NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 4, 1981

MEMORANDUM FOR: Dr. Dade W. Moeller, Chairman
Subcommittee on the TMI-1 Restart

FROM: Edward C. Abbott, ACRS Senior Fellow ECA

SUBJECT: REPORT BY THE STAFF OF THE HOUSE COMMITTEE ON INTERIOR AND INSULAR AFFAIRS ON THE REPORTING OF INFORMATION CONCERNING THE ACCIDENT AT THREE MILE ISLAND

I have attached both a copy of my review of the subject report and the subject report. You may find these items useful for your upcoming subcommittee meeting on the restart of Three Mile Island Unit 1.

I will be glad to answer any questions you may have about either item.

cc: ACRS Members
R. Major
T.G. McCreless
R.F. Fraley
M.W. Libarkin

1.0 Introduction

During the accident at Three Mile Island (TMI), state and federal officials received little meaningful information on the plant's status during initial phases of the transient. These officials, therefore, did not understand the seriousness of the accident and were unable to fulfill their responsibilities in the area of public health and safety. A recent report by the staff of the House Committee on Interior and Insular Affairs (hereinafter referred to as the Staff Report) reviews the reporting of information during the accident. Specifically, the Staff Report reviews key information and data available to the TMI management, the testimony of TMI's plant personnel and the adequacy of the reports made by Metropolitan Edison to state and local officials.

The purpose of my paper is to review the Staff Report's data, conclusions and other pertinent information. My summary of the Staff Report and the Staff Report's conclusion are presented in the next two sections. The fourth section analyzes some of the procedures and requirements in effect at the time of the accident. The final section draws conclusions based on the Staff Report and the information present in section four.

2.0 Summary

Based on my review of the Staff Report, additional information presented in section four and my previous experience as an operations supervisor, I agree with the conclusion presented in the Staff Report. The Staff's conclusion is contrary to other investigations conducted by the NRC[2,3]. In addition, the plant's operating procedures were followed but were inadequate while the site's emergency plan and procedures were adequate but not followed. The former led directly to a degraded core and the latter left state and public officials inadequately informed.

3.0 Conclusion of the Staff Report

The Staff Report concludes:

> "The record indicates that in reporting to State and Federal officials on March 28, 1979, TMI managers did not communicate information in their possession that they understood to be related to the severity of the situation. The lack of such information prevented State and Federal officials from accurately assessing the condition of the plant. In addition, the record indicates that TMI managers presented State and Federal officials misleading statements (i.e., statements that were inaccurate and incomplete) that conveyed the impression that the accident was substantially less severe and the situation more under control than what the managers themselves believed and what was in fact the case."

The data and testimony in the Interior Committee's Staff Report support the conclusion presented in the summary (Section 2.0) of this report. Additional information presented here reinforces that conclusion.

4.0 Procedures

The NRC requires that nuclear facilities be operated in accordance with written procedures. Plant management is required to have complete and detailed written procedures and operator licenses contain the statement "in manipulating the controls of the facility or facilities the licensee shall observe the operating procedures."[4]

Therefore, the plant management is bound by the plant's operating license to manage the plant according to written procedures and the individual licensed operator is bound by his license to operate the plant according to the operating procedures. Neither managing nor operating the plant according to the procedures constitutes a violation of the Commission's regulations. In part, these procedures include the plant's emergency operating procedures and the site emergency procedures. The basis for these procedures is the plant's operating license or more specifically an appendix to the license referred to as the Technical Specifications.

4.1 Emergency Operating Procedures

The Emergency Operating Procedures deal with the design basis accidents and transients. During the TMI accident the operators successfully used the procedures for loss of feedwater, reactor scram and turbine trip. The shift supervisor and the two control room operators verified the automatic scram, the transfer of electrical loads to offsite power and restored feedwater to the steam generators by opening the auxiliary feedwater pump block valves. It took only minutes to perform these actions and they were correct for the plant condition at the time of the transient.

