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MEMORANDUM FOR: Darrell G. Eisenhut, NRR
Edward L. Jordan, IE
Donald B. Mausshardt, NMSS
Robert M. Bernero, RES
Clemens J. Heltemes, Jr., AEOD
Joseph Scinto, ELD

FROM: Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

SUBJECT: CRGR MEETING NUMBER 20

The Committee to Review Generic Requirements will meet on Tuesday, September 21, 1982 from 1-4 p.m. in Room 6507 MNBB. This meeting is being scheduled on short notice in response to the NRR and IE Office Directors' request for expedient CRGR consideration of the following items:

1:00 - 2:00 pm R. Mattson (NRR) will brief the CRGR concerning the proposed resolution of the reactor coolant pump trip issue (Enclosure 1).

2:00 - 4:00 pm R. Baer (IE) will present for CRGR review, the proposed bulletin to address BWR pipe cracking. I have enclosed an information notice on the subject.

Persons making presentations to the CRGR are responsible for (1) assuring that the information required for CRGR review is provided to the Committee (CRGR Charter - IV.B), (2) coordinating and presenting views of other offices, (3) as appropriate, assuring that other offices are represented during the presentation, and (4) assuring that agenda modifications are coordinated with the CRGR contact (Walt Schwink, x24342) and others involved with the presentation. With regard to attendance at CRGR meetings, I request that Office Directors limit attendance of their staffs at CRGR meetings to those few senior staff needed to address the agenda item under discussion. As a minimum, Division Directors or higher management should attend meetings addressing agenda items under their purview.

Original signed by
Victor Stello

Victor Stello, Jr., Chairman
Committee to Review Generic Requirements

Enclosures:
As stated

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PDR REVGP NRGCRGR
PDR

cc w/o encl.:
Commission (5)
~~EDO~~
Office Directors
Regional Administrators
G. Cunningham
R. Mattson
R. Baer

OFFICE	DEDROGR/S	DEDROGR/D	DEDROGR		
SURNAME	WSchwink:bg	TEMurley	VStello		
DATE	9/17/82	9/17/82	9/20/82		

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FOR: The Commissioners

FROM: William J. Dircks, Executive Director for Operations

SUBJECT: STAFF RESOLUTION OF THE REACTOR COOLANT PUMP TRIP ISSUE

PURPOSE: To Inform The Commission Of The Staff Resolution Of The Reactor Coolant Pump Trip Issue (TMI Action Plan Item II.K.3.5)

DISCUSSION: Soon after the accident at TMI-2, the staff issued a series of IE Bulletins with guidance on pump operation for PWRs. Bulletin 79-05A was applicable to operating PWRs designed by Babcock and Wilcox. It was issued on April 5, 1979. The bulletin required that, in the event of HPI actuation when the reactor coolant pumps were operating, at least one reactor coolant pump (RCP) per coolant loop should remain operating. Bulletin 79-06A, applicable to operating PWRs designed by Westinghouse, was issued on April 14, 1979. It required that, in a similar event, at least one RCP should remain operating for two-loop plants and at least two RCPs should remain operating for three and four loop plants. The third such bulletin, 79-06B, was issued on April 14, 1979 and was applicable to operating PWRs designed by Combustion Engineering. It required that at least one RCP per loop should remain operating for an event causing HPI actuation with the Reactor Coolant Pumps operating.

The basis for the guidance provided in these three bulletins was twofold: during the TMI-2 accident the core remained cool as long as the RCPs were running, and significant core damage resulted after the RCPs were tripped.

CONTACT: Brian W. Sheron, NRR/DSI/RSB
49-27460

Subsequent to the issuance of these bulletins, the staff, in a letter on June 5, 1979, requested the PWR licensees to provide further evaluations of small break loss of coolant accident (SBLOCA) behavior for their plants. Included was a request to examine the effects of RCP operation on SBLOCAs.

In early July, 1979, B&W briefed the staff on the results of its evaluations. They informed us that for a certain range of small break sizes, leaving the RCPs running was calculated to increase the primary system coolant inventory loss compared to tripping the RCPs early. However, as long as the RCPs remained running, the analysis showed that there was sufficient flow of two-phase coolant to keep the core cooled. If the reactor coolant pumps were stopped (e.g., tripped, lost power, seized) during the period in which the primary coolant inventory loss was high, the two-phase mixture being circulated in the primary system would collapse. This resulted in an unacceptable degree of core uncovering and fuel cladding heatup. Their analyses also showed that if the RCPs were tripped relatively early (i.e., less than about 3 minutes) after the break was initiated, no unacceptable core uncovery was calculated to occur. Shortly after this briefing, B&W issued revised operating guidelines to its utility customers on July 20, 1979, instructing them all RCPs should be tripped in the event of reactor trip and HPI actuation on low pressure.

Subsequently, excessive inventory loss with continued RCP operation followed by delayed RCP trip for selected SBLOCAs was also shown by Westinghouse and Combustion Engineering to result in unacceptable core uncovery and heatup for their designs. Based on the available information, the staff and the three PWR vendors agreed that early RCP trip was the prudent action to take for the full spectrum of design basis LOCAs. Accordingly, on July 26, 1979, the staff issued IE bulletins 79-05C and 79-06C, reversing the guidance of the previous bulletins and instructing operators to trip all operating RCPs upon reactor trip and HPI actuation on low pressure. The bulletins also required that licensees evaluate the need for automatic tripping of the RCPs "under all circumstances in which this action may be needed."

The details of the excessive inventory loss phenomenon associated with pump operation during SBLOCAs were documented in NUREG-0623. This NUREG was issued by the Bulletins and Orders Task Force in November, 1979 and contained recommendations for automatic RCP trip during SBLOCA if RCP trip was concluded to be necessary. The industry argued that RCP trip in the event of SBLOCAs was appropriate, but that sufficient time existed so that the trip action could be performed manually by the operators. The ACRS also questioned the need for automatic trip and recommended the issue be studied further. The issue was designated item II.K.3.5 in the TMI Action Plan (NUREG-0660). The staff's agreement with the ACRS recommendation to study the issue further was documented in NUREG-0660.

Since the issuance of the TMI Action Plan, the staff and the industry have been working to resolve the RCP trip issue. The major effort by industry was to verify the analysis models used to predict SBLOCA behavior with the RCPs running. To accomplish this, we required each holder of an approved ECCS evaluation model to verify the dynamic thermal-hydraulic models in their Appendix K evaluation models <u>or</u> their best-estimate computer codes against LOFT test L3-6, a small cold leg break experiment in which the RCPs remained operational. From these analyses we concluded that each ECCS model holder with the exception of EXXON Nuclear, could acceptably calculate SBLOCA behavior with the RCPs operational. (The EXXON core reactors are still governed by Bulletins 79-05C and 06C which require prompt RCP trip. Thus, the analysis for RCP operation does not adversely affect the safety of any operating reactors.)

Using the verified computer models, analyses have been performed by both the industry and the staff to determine whether RCP trip is necessary during SBLOCAs. From these analyses, we have concluded that the RCPs should be tripped upon confirmation of a LOCA.

The basis for our conclusion, including discussions with industry representatives, is provided in enclosure (1). Our proposed letters to licensees and OL applicants, setting forth guidelines which must be addressed when determining both the pump trip setpoints and the method (manual or automatic) are provided in enclosure (2). These proposed letters were informally provided to each PWR vendor and owners group for comment. We have incorporated the industry's comments where appropriate; however all comments received were minor, and no objections to the major thrust of our conclusion or the guidelines were received.

The resolution provided in the enclosures is intended to ensure RCPs are tripped for SBLOCAs but remain running for non-LOCA transients, including steam generator tube ruptures up to the design event (one tube). This is consistent with the lessons learned from the Ginna event as we have earlier discussed with the Commission.

The Chairman of the CRGR has concurred that committee review of this action is not required. The ACRS has been briefed on several occasions as this solution was developed and has voiced no objection. This concludes the development phase of TMI Action Plan Item II.K.3.5. The implementation of the task will be designated as a new multiplant item. We expect to complete the implementation in FY'83, making maximum use of generic solutions by the various owners groups. We also expect to issue our letters to licensees and applicants within the next few weeks.

William J. Dircks
Executive Director for Operations

Enclosures: As stated

ENCLOSURE I

DISCUSSIONS WITH INDUSTRY REPRESENTATIVES ON RCP TRIP DURING ACCIDENTS

BACKGROUND

Representatives of the PWR vendors and utilities were involved since 1979 in developing the current RCP trip setpoints. Now that we have finished our reevaluation of the basis for the present setpoints and the need to automate the operation, we have informally solicited the view of the vendors on our conclusions. In particular, we asked them to address:

- o whether the need to trip the RCPs was based on a real safety concern or if the only reason was due to Appendix K requirements and 10 CFR 50.46 limits,

- o if RCP trip is necessary, what actions have been taken to better define the action setpoints? In particular, how will proposed setpoints allow for continued pump operation for steam generator tube rupture events?

The responses we received to this line of inquiry and other views from the three PWR vendors and representatives of two owners' groups are summarized below.

Westinghouse (the owners' group did not attend meeting)

The Westinghouse position is that the RCP's should be tripped on low pressure to avoid unacceptable cladding temperatures for small break LOCAs. This provides Westinghouse with confidence that the consequences of these events, as predicted with present models, will be acceptable. Moreover, Westinghouse pointed out that continued RCP operation could not be assured for small break loss of coolant accidents. Westinghouse also recommended that the RCP trip be manual, citing Zion simulator data to show that operators can be relied upon to trip the RCPs within one minute after the trip criterion is reached.

The low pressure setpoint Westinghouse proposes to use is:

$$P_{\substack{\text{RCP}\\\text{Trip}}} = P_{\substack{\text{Steam Generator}\\\text{Safety Valve}}} + 100\text{ psi} + P_{\substack{\text{Wide Range Pressure}\\\text{Uncertainty}}}$$

A detailed discussion of the basis for this pressure is found in NUREG-0623. According to Westinghouse, this setpoint will result in RCP trip when the primary system depressurizes to 1300-1600 psia. They conclude that this setpoint will result in RCP trip for SBLOCAs of concern (>2" diameter), but will not result in RCP trip for most non-LOCA, overcooling transients and very small LOCAs.

For the steam generator tube rupture (SGTR) event, the Westinghouse plants which have safety injection pumps with high shutoff heads (i.e., SI pumps are charging pumps with P (shutoff) > 2500 psi), no RCP trip is expected for tube ruptures up to the present design basis (double-ended rupture of one tube). For plants which have SI pumps with lower shutoff heads, RCP trip is expected for the design base steam generator tube rupture, but not for smaller size ruptures. The exact rupture size above which RCP trip would be required is not yet known.

