b) dead batteries (discharge or overheat); (c) high pump room temperatures (no HVAC); or (d) depletion of condensate.

Blackout without auxiliary feedwater leads to a shorter time window to save the core by AC recovery. This can be expected at a frequency of roughly 6×10^{-5}/yr. In either scenario, as the time to the point-of-no-return for core cooling approaches, the reactor coolant system pressure will be high, (around the pressurizer safety valve set point), and the level will be falling toward the top of the active core. Refilling the steam generators will be necessary but may not be sufficient, depending upon the effectiveness of reflux condensation and the extent of reactor coolant system leakage. A feed and bleed capability to enable HPI to refill the reactor coolant system fairly quickly might extend the window for AC recovery without core damage or melt by tens of minutes, perhaps more. A quantitative evaluation of the fraction of melt sequences that could be saved by feed and bleed would require extensive thermal hydraulic analysis and analysis of the likelihood of AC restoration vs time. However, it is clear that the most likely AC restoration times are before any point of no return. hus, an upper bound on the improvement in the blackout melt sequence frequency attributable to feed and bleed is of the order of 10^{-6}/yr or less.

To summarize, the principal concerns regarding the CE designs with low HPI shutoff head and no PORV's appear to be:

1. Risk of core melt via loss of all feedwater may be unacceptably high.

2. The adequacy of the design for very small LOCA mitigation is questionable. This may be coupled with operator behavior issues.

3. The reliability of the high pressure injection system may be unacceptably low, but the mere fact of an AFW requirement to mitigate very small LOCA's - given design adequacy - does not significantly degrade the reliability with which very small LOCA's may be mitigated.
4. It is important that either diesel generator be capable of energizing a motor driven AFW train given loss of offsite power.

Two questions remain to be answered: (1) what is it worth to equip these plants with feed and bleed capability? and (2) what are the attendant risks of the optional f'xes?

As assessment of the value of the fix follows. Those core melt accident sequences for which a feed and bleed capability could save the core are likely to be well-contained; they do not entail common mode failure mechanisms which would defeat containment isolation, sprays, or fan coolers. Thus the utility's economic risk dominates.

Let us take the cost of such a core melt event to be around $10 billion (low: $2 billion for TMI's; high: $100 billion for extensive shutdown orders). The value in $ is essentially:

$V(\$) = \Delta\lambda$ (events per year) x C($ per event) x T(exposure time in years)

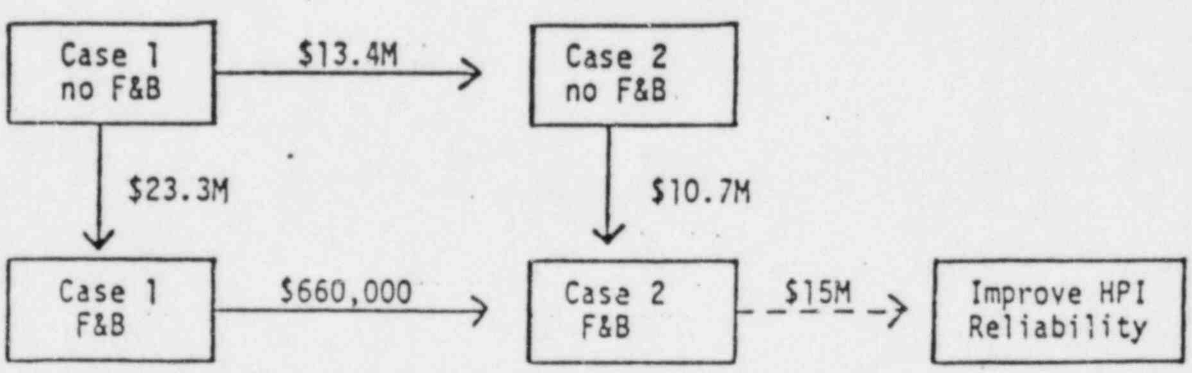
We can calculate a variety of $\Delta\lambda_{cm}$ differences from the following table:

λ_{cm}	Without Feed and Bleed	With Feed and Bleed
TML (first core)	9×10^{-4}	9×10^{-6}
TML (mature)	1×10^{-5}	1×10^{-7}
LOSP Case 1*	1.4×10^{-4}	1.8×10^{-5}
LOSP Case 2*	1.8×10^{-5}	1.2×10^{-5}
S_2D	1.509×10^{-4}	1.5×10^{-4}

*Case 1 - one of the diesel generators cannot energize a motor driven AFW train

Case 2 - both diesel generators can energize a motor driven AFW train

The economic incentives can be calculated by taking the exposure time for the first core as one year and for mature operation as ten years. The economic incentive is essentially the reduction in the present worth (at startup) of projected monitary losses due to accidents. They are shown on the following diagram:



This diagram can be understood as follows. Start with a CE plant that has no feed and bleed capability and only one diesel generator that can support a motor-driven auxiliary feedwater pump. It would be worth up to \$13.4M to enable the second diesel generator to power what is now the non-safety grade AFW pump. It would be worth up to \$22.3M to add feed and bleed capability, and so forth. The final "fix" has yet to be discussed. The value was arrived at by postulating design or operational changes such that the likelihood of an S_2D core melt is reduced from $1.5x10^{-4}$/yr to $1.0x10^{-5}$/yr. This might be achieved by either improving the reliability of HPI substantially, reducing the frequency of very small LOCA substantially, or some of each.

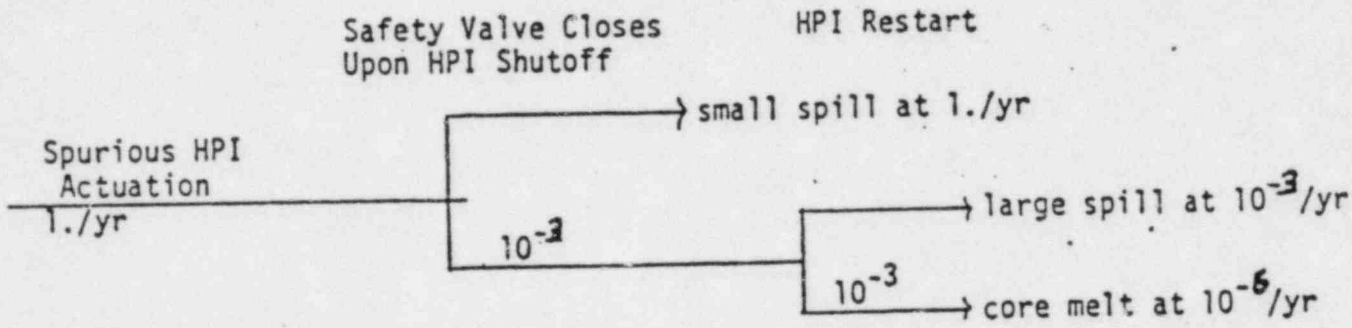
Now a feed and bleed capability could be achieved by installing suitably sized PORV's or by installing HPI pumps of very high head (over the pressurizer safety valve setpoint) or some of each. We have already examined the attendant risks of PORV addition. Care must be taken to design the control logic so that spurious "open" commands are rare, but it is safe to expect that this will be done well enough that the frequency of S_2 LOCA is not significantly increased. The effect on transient-induced LOCA is not important (this frequency is negligble with or without a PORV) and is compensated by the possibility of isolating PORV-LOCA's with the block valve.

If the HPI can force open a pressure relief valve (code safety or PORV in the pressurizer), then a spurious HPI actuation can cause a temporary, recoverable LOCA. Should the valve stick, we may have (without a block valve) a sustained LOCA. I assume that the operators will shut off HPI though not before a