In addition the operators followed another emergency operating procedure but it provided misleading symptoms for the existing plant conditions and contained incorrect operator actions. This procedure is titled "Loss of Coolant/Coolant Pressure" (2202-1.5). In a deposition for the President's Commission Bill Zewe,[5] Shift Supervisor during the accident, states the actions required by this procedure and his reasons for following it. A portion of that deposition is contained in Appendix A, Part A.

Three sections of the "Loss of Coolant/Coolant Pressure" procedure are important to the TMI-2 accident. (See Appendix B) The section entitled "Symptoms" (A.1.1), tells the operator to expect pressurizer level and coolant pressure to decrease and become stable during a leak or rupture within the capability of the make-up system. As noted elsewhere, this is not true for a small break on top of the pressurizer which leads to saturated conditions in the reactor coolant system.[5] The section entitled "Manual Action" (A.2.2.2), tells the operator to start a make-up pump* if required. This was done within the first minute of the reactor scram. The section entitled "Follow-up Action" (A.3.2.2), tells the operator to bypass the safety injection signal and throttle HPI flow. This was done within the first five minutes of the reactor scram. The procedure for loss of coolant and loss of coolant pressure was therefore followed but the procedure itself was wrong.

In the same deposition referenced earlier, Zewe shows his frustrations with the difference between actual and expected plant symptoms and his fears about having a water solid reactor coolant system. (See Appendix A, Part B) He states that during the accident pressurizer level would decrease then go off scale high. He could not find a logical explanation for that

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* A make-up pump is the same as a high pressure injection (HPI) pump.

behavior. Also, he feared losing pressure control and ruining the code safety valves as a result of being solid.

The operators were therefore placed in a position where compliance with the plant's emergency procedures caused the core to degrade. In addition the operators' fear of taking the plant water solid probably contributed to their continued attempts to decrease pressurizer level at the expense of core cooling.

4.2 Technical Specifications

Just as the operator must comply with plant's operating procedures, he must also comply with the plant's Technical Specifications. As stated earlier, the Technical Specifications are part of a plant's operating license. If the plant is operated in compliance with its Technical Specifications, the plant should not exceed its design limits and thus should not endanger public health and safety. In addition, part of the senior license examination deals with these specifications and administrative requirements to assure compliance.

As a limiting condition for operation in the hot standby condition (called mode 3), the TMI-2 pressurizer is required to be operable. Operable is defined as the presence of a steam bubble and a water volume which is between 240 and 1,330 cubic feet. This water volume corresponds to a measured water level of 45 to 385 inches. This requirement is repeated in Operating Procedure 2103-1.3, Pressurizer Operations.[7] The plant technical specifications and operating procedures, therefore, require the operators to avoid a water level that exceeds 385 inches with primary system temperature above 280°F. There is no exception to this requirement which allows solid conditions during an anticipated operational occurrence or a design basis accident.

Jim Floyd, the Operations Supervisor for Unit 2, expressed the nature of his subordinates' actions succintly in a deposition for the President's Commission.[8] In part, he said,

> "There is, in the B&W Limits and Precautions, or there was before March 28, the statement that the plant will never be solid except for hydrostatic testing.
>
> "The emergency procedures says that at 385 inches in the pressurizer you will trip the reactor which is about as drastic an action as we teach operators to take. So that he would normally be instructed when he got up 385 and was still driving, to trip the reactor and not let it go solid, and the only way you can do that is to knock off the high pressure injection and increase the letdown, which is exactly what they did on the morning of the 28th. So I think he responded exactly the way he was trained."

Therefore, the operators' actions were consistent with the Technical Specifications and the plant's procedures yet such compliance led directly to a degraded core and subsequent implementation of the site emergency procedure.

4.3 Site Emergency Procedures

NRC requires each licensee to submit an Emergency Plan for review and to develop Emergency Procedures that implement the plan. The applicable procedures in effect at the time of the accident were the Radiation Emergency Procedures (1670.2 and 1670.3).[9] The plan and these procedures require, among other things, that the Pennsylvania Bureau of Radiological Health (BRP) be notified of the status of the "consequence mitigation features and the possible need for protective action."