We have reviewed the spectrum of Westinghouse designs. The attached table summarizes the HPI characteristics of the plants licensed prior to TMI-2. From this table, it appears that 13 Westinghouse plants have HPI pumps with shutoff pressures above 2500 psi, 9 plants have HPI pumps with shutoff pressures below 1550 psi, and 3 plants have HPI pumps with shutoff pressures around 2170 psi.

We have informed Westinghouse that we do not believe the RCPs should be tripped for SGTR events such as Ginna (which was essentially equivalent to a design basis SGTR), and that they should examine methods for either improving the RCP trip setpoints or modifying the plants so that RCPs need not be tripped for design basis SGTRs. In granting restart permission for Ginna, we have required that supplementary permissives for RCP trip be developed to provide assuance of continued RCP operation for the design basis steam generator tube rupture event. Westinghouse also indicated that in the longer term, the probability of maintaining RCP operation during non-LOCAs can be improved by utilization of the reactor vessel liquid inventory system (RVLIS). We agree.

<u>Combustion Engineering</u> (chairperson of owners group analysis subcommittee attended)

Combustion Engineering has also concluded that for small break LOCAs, the RCPs should be tripped. CE has evaluated four possible schemes for RCP operation, as follows:

a) Trip all RCPs on Safety Injection Signal and then restart if event is not a LOCA;

b) Keep RCPs running for depressurization events;

c) Delay RCP trip until event is diagnosed. Trip RCPs if LOCA, leave running if non-LOCA;

d) Turn off two RCPs upon Safety Injection Signal. Diagnose event; then, if LOCA, turn off two remaining RCPs.

CE concluded that option d) is preferred. It maintains forced convection until the event is diagnosed, but reduces the flow sufficiently so that the time available for operator action is significantly extended.

CE does not recommend option b) since it could result in core damage if the RCPs stopped at some time during the accident event with best estimate assumptions. Moreover, CE says RCP damage is probable, citing concerns with bearings, seals, vibrations, and cooling water. CE does not recommend option a) for the same reasons that we are not satisfied with this option, even though it is the present scheme. The problem is that it trips pumps for events like SGTR events where it is probably unnecessary and compounds the problems of the operators. Finally, CE does not recommend option c) since it is difficult to find a set of symptoms which can be quickly diagnosed by the operator to distinguish between a LOCA and non-LOCA transients, and late operator action to trip the RCPs could result in unacceptable uncovering of the core.

CE has not performed many best estimate analyses, but limited work shows that continued pump operation results in acceptable consequences regardless of RCP trip delay time if best estimate conditions prevail and analysis models are correct. But with failure of one HPI pump there is unacceptable core heatup for a range of hot leg break sizes (2" to 4") as well as for other break sizes and locations.

Babcock and Wilcox (Two representatives of the B&W "Owners Group" attended)

B&W proposed to continue to trip the RCPs for small break LOCAs. The need to trip RCPs is considered to be a real one, and not simply a figment of Appendix K. B&W has performed one realistic calculation for a small break LOCA (i.e., two HPI pumps, 1.0 x ANS decay heat, realistic axial core power shape), and the cladding temperature results were in the 1900°F to 2000°F range when the RCP trip was delayed. B&W cannot conclude that a break does not exist for which the peak cladding temperature exceeds 2200°F if the RCPs are tripped at the most inopportune time. This can be alleviated by the addition of an automatic trip or adequate manual trip procedures.

B&W proposed to trip the RCPs on loss of subcooling margin as indicated by the subcooling meter. They conclude this would not result in tripping the RCPs for steam generator tube ruptures for best estimates of their plant's response. Moreover, B&W pointed out that the new emergency procedure guidelines have criteria for a more rapid RCP restart.

B&W was not prepared to discuss whether or not the RCP trip should be manual or automatic at the meeting.

Summary

The three PWR vendors continue to recommend that the reactor coolant pumps be tripped for LOCAs. The recommendation is based on:

- o The need to trip RCPs is a real safety concern and not an artificiality of Appendix K and 10 CFR 50.46.
- o A concern that RCP functionality cannot be assured if the pumps are run for extended periods in a voided system, and they may be needed later on in a severe accident.

Very few "best estimate" SBLOCA analyses with delayed RCP trip have been performed by the industry. Conceivably, these best estimate analyses might show that SBLOCA with delayed RCP trip results are acceptable. However, the vendor representatives were not very encouraged that acceptability for the entire spectrum of SBLOCA's would be found, and that at least some small breaks would result in unacceptable consequences. Even if the results for the entire SBLOCA were found acceptable, "the acceptability" would likely be marginal (i.e., the core would uncover and high cladding temperatures and oxidation would result). For certain failures, in particular the failure of one HPI pump to start, the margin would be reduced and in some instances would probably result in unacceptable core heatup.

As we have known, Appendix K does not provide the same degree margin for small break LOCAs as it does for large break LOCAs. For small breaks, the only significant conservatisms are the prescription on decay heat and power peaking and the assumption of the worst single failure. We also know that because of inherent design differences among the three PWR vendors, the optimum RCP trip setpoints will not be universal.

It appears that the CE proposal will retain forced convection for all events except the LOCA. The B&W proposal appears to have merit in that it should preclude RCP trip for almost all non-LOCA events except severe steam line breaks and steam generator tube ruptures beyond the design basis. The Westinghouse proposal may be acceptable for plants with high head SI pumps, since it should not result in RCP trip for design basis SGTR events. However, the most recent estimates in wide-range pressure measurement uncertainty by Westinghouse indicate that analyses confirming their conclusions are probably necessary. For the Westinghouse plants with low head SI pumps, the RCP's will probably continue to be tripped for SGTRs such as occurred at Ginna. With the present setpoints proposed by Westinghouse, we have informed

Westinghouse we do not believe we will find this acceptable, and will require them to develop better setpoints for tripping RCPs. We expect the Westinghouse owners to reply to this request after the letters exemplified by Enclosure Two have been transmitted.

On arriving at the pump trip setpoints we recommend that the licensees utilize event trees to systematically evaluate their setpoints to minimize the potential for undesirable consequences due to a misdiagnosed event. Specifically, the setpoints should be evaluated for events where the pumps are tripped when it is preferable they remain operational. They should also be evaluated for the case when the pumps are not tripped early in the event and a delayed trip may lead to undesirable consequences.

Utility/Plant	Shut Off Head, ft/psig	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	Positive Displacement Changing Pump Capacity, gpm
Alabama Power Co., Farley 1	6000/2600	750	550	-	250/325
Carolina Power & Light, H. B. Robinson	3300/1430	350	0	231	N/A
Commonwealth Edison, Zion 1&2	6000**/2600	490**	300**	-	165/230
Connecticut Yankee, Haddam Neck	6000**/2940	~800**	575**	-	325/380
Consolidated Edison, Indian Pt. 2	3500**/1517	405	0	294	N/A
Power Authority of State of New York Indian Pt. 3	3500**/1517	405	0	294	N/A
Duquesne Light, Beaver Valley 1	5050/2536	495	300	-	175/225
Florida Power & Light, Turkey Pt. 3&4	3500/1517	410	0	231	N/A
Indiana & Mich. Electric, D.C. Cook 1	5000/2514	560	400	-	70/225
D.C. Cook 2	5000/2514	560	400	-	170/225
Northern States Power Prairie Is. 1&2	5000/2168	760	600	100	N/A
Portland Gen. Electric, [illegible]	600	495	300	-	100/230

Utility Plant	Shut Off Head ft/psig	gpm @ 1000 psig	gpm @ 1600 psig	gpm @ Safety PORV Set Pt.	Positive Displacement Charging Pump Capacity, gpm
Public Service Electric & Gas, Salem 1	6160/2670	490	300	-	150/210
Rochester Gas & Electric, Ginna	3426/1485	205	0	100	N/A
Southern Calif. Edison, San Onofre 1	6000/2600	570	460	-	165/300
Virginia Elec. Power, Surry 1&2	6000/2600	520	420	-	255/250
North Anna 1	5900/2550	650	550	-	250/325
Wi[illegible]sin Electric [illegible], Point Beach 1&2	3550/1539	900	0	100	N/A
Wisc. Pub. Serv., Kewaunee	5000/2167	750	500	100	N/A
Yankee Atomic Electric, Yankee Rowe	1950/844	0	0	99	N/A

†Data not provided
*To be increased to 78.7 about June 1979
**Charging pump (centrifugal)

ENCLOSURE II

Letters to Licensees and Applicants

1. Cover letter to W licensees

2. Cover letter to B&W licensees

3. Cover letter to CE licensees

4. Cover letter to Yankee Atomic

5. Enclosure to above letters

TO: All licensees and Applicants with Westinghouse Designed NSSSs

The purpose of this letter is to inform you of the staff's conclusions regarding the analysis of LOFT Test L3-6 submitted by the Westinghouse Owner's Group, the continued acceptability of the Westinghouse ECCS evaluation model for predicting small break LOCAs with RCP operation, and criteria for resolution of the RCP trip issue.

We have completed our evaluation of the analyses of LOFT Test L3-6 performed by the Westinghouse Owner's Group and conclude the evaluations acceptably predict the test results. Therefore, we find the currently approved W evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50. For the case of limited RCP operation after reactor trip and for the range of licensed Westinghouse reactor designs, such evaluations show compliance with 10 CFR 50.46. Your plant should be assessed to assure that pump trip capability is satisfactory.

The enclosure to this letter provides guidance for the development of satisfactory setpoints for reactor coolant pump trip in your plant. As stated in the enclosure, manual tripping of the RCPs for LOCA can be allowed under certain conditions.

For plants with low head SI pumps, we understand that RCP trip is still expected to occur on the low pressure trip setpoints presently proposed by W for the design basis steam generator tube rupture. The staff does not find this acceptable and these licensees should identify a more distinguishing criterion for RCP trip that would allow continued RCP operation for tube leaks up to the design basis steam generator tube rupture.

Please provide, within 60 days of receipt of this letter, your plans and schedule for resolving this issue for your plant and submitting the required information in accordance with our guidance. For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual plants, we strongly urge you to work collectively with owners of plants similar to yours (i.e., owners groups) and propose setpoints and methods for RCP trip consistent with other licensees.

When you develop pump trip setpoints which you believe substantially meet the guidance provided in the enclosure, we encourage you to begin implementation of these new setpoints at your operating plant(s) prior to staff review and approval of your formal information submittal.* We caution that careful judgment should be used when developing your proposed setpoints in accord with the guidance in the enclosure. On arriving at the pump trip setpoints we recommend that you ut lize event trees to systematically evaluate your setpoints to minimize the potential for undesirable consequences due to a misdiagnosed event. Specifically, the setpoints should be evaluated for events where the pumps could be tripped when it is preferable they remain operational.