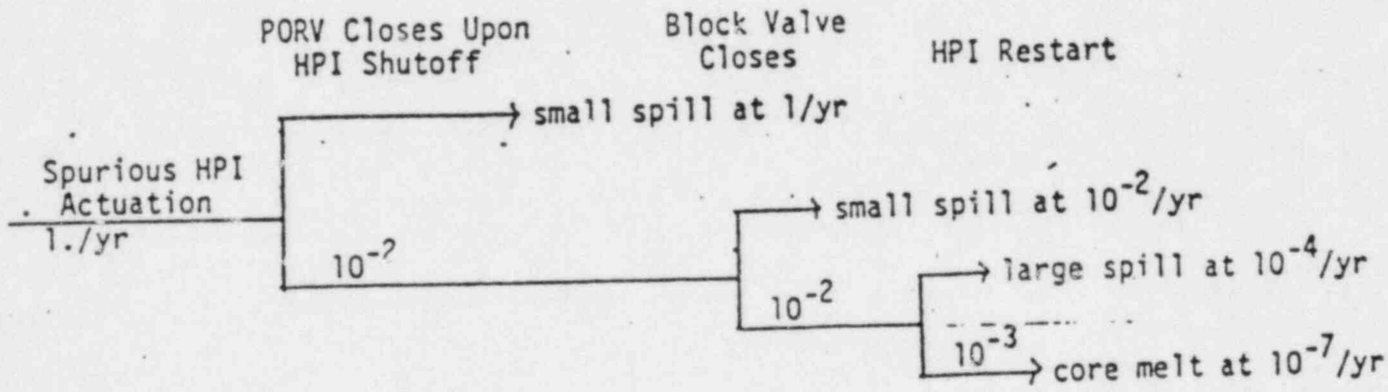
pressurizer valve opens, the pressurizer quench tank rupture disk blows, and a small spill occurs. If the valve sticks open (and cannot be isolated), the operators must restart HPI. Spurious HPI actuations are quite common. We assume here that the frequency of spurious HPI actuations which remain on long enough to challenge a pressurizer valve is one per year.

Borrowing from the prior analyses we can draw the following event trees for the high head HPI design:

Without PORV (or PORV left blocked)



With PORV installed and unblocked



Note that if a PWR has a PORV and high head HPI, it is better to run with the block valve open, so the isolatable PORV can take the brunt of spurious HPI actuations as well as feedwater transient-induced LOCA's. Note also that the core melt sequences caused by spurious HPI actuation in plants with high head HPI is acceptably small and can be made smaller still if the PORV only lifts (block valve left open). It is roughly balanced by comparable risk reductions in that for these designs, the PORV need not open to accommodate feed and bleed.

However, we should note that there is a real economic incentive to avoid the blown pressurizer quench tank rupture disk and the attendant small spills. If we assume a five day outage at one million dollars a day for small spills and a 100 day outage for a large spill, then the present worth of expected losses due to spurious HPI actuation in these designs is:

1 event/yr x 5×10^6 \$/event x 10 year exposure = \$50 million from the small, frequent spills with either design variant. For the large spills (unisolated LOCA) we have:

$$\left.\begin{array}{l}\text{Without PORV: } 10^{-3}/\text{yr} \\ \text{With PORV: } 10^{-4}/\text{yr}\end{array}\right\} \times 10^8 \text{ \$/event} \times 10 \text{ yr} = \left\{\begin{array}{l}\$10^6 \\ \$10^5\end{array}\right.$$

Thus utilities are subject to a significant incentive (present worth of projected losses of \$50 million) either to employ HPI pumps that cannot lift a pressurizer relief valve or to go after improved prevention of spurious HPI actuations or both.

There appears to be no economic penalty (other than first cost) in providing HPI pumps whose shutoff head is at normal RCS pressure, i.e., around 2250 psi.

In summary, then, this limited risk analysis cannot distinguish a difference in safety among the several ways to achieve feed and bleed capability: install one or more large PORV's, raise the HPI head above the pressurizer safety valve setpoint, or install a smaller PORV and raise the HPI head to near normal operating pressures. These choices must be made on the basis of design adequacy or thermal hydraulic considerations, preferably considering ATWS as well as the design to assure that very small LOCA's can be mitigated even though HPI or AFW may be late in starting or might be throttled temporarily by the operators. We have, however, found a plant availability incentive to avoid an HPI head so high that it can lift a pressurizer relief valve. No such penalty accrues to HPI designs with a shutoff head at the normal RCS pressure.