The Staff Report cites several examples where the plant's status as indicated by power operated relief valve's position, core exit thermocouple readings, hot leg temperatures, etc. were known to individuals (both management and non-supervisory). This known data coupled with that known to the plant's emergency command team (established with the expressed purpose of analyzing these abnormal conditions) make it difficult to understand the reason for the lack of information sent to BRP during the accident. For example, the command team's understanding of the accident may have been enhanced by a similar occurrence during "hot functional testing" prior to the initial fuel load (see Appendix C). In a telephone conversation between two Babcock & Wilcox employees, Don Roy at Lynchburg and Leland Rogers, at the plant site, reference is made to steam binding in the loops during this test phase.

Although no other data or record on this incident can be found, several possible sequences of events could result in steam binding in the loops without the decay heat from an exposed core. One such method is to heat-up the primary system to rated temperature and pressure by running the reactor coolant pumps. If the pressurizer temperature is allowed to fall below the primary coolant temperature during this heat-up, the primary system will go to saturation and steam voids will form in high curved sections of the hot legs (commonly referred to as the "candy canes"). The voids will remain until the latent heat of vaporization is removed. Increasing system pressure in the absence of a reliable heat sink by increasing HPI flow merely decreases the void volume. The steam void acts much like an ideal gas on increasing pressure at constant temperature. In fact, pressurizer pressure increases are modeled like an ideal gas during primary system insurges. The plant conditions described by Rogers during hot functional testing were similar to those experienced during the accident.

In addition, the extended duration of the pre-accident problem may have delayed the test program and probably would have been a matter of management review. It is difficult to believe it would have gone unnoticed or have been ignored by the plant's management and operators.

There exists a clear and obvious information gap between what the personnel at the plant knew during the initial phase of the accident and what the state knew. This is indicated by the record of the testimony, gathered by the Interior's staff. A portion of a written report by Leland Rogers of Babcock & Wilcox illustrates the nature and extent of this gap.[9] Upon his arrival in the control room at about 7:00 a.m. the day of the accident, a caucus among members of the command team took place. Rogers described the caucus as follows:

> "Garry Miller, Bill Zewe, Mike Ross, George Kunder and I held a short caucus about immediate action to attempt with the plant. An agreement was reached to attempt to run a reactor coolant pump in the "A" loop, since the last pump run was in the "B" loop, to see if the same effects

would be apparent in the "A" loop. System pressure was approximately 1500 psi. The 1A pump was started and showed no flow indication an' minimum current on the motor. I was then convinced that by some method the plant had converted both Th legs volumes to steam and it was as large a void as to extend essentially to the pump suctions. I suggested that further pump runs at that time would do nothing for the system and possibly destroy a shaft seal assembly. Miller asked all of the supervisory and assistance people to retire to the Shift Supervisors Office in the rear of the Control Room." (underline added)

The information given to the state conveys an entirely different impression of the seriousness of the accident when compared to Roger's initial assessment. Bill Dornsife, an engineer with BRP, received information on the phone from Gary Miller, the station manager, in accordance with the plant's emergency procedures. During their discussion, Dornsife took notes. Based on these notes and Dornsife's memory, Dornsife wrote a more detailed summary.[3] According to these "recollections", (see Appendix D) Miller told Dornsife the plant had experienced an unexpected turbine trip, reactor scram, a corrected mechanical malfunction (i.e., a stuck open PORV with a closed block valve) and possible flashing or bubbles in the primary system ending with a stable, normal cooling mode on one steam generator. This impression of stability was reiterated by Metropolitan Edison personnel in a meeting with Thomas Gerusky of BRP and Lieutenant Governor Seanton in the Lieutenant Governor's Office at 2:00 p.m. Gerusky describes the impression as one of "the accident is over, and now all it was, was clean-up." (Reference 1, page 112). In addition, there is nothing in the Staff report that indicates the situation had improved between the assessment made by Rogers at about 7:00 a.m. and the 2:00 p.m. meeting in the Lieutenant Governor's Office. During this time, hot leg temperatures remained high, exit core thermocouple remained high, the primary system remained bound with steam and the command team remained concerned about whether or not the core was covered.