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*Unless for your plant such implementation entails a change of technical specifications or an unreviewed safety question, which require NRC approval prior to implementation.

They should also be evaluated for the case when the pumps are not tripped early in the event and a delayed trip may lead to undesirable consequences. To this effect, we will be pleased to discuss any questions you may have on the enclosed guidance in order to assist your early implementation efforts.

The requirements set forth in this letter supercede the actions required in IE Bulletins 79-05C and 79-06C.

If you believe further clarification regarding this issue is necessary or desirable please contact your NRC project manager.

Sincerely,

Darrell Eisenhut, Director
Division of Licensing, NRR

TO: All licensees and applicants with Babcock and Wilcox-Designed NSSSs

Dear :

The purpose of this letter is to inform you of the staff's conclusions regarding your analysis of LOFT Test L3-6, the continued acceptability of your ECCS evaluation model for predicting small break LOCAs with RCP operation, and criteria for resolution of the RCP trip issue.

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude your evaluations acceptably predict the test results. Therefore, we find the currently approved B&W evaluation model for small breaks LOCAs in continued conformance with Appendix K to 10 CFR 50 subject to the need for additional experimental verification discussed below. For the case of limited RCP operation and for the range of licensed B&W reactor designs, such evaluations show compliance with 10 CFR 50.46. Your plant should be assessed to assure that trip capability is satisfactory.

As you are aware, we have been studying the need for additional experimental verification for the B&W small break LOCA model which you reference in your analyses. At present, we have not been convinced that additional experimental data is not needed. However, we believe that this issue can be resolved separately from the RCP trip issue. Nevertheless, the additional experimental verification issue must be promptly resolved to the staff's satisfaction.

The enclosure to this letter provides guidance for the development of satisfactory setpoints for reactor coolant pump trip in your plant. As stated in the enclosure, manual tripping of the RCPs for LOCA can be allowed under certain conditions.

Please provide, within 60 days of receipt of this letter, your plans and schedule for resolving this issue for your plant and submitting the required information in accordance with our guidance. For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual plants, we strongly urge you to work collectively with owners of plants similar to your (i.e., owners groups) and propose setpoints and methods for RCP trip consistent with other licensees.

When you develop pump trip setpoints which you believe substantially meet the guidance provided in the enclosure, we encourage you to begin implementation of these new setpoints at your operating plant(s) prior to staff review and approval of your formal information submittal.* We caution that careful judgment should be used when developing your proposed setpoints in accord with the meet our guidance in the enclosure. On arriving at the pump trip setpoints we recommend that you utilize event trees to systematically evaluate your setpoints to minimize the potential for undesirable consequences due to a misdiagnosed event. Specifically, the setpoints should be evaluated for

*Unless for your plant such implementation entails a change of technical specifications or an unreviewed safety question, which require NRC approval prior to implementation.

events where the pumps could be tripped when it is preferable they remain operational. They should also be evaluated for the case when the pumps are not tripped early in the event and a delayed trip may lead to undesirable consequences. To this effect, we will be pleased to discuss any questions you may have on the enclosed guidance in order to assist your early implementation efforts.

The requirements set forth in this letter supercede the actions required in IE Bulletins 79-05C and 79-06C.

If you believe further clarification regarding this issue is necessary or desirable please contact your NRC project manager.

Sincerely,

Darrell Eisenhut, Director
Division of Licensing, NRR

TO: All licensees and applicants with Combustion Engineering Designed NSSSs

Dear :

The purpose of this letter is to inform you of the staff's conclusions regarding the analysis of LOFT Test L3-6 submitted by the Combustion Engineering Owner's Group, the continued acceptability of the Combustion Engineering ECCS evaluation model for predicting small break LOCAs with RCP operation, and criteria for resolution of the RCP trip issue.

At a meeting with the staff on April 28, 1981, CE and the CE Owners Group presented the results of their calculations of the LOFT L3-6 test. This test was a small break simulation in which the reactor coolant pumps remained operational. During the meeting, the staff resolved a number of questions involving the ability of the CE model to predict small break behavior with the pumps operational. Based on our review of the submittals, and the information provided at the April 28, 1981, meeting, we have, with one exception, concluded that the currently approved CE evaluation model for small break LOCAs has acceptably calculated the results of LOFT Test L3-6. The one exception results from our conclusion that you have not yet provided sufficient information to demonstrate that the use of a conservative bubble rise model in the core region always results in a conservative peak cladding temperature even though it does not result in a conservative coolant inventory prediction. This matter must be promptly resolved to the satisfaction of the staff.

When we receive information needed to confirm your modeling assertion, we will be able to find the currently approved CE evaluation model for small break LOCAs in continued conformance with Appendix K to 10 CFR 50. For the case of limited RCP operation after reactor trip and for the range of licensed CE plants, the evaluations based on the CE model show compliance with 10 CFR 50.46. Your plant should be assessed to assure that trip capability is satisfactory.

The enclosure to this letter provides guidance for the development of satisfactory setpoints for reactor coolant pump trip in your plant. As stated in the enclosure, manual tripping of the RCPs for LOCA can be allowed under certain conditions.

Please provide, within 60 days of receipt of this letter, your plans and schedule for resolving this issue for your plant and submitting the required information in accordance with our guidance. For the purposes of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual plants, we strongly urge you to work collectively with owners of plants similar to yours (i.e., owners groups) and propose setpoints and methods for RCP trip consistent with other licensees.

When you develop pump trip setpoints which you believe substantially meet the guidance provided in the enclosure, we encourage you to begin implementation of these new setpoints at your operating plant(s) prior to staff review and approval of your formal information submittal.* We caution that careful judgment should be used when developing your proposed setpoints in accord with the guidance in the enclosure. On arriving at the pump trip setpoints we recommend that you utilize event trees to systematically evaluate your setpoints to minimize the potential for undesirable consequences due to a misdiagnosed event. Specifically, the setpoints should be evaluated for events where the pumps could be tripped when it is preferable they remain operational. They should also be evaluated for the case when the pumps are not tripped early in the event and a delayed trip may lead to undesirable consequences. To this effect, we will be pleased to discuss any questions you may have on the enclosed guidance in order to assist your early implementation efforts.

The requirements set forth in this letter supercede the actions required in IE Bulletins 79-05C and 79-06C.

If you believe further clarification regarding this issue is necessary or desirable, please contact your NRC project manager.

Sincerely,

Darrell Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

*Unless for your plant such implementation entails a change of technical specifications or an unreviewed safety question, which require NRC approval prior to implementation.

TO: Yankee Atomic Electric Company

Dear:

The purpose of this letter is to inform you of the staff's conclusions regarding your analysis of LOFT Test L3-6, the continued acceptability of your ECCS evaluation model for predicting small break LOCAs with RCP operation, and criteria for resolution of the RCP trip issue.

We have completed our evaluation of your analyses of LOFT Test L3-6 and conclude your evaluations acceptably predict the test results. Therefore, we find the currently approved YAEC evaluation model for small breaks LOCAs in continued conformance with Appendix K to 10 CFR 50 for the case of limited RCP operator after reactor trip.

The enclosure to this letter provides guidance for the development of satisfactory setpoints for reactor coolant pump trip in your plant. As stated in the enclosure, manual tripping of the RCPs for LOCA can be allowed under certain conditions.

Please provide, within 60 days of receipt of this letter, your plans and schedule for resolving this issue for your plant and submitting the required information in accordance with our guidance. For the purpose of providing uniformity of setpoints and methods and for minimizing potential confusion that could arise because of diverse actions by individual plants, we strongly urge you to work collectively with owners of plants similar to yours (i.e., owners groups) and propose setpoints and methods for RCP trip consistent with other licensees.

When you develop pump trip setpoints which you believe substantially meet the guidance provided in the enclosure, we encourage you to begin implementation of these new setpoints at your operating plant(s) prior to staff review and approval of your formal information submittal.* We caution that careful judgment should be used when developing your proposed setpoints in accord with the guidance in the enclosure. On arriving at the pump trip setpoints we recommend that you utilize event trees to systematically evaluate your setpoints to minimize the potential for undesirable consequences due to a misdiagnosed event. Specifically, the setpoints should be evaluated for events where the pumps could be tripped when it is preferable they remain operational. They should also be evaluated for the case when the pumps are not tripped early in the event and a delayed trip may lead to undesirable consequences. To this effect, we will be pleased to discuss any questions you may have on the enclosed guidance in order to assist your early implementation efforts.

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*Unless for your plant such implementation entails a change of technical specifications or an unreviewed safety question, which require NRC approval prior to implementation.

The requirements set forth in this letter supercede the actions required in IE Bulletins 79-05C and 79-06C.

If you believe further clarification regarding this issue is necessary or desirable please contact your NRC project manager.

Sincerely,

Darrell Eisenhut, Director

Division of Licensing, NRR

ENCLOSURE

Resolution of RCP Trip Issue

The NRC, its licensees, and the PWR vendors have been evaluating the RCP trip issue since the accident at TMI. The technical understanding of the industry and the requirements of NRC have changed twice in that period. As a result, there have been extensive studies to better understand the dynamic response of all classes of PWR to small break LOCAs. Although some confirmatory information is still to be received concerning some models, we conclude that the analytical models are sufficiently reliable to be used by licensees to choose their own best method to assure that reactor coolant pumps (RCP) are tripped upon indication that a LOCA has occurred. The tripping of RCPs upon confirmation of a LOCA is a generally accepted practice because of(1) the need to limit system inventory loss during small break LOCAs, and (2) the need to assure that pump performance and integrity will not be degraded by operation in a highly voided system. Because the tripping of RCPs is generally accepted practice, the NRC will no longer require any particular method of RCP tripping.

In addition, the NRC staff recognizes that with certain design additions (e.g., additional HPI pump), continued RCP operation during LOCA may be possible and that option is available to licensees on a voluntary basis for those who would prefer the operational flexibility that would accrue to a system design for continous reactor coolant pump operation in the event of a LOCA.

Since our criteria development has concentrated on RCP trip, and since no vendor or utility licensee has to date proposed continuous RCP operation during a LOCA, detailed criteria for demonstrating the acceptability of continuous RCP operation have not been developed. If you intend to pursue this option, you should notify the staff early so that the design criteria can be agreed upon before significant utility resources are committed.

The staff has also concluded that if sufficient time exists, then manual action is an acceptable means for tripping the RCPs following a LOCA. We have based this conclusion in part upon our own probabilistic assessment. It showed that the failure of a designated operator to trip the RCPs within five minutes following receipt of a RCP trip signal is approximately six times more likely than is the failure of an automatic trip. Our probabilistic assessment was limitied by a lack of comprehensive information about the complex interrelationship among break size, break location, RCP trip delay time, and peak cladding temperature (PCT) for each type of NSSS. A complete map of this interrelationship for each design would be prohibitively expensive to generate (tens of computer runs for each design at thousands of dollars per run and hundreds of hours of analyst time). Without such a map, we cannot accurately define the bounds of the region where unacceptable consequences might result from delay in RCP trip. However, based on our understanding of the phenomena in question, analyses performed by the NSSS vendors, limited independent analyses performed by the staff, tests performed in both Semiscale and LOFT, and our probability assessment, we conclude that allowing manual RCP trip is acceptable provided certain conditions are satisfied. Our guidelines for RCP trip setpoints and methods are set forth below. We request that you modify your RCP trip setpoints and methods to best satisfy these guidelines and report back on your final conclusions. We do not anticipate a need for further regulatory action on this matter except for those licensees that would propose not to trip the RCPs because of the design modification option mentioned above.

In developing your RCP trip setpoints and methods, you should be especially mindful of two potential problems with RCP trip that continue to show up in reactor operations. The first problem is caused by the fact that the loss of pressurizer sprays upon RCP trip for transients and for small break LOCAs results in a need in some plants to use PORVs for primary system pressure control. Despite extensive testing of prototypes and improved reliability engineering, these valves continue to show a high propensity for failing to close. Although the question of PORV functionality has been better characterized by the EPRI valve testing program since the accident at TMI, there does not appear to be significant progress in improving the overall operational reliability of PORV systems. A second problem associated with RCP trip is that it tends to produce a stagnant region of coolant in the upper elevations of the reactor vessel. In a number of recent operational events, this hot, stagnant fluid has flashed and partially voided the upper vessel region during depressurization or cooldown situations. Despite wide dissemination of information about these operating events and the learning opportunities that they present, we still perceive that operators (1) are not completely familiar with the significance of a steam bubble in the upper head, (2) have difficulty controlling coolant conditions so as to avoid or control flashing where possible, and (3) may have a tendency to take precipitous actions when a steam bubble exists. We are particularly concerned about the implications for pressurized thermal shock if operators repressurize the primary system using safety injection or charging pumps during attempts to condense a steam bubble in the primary system.

In developing your RCP trip setpoints and methods, the following guidelines should be followed:

1. <u>Setpoints for RCP Trip</u>

a) The setpoints should be designed to assure that the RCPs will be tripped for all losses of primary coolant. The setpoints should also ensure continued forced RCS flow during steam generator tube ruptures up to and including the design basis tube rupture. Safety analyses should be performed to demonstrate the achievement of these goals. The symptoms and signals used to alert an operator of the need to manually trip RCPs should be, to the extent possible, uniquely attributable to LOCAs and not other depressurizing transients and actions for which continued pump operation is desirable. In this regard, consideration should be given to partial or staggered RCP trip schemes (e.g., in two loop, four pump plants, trip one pump per loop immediately and trip remaining pumps once the existence of a LOCA is confirmed). If selected pumps are tripped during the initial phase of the transient, licensees should assure that training and procedures provide direction for use of individual steam generators with and without RCPs in operation. Representative analyses should be performed to demonstrate that the proposed RCP trip setpoints are adequate for small LOCAS but will not result in RCP trip for other non-LOCA transients and accidents (e.g., steam generator tube ruptures).

b) The RCP trip setpoints should be selected so as to exclude extended RCP operation in a voided system (e.g., pump head degradation > 10%) unless engineering analysis or tests are available to justify that pump and pump seal integrity will be maintained under those conditions.

c) If, for some transients and accidents within the current design basis, and with offsite power available, the setpoints selected by the licensees will lead to RCP trip even though it is neither required nor desirable, then systems analyses and operating procedure evaluation should be conducted to assure that these events will not result in challenges , either automatic or from the operators, to the PORVS to accomplish depressurizing actions normally accomplished by pressurizer sprays. Heated auxiliary spray capability not derived from RCP discharge pressure should be considered as one possible means of eliminating this reliance on the PORVs. On the other hand, if PORV operation is continued to be recommended for use in depressurization, then the licensee should develop a program for upgrading the operational reliability of the PORVs.

d) For any conditions which require or result in RCP trip and the establishment of a hot, stagnant, fluid region at high points in the primary system, emergency procedure guidelines and emergency procedures should specifically describe symptoms of primary system voiding due to flashing of stagnant regions of hot coolant. They should also contain specific guidance on detecting, managing and removing the coolant voids that result from flashing. Operator training programs should specifically address the significance of primary system voids under non-LOCA and LOCA conditions.

e) Transients and accidents which produce the same initial symptoms as a LOCA (i.e., depressurization of the reactor and actuation of engineered safety features) and result in containment isolation may result in the termination of systems essential for continued operation of the reactor coolant pumps (i.e., component cooling water and/or seal injection water). It was the intent of TMI Action Plan Item II.E.4.2 to have licensees reevaluate essential and non-essential systems with respect to containment isolation. In particular, if a licensee's design terminates water services essential for RCP operation, then the licensee should assure that these water services can be restored in a timely manner once a non-LOCA situation is confirmed, and prevent seal damage or failure.

Licensees should confirm that containment isolation with continued pump operation will not lead to seal or pump damage or failure.

f) Parameters used to determine when RCPs should be tripped should provide unambiguous indicators of a LOCA. The inadequate core cooling instrumentation required by the Commission and described in NUREG-0737 should be factored into the emergency procedure guidelines where useful in indicating the need for RCP trip. Also of note is the recent experience from the LOFT facility confirming that pump current is a potentially useful parameter for distinguishing a LOCA from other transients and accidents. A recent report on this subject from the LOFT program is enclosed for your information and use.

2. <u>Guidance for Justification of Manual RCP Trip</u>

Our review of this subject leads us to conclude that it is preferable to manually (rather than automatically) trip the reactor coolant pumps where it is at all possible to justify it. However, our review indicates that there may be a few plants for which it is not possible to justify manual trip. The information requested below is intended to develop complete justification for those plants that can and should rely on manual trip and to clearly identify those few plants that may not be able to rely on manual trip.

a) Based on the RCP trip setpoints developed according to the guidance in item one above, provide analyses* and demonstrate that the limits set forth in 10 CFR 50.46 are not exceeded for the limiting small break size and location. For the purposes of showing compliance with 10 CFR 50.46, operator action to trip the RCPs should be assumed no earlier than two (2) minutes following the onset of reactor conditions corresponding to the RCP trip setpoint. Allowances should be made for instrument error.

b) If manual RCP trip is proposed, then for the limiting small break size(s) and location(s) identified from (a) above, provide a most probable** best estimate analysis of the amount of time available to the operator to trip the RCPs following the existence of the RCP trip signal. If this time is less than that recommended in Draft ANSI Standard N660, justify the acceptability of this time. Please include an evaluation of operating experience data when addressing this justification. Discuss the consequences if RCP trip is delayed beyond this time. Describe contingency procedures available to the operator in the event the RCPs are

*We will accept generic analyses of general reactor types in lieu of plant specific analyses. The generic analyses should be shown to bound plant specific evaluations.

**Each licensee should identify and justify the most probable plant conditions. Conservative estimates are acceptable in the absence of justifiable most probable plant conditions.

not tripped in the preferred time frame. If the time available is in excess of the standard, no further justification is necessary.

d) If the guidance set forth in (a), (b), and (c) above cannot be met, we will consider your further justification for allowing manual RCP trip on a case-by-case basis.

3. Other Considerations:

Although we do not intend to specify acceptance criteria in the following areas, we do require assurrance that they have been considered and good engineering practice has been followed.

a) For the parameter(s) employed in your RCP trip setpoint, describe the level of quality you intend to establish for the instrumentation that will signal the need for RCP trip. In particular, identify your basis for:

o The design features chosen for the sensing instruments (e.g., seismic and environmental qualifications, reliability, etc.),

o The degree of redundancy in the sensing instruments,

b) Identify the emergency operating procedures for the timely restart of the reactor coolant pumps when conditions which will support safe pump operation are established.

c) Describe your training program, to instruct operators in their responsibility for performing RCP trip in the event of a SBLOCA. In particular, discuss the training in prioritization of actions following engineered safety features actuation.

LOFT RESEARCH MEMORANDUM

REACTOR COOLANT PUMP MOTOR POWER OR CURRENT CRITERIA FOR REACTOR COOLANT SYSTEM INVENTORY MANAGEMENT IN COMMERCIAL PWR's DURING ACCIDENTS

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G. E. McCreery
J. E. Koske
B. E. Stitt

March 1982

1.0 Summary

One of the nuclear industry's concerns, since the Three Mile Island accident, has been the proper operation of the reactor coolant pumps (RCP's).[1] The generic issue is reactor coolant system (RCS) inventory control. In the course of experimentally investigating this concern, LOFT experiments were conducted which showed how RCP operation influenced inventory control. Based on vendor calculations supported by LOFT and Semiscale data, present procedures require tripping pumps prematurely in many reactor transients (small break loss-of-coolant accidents (LOCAs) and overcooling). This is undesirable as operation of the pumps aids in:

1. Maintaining pressure control (pressurizer spray).

2. Cooling the reactor core.

3. Minimizing risk of pressurized thermal shock (mixing and pressure control).

4. Providing head cooling and minimizing bubble development.

For these reasons, additional analysis has been performed to determine if an alternate pump trip criteria could be determined.

This memorandum reports research results obtained during the conduct of the Loss-of-Fluid Test (LOFT) Program, on reactor coolant pump (RCP) motor power and current measurements and their utility as indicators of RCS inventory. These results confirmed the previously held notion that pump motor power and current are related to RCS coolant density. Because of this relationship, RCP motor power and current are useful parameters for the operator to use in controlling RCS inventory. Use of the proposed methods will result in delayed tripping of pumps during a LOCA and have the following benefits in addition to those listed above:

1. Allows separation of overcooling transients from LOCA events,

for example, pumps will not be tripped during overcooling events.

2. Provides measurement of loop voiding as a trigger for RCP and high pressure safety injection (HPSI) control.

3. Allows the operator to leave the RCPs on during LOCAs when HPSI can make up the break flow.

4. Minimizes mass loss from the RCS.

5. Minimizes radiation release to environment.

6. Minimizes operator action uncertainty.

Analysis and experimental data are combined to develop the relationship between pump motor power and current and PCS inventory for the LOFT pumps. The results are characteristic of centrifugal pumps driven by constant speed induction motors. Since both the LOFT pumps[2.,3.] and the pumps in commercial PWRs are of this type, the results are also applicable to commercial nuclear power plants.