Dornsife's recollections could be matched with other statements made by operators, technicians and supervisors regarding their assessment of the plant's status during the first 24 hours of the accident. The Staff Report cites many of these statements as examples of the knowledge level of the plant's personnel. There are others which could be cited here especially those regarding the adequacy of core cooling and primary system water inventory. (Sections III.H and I of the reference 1).

Based on my review and on the Interior staff's review, neither the Emergency Plan nor the Emergency Procedures were followed. Proper notifications were made in accordance with these plans and procedures but very little meaningful information was given to those agencies notified, especially the BRP. Furthermore it is reasonable, again based on record, that the management of the plant knew much more than they were telling people in their notifications during the accident.

5.0 Conclusion

My review of the Staff Report and additional information relating to the accident does not support the popular conclusion that either the operators were inadequately trained or that TMI's management was confused and disorganized. In fact, I reached a contrary judgment.

The operators were adequately trained on the existing operating procedures and adhered to those procedures in attempting to maintain one limiting condition for operation within licensed limits (i.e., pressurizer level). They did exactly what was expected of them. Therefore, to characterize the accident as "operator error" is misleading. The responsibility of management is to supply the operator with adequate procedures and proper training in those procedures. The responsibility of the operator is to understand and follow those procedures in accordance with his training and operating license.

In implementing the Radiation Emergency Procedures, management was organized in such a manner to analyze the plant's data, formulate a plan to assure an eventual plant cooldown and manipulate the plant to accomplish that end. The individual members of the emergency management command team were experienced engineers and at least one was familiar with a similar plant condition during hot functional testing. The command team, however, failed to comply with reporting requirements of those procedures.

Lessons learned from this review can be summarized as follows:

Operators should not be forced to violate license limits in order to maintain core cooling.

2) Operating procedures should be correct.

3) During emergencies, plant management must give precise and comprehensive plant status information to public officials responsible for protecting health and safety.

The significance of these lessons is that much of the NRC's Action Plan is incorrect being based on the false conclusion that the seriousness of the accident was the result of operator error. In fact, it was the result of faulty procedures.

References

1. Reporting of Information Concerning the Accident at Three Mile Island, The Staff of the House Committee on Interior and Insular Affairs, January, 1981.

2. U.S. Nuclear Regulatory Commission, NUREG-0600, "Investigations Into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement", Investigative Report No. 50-320/79-10, August 1979.

3. U.S. Nuclear Regulatory Commission, "Three Mile Island, A Report to the Commissioners and to the Public", Special Inquiry Group (Rogovin Report), January 24, 1980.

4. U.S. Nuclear Regulatory Commission, Form NR78 (9-76), 10 CFR 55, Senior Operator and Operator License.

5. Memo from C. Michelson to H. Etherington, "Adequacy of TMI-2 Emergency Procedures for the Case of a Loss of Reactor Coolant at the Top of the Pressurizer," November 6, 1979.

6. U.S. Nuclear Regulatory Commission, NUREG-0432, "Operating Technical Specification for Three Mile Island Unit 2", February 8, 1978.

7. Metropolitan Edison, "TMI Unit 2 Operating Procedure 2103-1.3," Pressurizer Opeartions Revision 3, July 19, 1978.

8. James R. Floyd deposition for the President's Commission on the Accident at Three Mile Island, August 1, 1979.

9. Radiation Emergency Procedure 1670.2 amd 1670.3, Site Emergency Plan Procedure and General Emergency Plan Procedure.

10. Statement of the 3/28/79 Unit 2 Transient, Leland C. Rogers, The Babcock & Wilcox Company, June 12, 1979.

Appendix A

Selected Portions of a Deposition by William H. Zewe
for the President's Commission on the
Accident at Three Mile Island, July 26, 1979

Part A

Q Can you show me the procedure?

A Sure, right here (Indicating.)
All right, where you have safety injection manually initiated, and then you just follow the actions here to where you verify the pumps start and you shut the suction from the makeup tank, and then you shut the normal pressurizer level isolation valve and you bypass safety injection and then throttle the high-pressure injection valves to maintain pressurizer level.