This paper contains the theory and supporting data, an operator display of pump motor power and cold leg temperature and the associated procedures and criteria to use in managing RCS inventory during an accident, the advantages of these criteria compared with the present criteria, and the additional activities which would compliment these results.

2.0 Relationship of RCP Motor Power and Current to Coolant Density at the Pump Inlet

The LOFT system has two, parallel RCPs in its single intact loop where a commercial PWR has a single pump in each loop. Consequently, the LOFT pump parameters in the following discussion are shown across both pumps (pressure drop) or the parameters from the two pumps are added together (pump motor power and motor current).

The differences between the same parameters from the separate pumps can be accounted for but are not significant during the first 300 s of the LOFT data which is the time of interest.

2.1 Analytical Relationships

The power required for a constant speed centrifugal pump in single-phase flow and two-phase bubbly flow is

$$\frac{P^P}{P_o^P} = \frac{\eta_o^P}{\eta^P} \frac{Q}{Q_o} \frac{H}{H_o} \frac{\rho}{\rho_o} \qquad \text{Equation 1}$$

Where

P^P = Pump power

Q = Volumetric flow rate

H = Pump head

ρ = Average coolant density in the pump impeller

η = Pump efficiency

The subscript denotes a reference condition.

The pump power is related to the pump motor power by

$$P = \frac{P^P}{\eta} \qquad \text{Equation 2}$$

Where

P = Motor power

η = Motor efficiency

Combining Equations 1 and 2 produces the relationship between pump motor power and coolant density,

$$\frac{\rho}{\rho_o} = \left[\frac{\eta_o^{\rho}}{\eta^{\rho}} \frac{Q_o}{Q} \frac{H_o}{H}\right]_{pump} \left[\frac{\eta}{\eta_o} \frac{P}{P_o}\right]_{motor} \qquad \text{Equation 3}$$

Pump motor current is also directly proportional to coolant density. Pump motor current and power are related by

$$P = IV \cos \theta \qquad \text{Equation 4}$$

Where I = RMS current

V = RMS voltage

θ = Phase angle

For an induction motor running at constant speed, the voltage is constant. Therefore,

$$\frac{P}{P_o} = \frac{\cos\theta}{\cos\theta_o} \frac{I}{I_o} \qquad \text{Equation 5}$$

The equations defining the analytical relationship between coolant density in single-phase and two-phase bubbly flow and RCP motor power and current are

$$\frac{\rho}{\rho_o} = \frac{Q_o}{Q}\,\frac{H_o}{H}\left[\frac{\eta}{\eta_o}\right]\frac{P}{P_o} \qquad \text{Equation 6}$$

and

$$\frac{\rho}{\rho_o} = \frac{Q_o\,H_o}{Q\,H}\left[\frac{\eta\cos\theta}{\eta_o\cos\theta_o}\right]\frac{I}{I_o} \qquad \text{Equation 7}$$

Equations 6 and 7 assume $n^P = n_o^P$ because the coolant Reynold's number in an RCP is sufficiently large.[5] The terms in brackets in both equations can be obtained from the pump motor manufacturer's specifications.

In general, the system volumetric flow rate and the pump head are not measured in commercial plants. However, the change in both parameters is relatively small in single and two-phase bubbly flow. Therefore, assume that

$$H_o = H$$

and

$Q_o = Q$ in Equations 6 and 7. The results are

$$\frac{\rho}{\rho_o} \cong \left[\frac{\eta}{\eta_o}\right]\frac{P}{P_o} \qquad \text{Equation 8}$$

and

$$\frac{\rho}{\rho_o} \cong \left[\frac{\eta\cos\theta}{\eta_o\cos\theta_o}\right]\frac{I}{I_o} \qquad \text{Equation 9}$$

Equations 8 and 9 are compared with L3-6 data in Figures 1 and 2. The agreement between the pump inlet density data and the theory is sufficient for prescribing criteria and specifying operator actions, as will be shown.

2.2 Analytical Relationship Limitations

Equations 6 and 7 are limited to single and two-phase flow until the bubbly-to-churn two-phase regime transition occurs. The system coolant transitions from single-phase to two-phase bubbly flow when the coolant saturates. The bubbly flow regime contains small bubbles homogeneously distributed and traveling at about the same velocity as the liquid. As the bubbles become larger, they coalesce causing a two-phase regime transition from bubbly to partially stratified churn turbulent flow.[4.] In churn turbulent flow, the vapor and liquid separate more distinctly and travel at different velocities. The result is a significant degradation in pump performance as shown in Figure 3. Figure 3 contains data from LOFT loss-of-coolant Experiment (LOCE) L3-6.[6.] At 30 s, the coolant saturated. Then between 30 and 100 s, the pump condensed the vapor entering the pump inlet before reaching the outlet. After 100 s, the coolant was two-phase from pump inlet to outlet. The bubbly-to-churn transition occurred 290 s (about 5 min) into the transient.

3.0 Parametric Display for Operator's Use

In the previous section, the pump motor power or current was shown to be a function of the pump inlet coolant density. The question is how can this information be displayed and used by the operator in making accident management decisions? This section discusses the operator display.

An operator's display of pump motor power and cold leg temperature is shown in Figure 4. The data in Figure 4 are from LOFT Experiments L6-7, a rapid secondary cooldown like a steam line break; L9-1, a

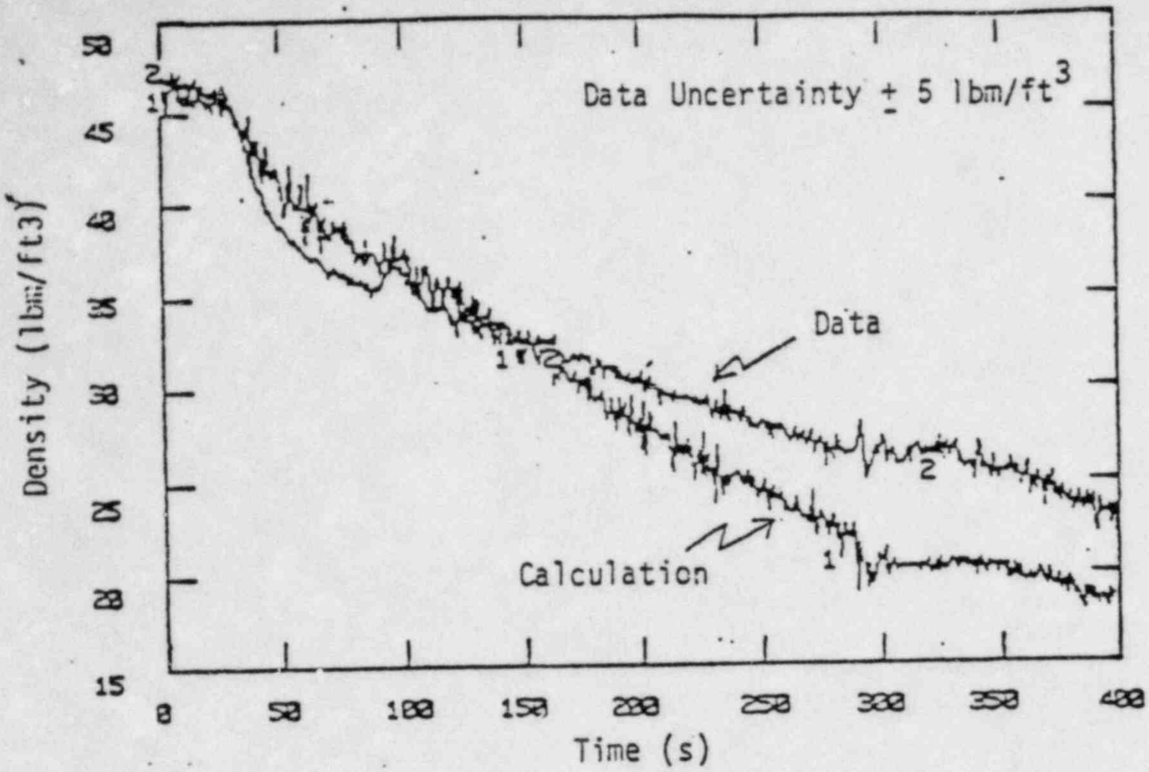

Figure 1. Measured Density Upstream of Pump Compared With Prediction Using Pump Motor Power Assuming Constant Volumetric Flow and Pump Head

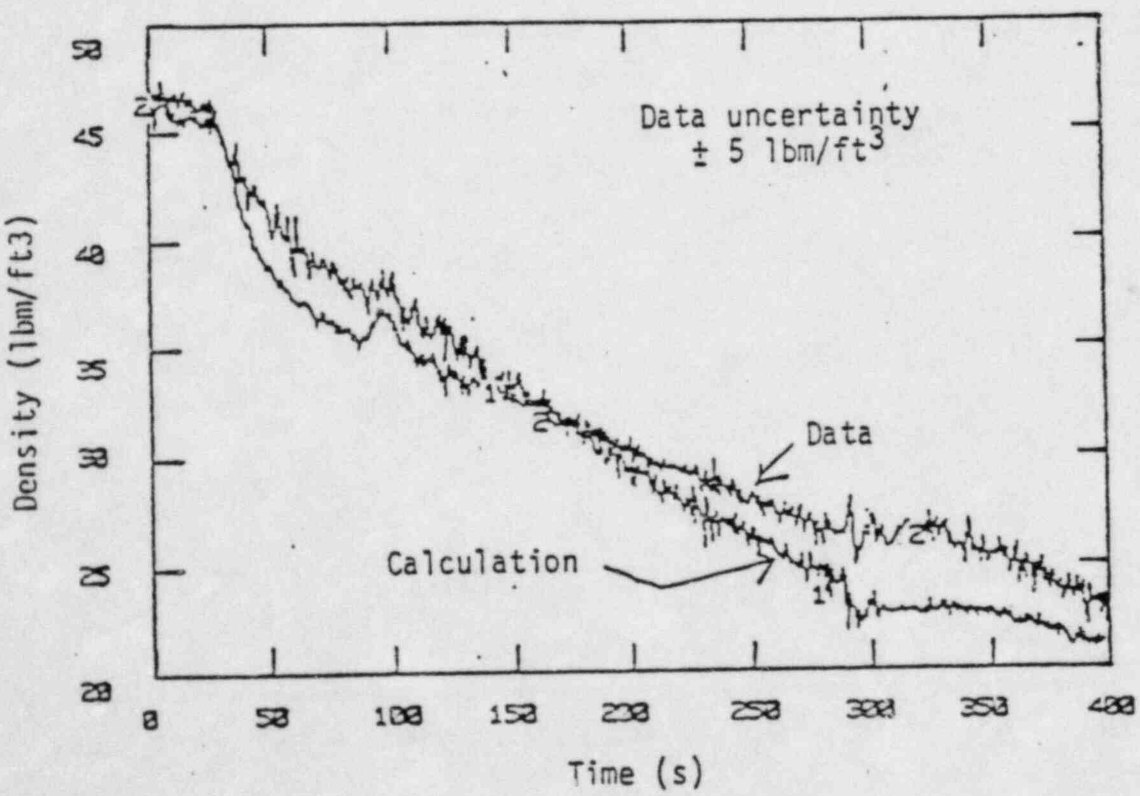

Figure 2. Measured Density Upstream of Pump Compared With Prediction Using Pump Motor Current Assuming Constant Volumetric Flow and Pump Head

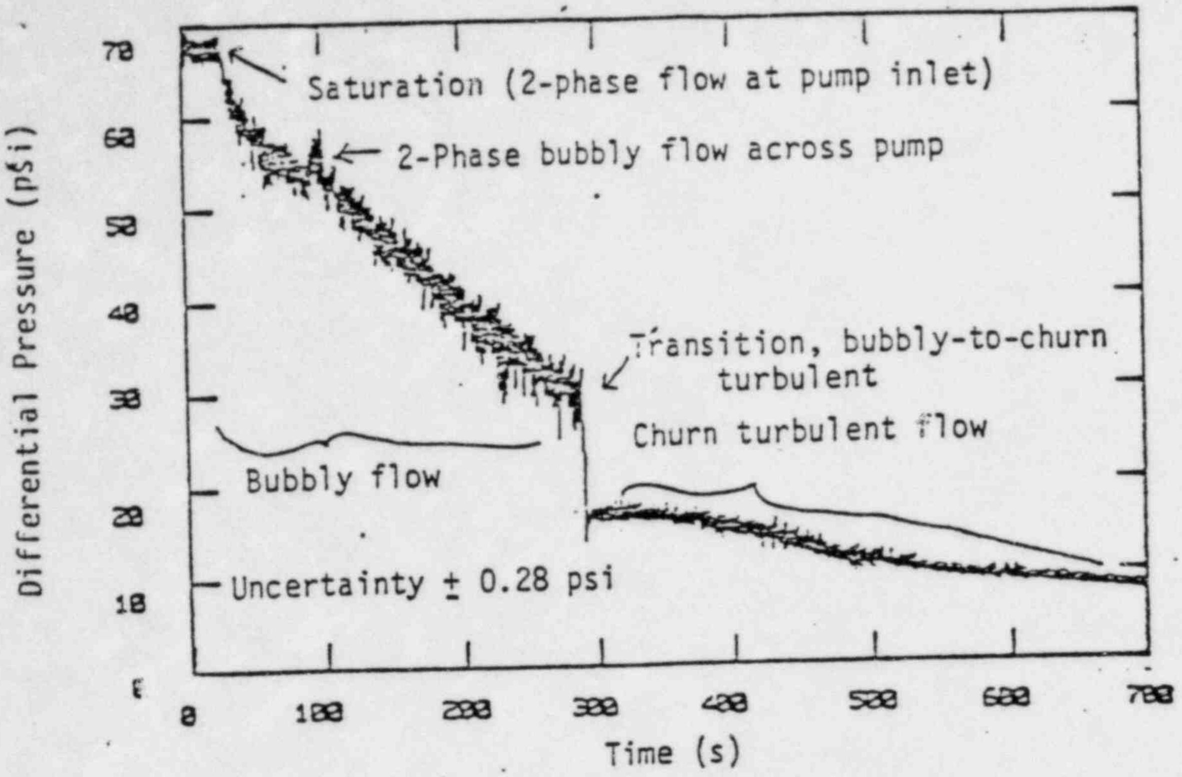

Figure 3. Differential Pressure Across Primary Coolant Pumps

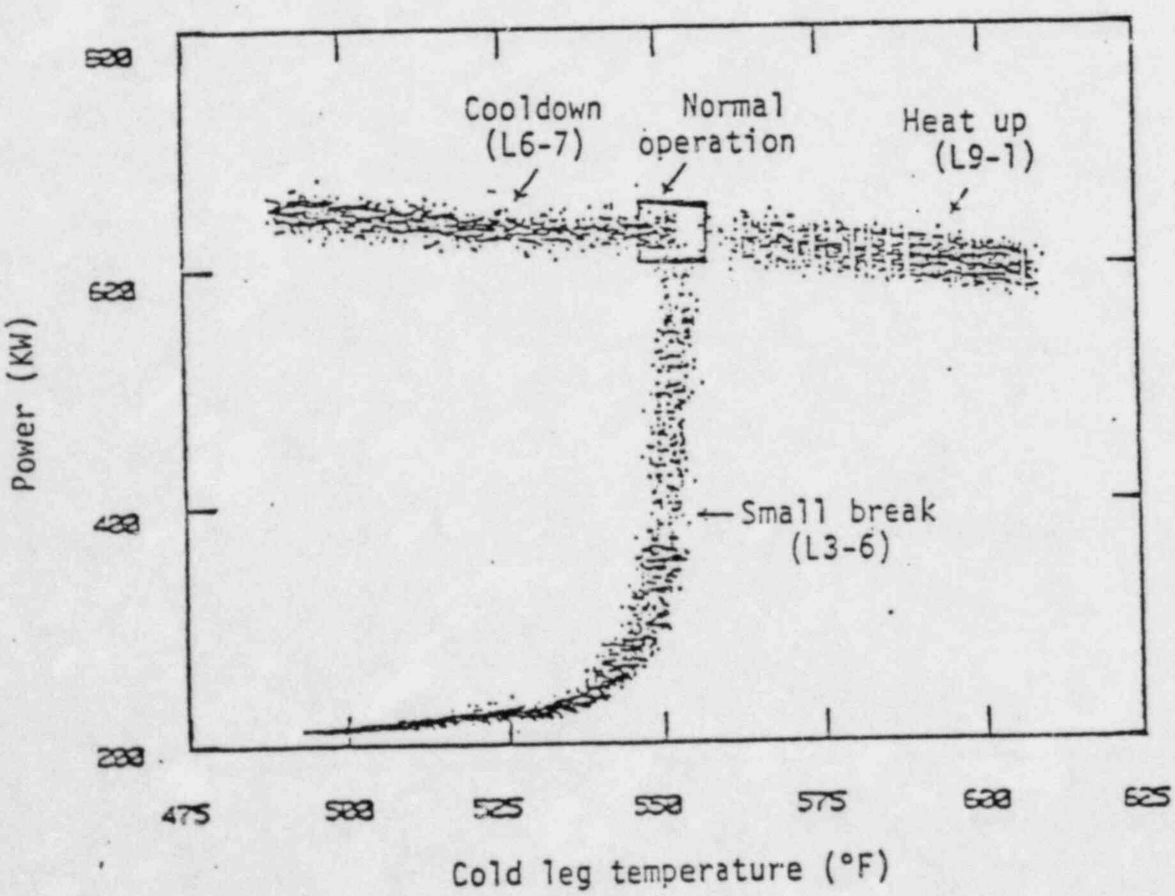

Figure 4. Operator's display with data from LOFT experiments

rapid heatup simulating the loss of all steam generators; and L3-6, a small break LOCA where the HPSI could not make up the system mass loss. Obviously, the pumps were on during all three tests.

The display allows the operator to unambiguously distinguish between a transient and a LOCA. Should multiple failures compound a transient with a break, the pump motor power will decrease from an off-normal operating point telling the operator a break has occurred.

Implementation of the display should not provide any unusual or unreasonable problems. Both cold leg temperature and pump motor power or current are measured in most if not all operating nuclear reactors. This display is being implemented on the LOFT system and will be used for control of future experiments.

The next step is to integrate the theory with the display to provide a basis for operating procedures and criteria. Using Equation 8 and the homogeneous representation of the density,

$$\rho = \rho_f - \alpha(\rho_f - \rho_g) \qquad \text{Equation 10}$$

Where ρ_f = Liquid density

ρ_g = Vapor density

α = Homogeneous void fraction

an equation for constant α can be derived and superimposed on the display in Figure 4. The equation is

$$P = \frac{P_0 \eta_0}{\rho_0 \eta}\left[\rho_f - \alpha(\rho_f - \rho_g)\right] \qquad \text{Equation 11}$$

The result is shown in Figure 5. (Table 1 contains a resume of equations.)

Lines of α = 0, .1, and .15 are shown on Figure 5 along with the data from the three LOFT experiments. In the next section, the procedures and criteria are presented to demonstrate how the operator would use the display to make decisions about RCP and HPSI operation.

TABLE 1

RESUME OF PUMP POWER AND CURRENT EQUATIONS FOR SINGLE PHASE AND TWO-PHASE BUBBLY FLOW

Pump Motor Power	Pump Motor Current
(6)* $\frac{\rho}{\rho_0} = \frac{Q_0\, H_0}{Q\, H}\left[\frac{\eta}{\eta_0}\,\frac{P}{P_0}\right]_{motor}$	(7) $\frac{\rho}{\rho_0} = \frac{Q_0\, H_0}{Q\, H}\left[\frac{\eta}{\eta_0}\,\frac{\cos\theta}{\cos\theta_0}\,\frac{I}{I_0}\right]_{motor}$

Assuming $Q = Q_0$ & $H = H_0$

(8) $\frac{\rho}{\rho_0} \cong \left[\frac{\eta}{\eta_0}\,\frac{P}{P_0}\right]_{motor}$	(9) $\frac{\rho}{\rho_0} \cong \left[\frac{\eta}{\eta_0}\,\frac{\cos\theta}{\cos\theta_0}\,\frac{I}{I_0}\right]_{motor}$

The operator display equation for constant void fraction is

(10) $$P = \frac{P_0\, \eta_0}{\rho_0\, \eta}\left[\rho_f - \alpha\,(\rho_f - \rho_g)\right]$$

* The numbers in parenthesis correspond to the equation numbers in text.

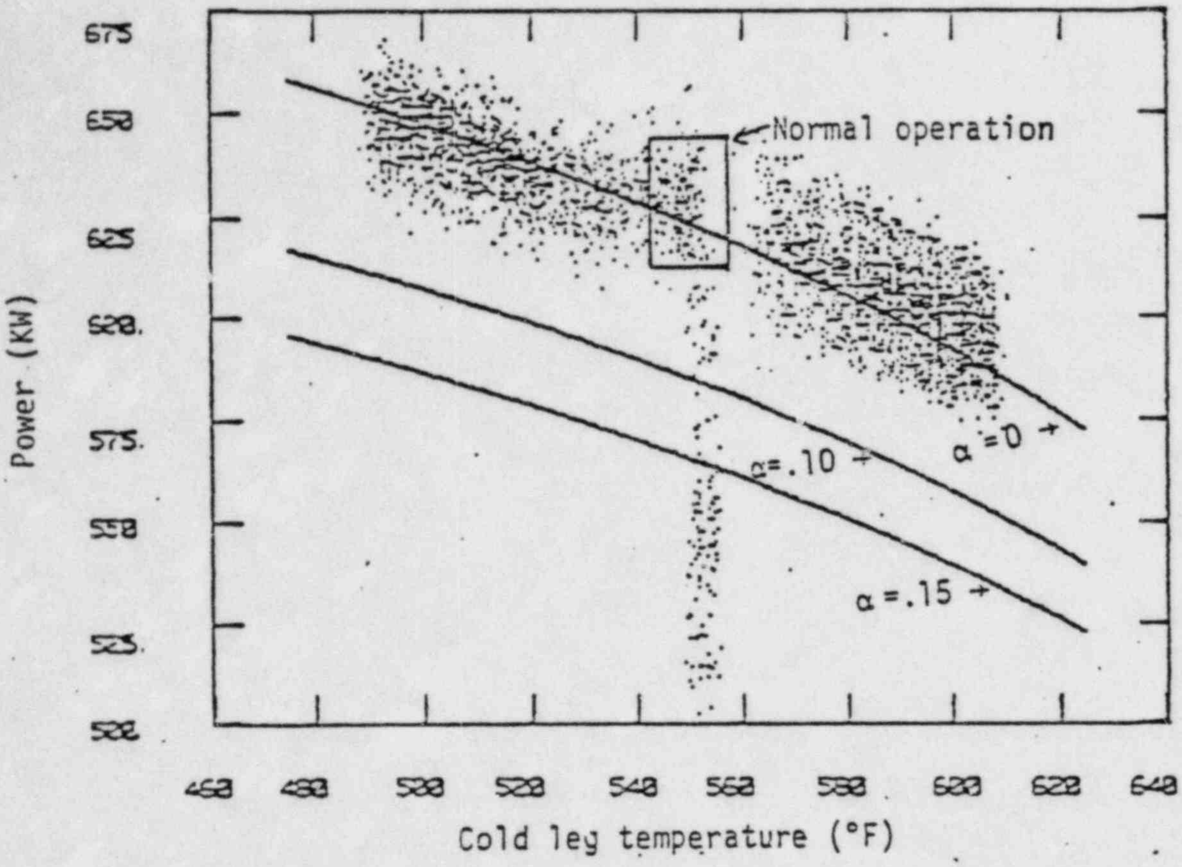

Figure 5. Operator's display with criteria lines and LOFT data

4.0 Proposed Accident Management Procedures and Criteria When Offsite Power Is Available

The use of pump motor power by the operator to make RCP and HPSI operation decisions assumes the availability of offsite power. This is more probable than not having offsite power available. Criteria, procedures, and the required measurements for the loss of offsite power situation are important but not consistent with the scope of this document.

4.1 The Information Needed By The Operator

If an event occurs in which the RCS pressure and reactor vessel liquid level start to decrease, the reactor trips inserting the control rods, and the HPSI comes on, the operator must determine what to do with the RCPs, turn them off or leave them on. If the event is a rapid cooldown, the RCPs should remain on to retain RCS pressure control using the pressurizer sprays and to aid in decay heat removal. If, however, coolant is being lost from the primary system, either to containment or the secondary system, it may be necessary to trip the RCPs to decrease the rate of inventory loss.

The probability of having a break large enough to warrant RCP trip is relatively low. In fact, none of the initiating events which have occurred in commercial reactors to date have required RCP trip to slow down inventory loss, including TMI.[7.] If HPSI flow exceeds break flow sufficiently early in the transient so that the core will remain covered with coolant, it is not only unnecessary but undesirable to trip the RCPs.

Therefore, the information needed by the operator is, can the HPSI system make up the inventory being lost by the break? If HPSI can make up the mass loss, the RCPs should be left on and the HPSI should be throttled to keep the pressurizer liquid level from getting too high. If the HPSI can't make the inventory loss, the RCPs should be tripped and the HPSI should remain full on.

4.2 Present Criteria and Instrumentation Are Not Optimum

The present criteria for RCP trip are known not to be optimum.[1] The RCPs are tripped on low RCS pressure somewhere between 1350 and 1650 psig depending on plant type. Another RCP trip criteria being considered is loss-of-subcooling, measured by the subcooling meter. The low pressure and loss-of-subcooling criteria have a common deficiency. Neither criterion tells the operator the information needed, namely, is the HPSI flow capable of making up the coolant being lost from the break or not? The reason is neither criterion is directly a function of RCS coolant density or inventory, as is pump motor power or current.

The results of using the low pressure or loss-of-subcooling are

1. In many events the operator is forced to assume the least probable event has occurred, that is the HPSI cannot make up the mass loss, and trip the RCPs.

2. Without the RCPs,

   a. The pressurizer spray is not available for RCS pressure control. Without pressurizer spray pressure control and a credible pressurizer level indication, the risks of offsite release during a steam generator tube rupture and pressurized thermal shock are increased.

   b. The operator must rely on pressurizer level alone for information on how to control the HPSI. Pressurizer level alone, however, may provide ambiguous information to the operator on system inventory, as at TMI, and lacks credibility as demonstrated at GINNA. This lack of credibility increases the probability of operator error.

TABLE 2

PROPOSED OPERATOR ACTIONS AND CRITERIA FOR RCS INVENTORY MANAGEMENT WHEN OFFSITE POWER IS AVAILABLE

Operator Actions	Proposed Criteria
Trip RCPs	1. Reactor tripped and 2. HPSI on and 3. Pump motor power or current indicates α = .15 and decreasing at pump inlet.
Throttle HPSI to maintain pressurizer level low in the pressurizer	1. Pump motor power or current indicates α = 0 at pump inlet and 2. Pressurizer level returns to the pressurizer or reverses and starts to increase.
Reinitiate RCPs one at a time while monitoring pumps inlet density to see if α = 0	1. Subcooling reestablished after break isolated or 2. Pressurizer level returns after break isolation.

4:3 Pump Motor Power Provides Better Information to the Operator

Pump motor power or current is proportional to coolant inventory in particular, the void fraction at the pump inlet. Therefore, pump motor power or current provides the operator the added information required to optimally manage the RCS inventory for the most probable events and decreases the probability of operator error.

The proposed operator actions and criteria using pump motor power or current, as displayed in Figure 5, are shown in Table 2. The table includes recommended actions for RCP trip and reinitiation and for HPSI operation. The criteria rely on pump power or current pressurizer liquid level, and the subcooling meter. The guidelines used to select the proposed RCP trip criteria were

1. The trip should be delayed sufficiently to allow the HPSI to make up the inventory loss for the more probable small breaks.

2. The trip should be based on a pump inlet void fraction sufficiently large to be outside the normal noise band around the $\alpha = 0$ line on the display.

3. The trip void fraction should be sufficiently small to occur before the bubbly-to-churn transition void fraction occurs in the pump.

4. The proposed criteria should not cause more RCS mass loss than the present criteria in case the RCPs are tripped.

The information in Figure 6 addresses the first guideline. The relationship between break diameter and saturation conditions are based on ZION HPSI flow rates and the HEM critical flow correlation. With full or degraded HPSI, the system can make up the break flow of 1 inch diameter single ended breaks and larger by the time the RCS reaches saturated conditions. Almost 80% of the RCS penetrations in a ZION type reactor are 1 inch or less in diameter.

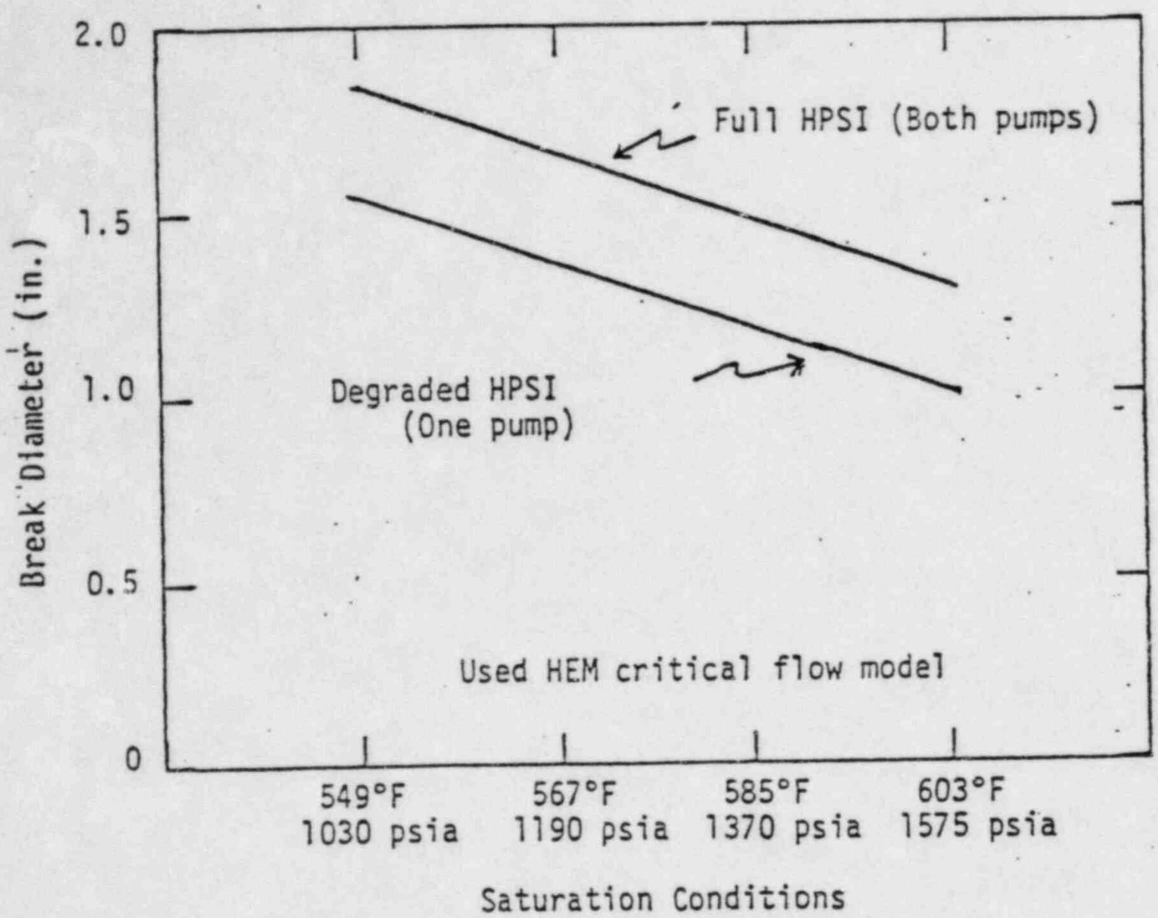

Figure 6. Break Diameter vs. Saturation Conditions at Which HPSI Flow = Break Flow in Zion

A double ended steam generator tube rupture in ZION is equivalent to a single ended break diameter of about 0.9 in.

The RCP trip void fraction recommended in Table 1 is .15. The .15 line in Figure 5 is well below the $\alpha = 0$ noise band of the data.

The CE/EPRI pump data[7] degrade at an inlet void fraction of about .18. When the pump starts degrading, the degradation is much more gradual than the LOFT pumps. Therefore, a void fraction criteria of .15 should be sufficiently small to occur before the bubbly-to-churn transition occurs in a commercial PWR reactor coolant pump.

The proposed criteria appears not to cause added system mass loss if the RCPs must be tripped. In fact, the proposed criteria may cause less mass loss than the present criteria. During LOFT Experiments L3-5 and L3-6, more mass was lost early in time when the pumps were off than when they were on, as seen in Figure 7. Once the break uncovered, when the pumps were off, the relative rate of mass loss between the two experiments was reversed and by 6 min into the transients, the net mass loss pumps on and pumps off was the same. A commercial PWR has about 10% more of its mass inventory above the RV nozzles than does LOFT. Therefore, the time of mass inventory equilibration should be later than 6 min for an equivalent sized break in a commercial PWR.

The optimum RCP trip time during LOFT Experiment L3-6 (pumps on) would have been 90 s. The present criteria would have tripped the pumps at 2 s and a loss of subcooling trip would have tripped the RCPs at 28 s. The proposed .15 inlet void criteria would have tripped the pumps at 60 s. No mass loss penalty would have been caused by the proposed criteria, in fact, less mass would have been lost. This analysis does not include all possible break locations. However, there is no obvious reason why consideration of other break locations would change the conclusion.

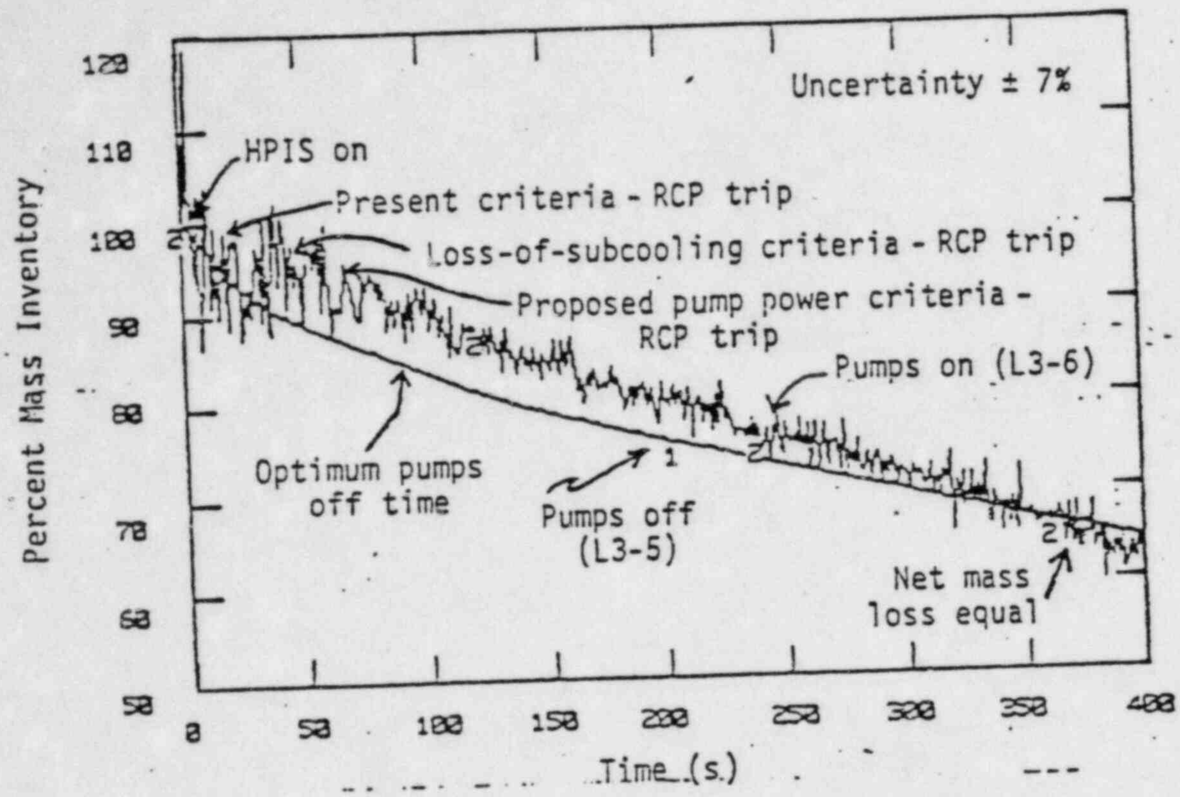

Figure 7. Mass inventory during pumps off (L3-5) and pumps on (L3-6) with annotated criteria RCP trip times

5.0 Proposed Additional Work

Work is presently continuing to compare the theory with data from other pumps. The data from an MIT pump, the Semiscale pump, the CE/EPRI pump and a limited amount of data from the LOBI pump appear consistent with the LOFT data, based on a preliminary review.

Other questions to be investigated are

1. If offsite power is not available, what information must the operator have, what measurements are needed, and what criteria should be used to manage RCS inventory?

2. How would the proposed display be used in a multiple loop context?

3. What plant specific issues would modify or complicate the use of pump motor current or power that have not been considered?

When this work is completed, the nuclear industry will be in a better position to resolve the RCS inventory management and reactor vessel liquid level measurement issues.

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7. J. W. Bolstad, Haarman, R. A., "Summary of Thermal-Hydraulic Calculations for a Pressurized Water Reactor," NUREG/CR-1480, May 1980.

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IN 82-39

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NO. 82-056

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D. C. 20555

September 21, 1982

IE INFORMATION NOTICE NO. 82-39: SERVICE DEGRADATION OF THICK WALL STAINLESS STEEL RECIRCULATION SYSTEM PIPING AT A BWR PLANT

Addressees:

All boiling water reactor facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This notice is to provide licensees and construction permit holders available information about the degradation of the primary pressure boundary at Nine Mile Point Unit 1 due to intergranular stress corrosion cracking. Recipients should review this information relative to their facilities. If NRC evaluation so indicates, further licensee action may be requested. In the interim, we expect licensees to review the relevance of this information for applicability to their facilities.

Description of Circumstances:

The Nine Mile Point Nuclear Station Unit 1 (NMP Unit 1) was shut down in order to replace recirculation pump seals. On March 23, 1982, leakage was visually detected at two of the ten recirculation loop safe ends during a primary system hydrotest at 900 psig to test the seals. Further visual inspection identified three pin-hole indications and a single ½-inch long axial indication, all of which were located in the heat affected zone of the welds where the safe end joined the pipe.

On March 26, 1982, an ultrasonic examination of the two affected safe ends and one other safe end confirmed the presence of intermittent cracking indications around the pipe's inside diameter. Further ultrasonic examination of the welds joining the pump discharge casting to the riser elbow also revealed cracking in weld heat affected zones on the inside diameter (ID) of the elbows. This was later confirmed by dye penetrant examination.

Because the cracks were confirmed at the welds of the safe ends and riser elbows, the ultrasonic examination was extended to all of the remaining welds in the five loops of the primary system, wherever radiation levels permitted. The results of this examination show ID cracking at a large number of the welds examined.

Two boat samples removed from the area of the through-wall cracks in one safe end were sent to General Electric and Battelle Laboratories, respectively, for evaluation. A boat sample from the crack region of the elbow weld was also evaluated by Sylvester Associates, consultants to the licensee. The results

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of these metallurgical evaluations concluded the degradation was due to intergranular stress corrosion cracking (IGSCC) in the sensitized region of the welds' heat affected zones. Further metallurgical investigation is being pursued to determine, as far as possible, the probable cause(s) of the problem.

Based on the results of the examinations and investigations to date, the licensee will replace the safe ends and 28-inch recirculation piping in all five loops of the system. Replacement of the branch piping out to the first isolation valve is also being considered; however, no final decision in this regard has been made at this time.

All replacement material will be stainless steel type 316 nuclear grade consistent with NUREG-0313, Revision 1 requirements. The actual replacement will be accomplished in accordance with ASME Boiler and Pressure Vessel Code, Section XI, 1977 Edition and Addenda through summer 1978. Welding will be performed in accordance with Section IX, 1978. Fitup requirements will be in accordance with ANSI Pressure Piping Code B31.1-1977 and Addenda through winter 1979. The replaced system configuration will duplicate the original design.

All ten recirculation system safe ends at NMP Unit 1 had been previously examined volumetrically by ultrasonic techniques at each refueling outage under an augmented inservice inspection program. This was in addition to the ASME code required inservice inspection program applied to other system welds. The augmented program was required because of IGSCC problems experienced with furnace-sensitized safe ends at this and other BWR plants.

It is important to note that the programs conducted under the normal and augmented programs did not indicate a pending problem. Examinations were performed during 1979 and 1981. The procedure employed during the 1981 augmented program for the safe ends was based on ultrasonic test (UT) using the EPRI transducer for a flat calibration block which was stated to be capable of detecting IGSCC at the code required gain or sensitivity level. The procedure differed from the GE recommended procedures in specifying less gain, and differed significantly in the calibration standards and data recording requirements, thus resulting in reduced sensitivity compared to the GE recommended procedures.

After leakage was visually observed on March 23, 1982, a UT examination of the safe ends was performed using the same method employed in the 1981 augmented program. Many safe ends exhibited code "reportable," but not rejectable indications. However, when an ultrasonic sensitivity above code calibration sensitivity was employed, greater reliability was realized in detecting the presence and full extent of the IGSCC problems with the thick wall piping welds, both at the safe ends and at other locations in the reactor coolant system. The generic implications of the above variances is under further review by the NRC staff.

This IE information notice is to advise licensees of further occurrences of the prevailing IGSCC problem that is under continuing review by the NRC staff.

If you have any questions regarding this matter, please contact the Regional Administrator of the appropriate Regional Office, or this Office.

E. Jordan

Edward L. Jordan, Director
Division of Engineering and
Quality Assurance
Office of Inspection and Enforcement

Technical Contact: W. J. Collins
301-492-7275

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List of Recently Issued IE Information Notices