Q You were referring to the procedures beginning on Page 2 with 3.0 to 3.2.2, correct?

A. That is the followup actions for a leak or rupture within the capability of the system operation.

Q Were these procedures you had on March 28?

A I did refer to these procedures, yes.

Q You pulled this procedure and referred to it on March 28?

A I did, yes, for the indications that we had, I picked this portion of the procedure to use, all right, but not knowing that I had a small break, but knowing that I had high pressure injection manually on before we had the automatic actuation.
I felt that this was the place in the procedure that was applicable, not knowing that we still had or that we had at that point a loss of coolant from the system.

Appendix A
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Part B

Q What was your best guess at the time of where the water was coming from?

A It had to come from the makeup system, because that is the only pressure that is higher than the reactor coolant system, in order to put in water.

Q You thought water was coming from the makeup system?

A And I really didn't know if the water was still going in. I felt we had it in and that we had secured putting in the water, but for some reason we could not get the water back out because we had throttled the high-pressure injection down sufficiently to where we should have seen a reduction in level. That reduction in flow, coupled with the increase in letdown flow, we should have been able to bring down pressurizer level, and we did have indications of the pressurizer level trying to come on-scale, and then it would go off again and would come down again. It was a real problem, in that I really couldn't determine why [illegible]ting that way. I really couldn't think of any logical explanation on why.

Q Were your fears that the plant would be damaged if you went solid?

A Yes. That is always a possibility that if you go solid, you will have the uncontrolled pressure surge. At the time, I didn't want to go solid, but I really didn't say in my own mind that they would break it all in the same train of thought, I just know I didn't want it to approach going solid. I really didn't say, "Well, in a few minutes, I will be solid and I could rupture some of the primary pipe." It is true that if you go solid, you worry about an overpressure condition; you also worry about an underpressure condition, too; and the uncontrolled aspect of it.

Q In what specific ways did you think that you could damage the plant if you did go solid?

A I don't remember if I really set down and thought about what particularly would I ruin. I remember thinking, if we do go solid, we will probably ruin the code safety valves, because they are made to release steam; they aren't made to discharge water.

THREE MILE ISLAND NUCLEAR STATION

UNIT #2 EMERGENCY PROCEDURE 2202-1.3

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM

PRESSURE

A. Leak or Rupture Within Capability of System Operation.

1.0 SYMPTOMS

1.1 Initial loss of reactor coolant pressure & decrease in pressurizer level becoming stable after short period of time.

1.2 Possible reactor building high radiation and/or temp. alarm.

1.3 Possible reactor building sump high level alarm.

1.4 Make-up tank level decreasing > 1" in 3 min.

1.5 Possible makeup line high flow alarm. ?

1.6 RB Fan Drip Pan Level Hi Alarms.

NOTE: The operator may distinguish between a loss of coolant inside containment, an OTSG tube rupture and a steam line break by the following symptoms which are unique to the aforementioned accidents.

1. Loss of coolant inside Rx Bldg. - particulate, iodine & gas monitor alarm on HP-R-227 "Reactor Building Air Sample".

2. OTSG tube rupture - gas monitor alarm on VA-R-748.

3. Steam line break

(1) Low condensate storage tank level alarm - and or low hot well level alarm.

(2) FW Latch System Actuation.

2.0 IMMEDIATE ACTION

2.1 Automatic Action:

2.1.1 MU-V17 will open to compensate for reduced pressurizer level.

2.1.2 Additional pressurizer heaters will come on in response to reduced reactor coolant pressure.

2.2 Manual Action

2.2.1 Verify MU-V17 open and pressurizer heaters on.

2.2.2 "CLOSE" MU-V376 letdown isolation valve, & "START" the backup MU pump, if required.

2.2.3 Reduce load at 10% minute & proceed with normal shutdown.

2.2.4 "LINE-UP" waste transfer pump from a R.C. Bleed Holdup Tank & pump to the makeup tank to maintain required level.

2.2.5 If for any reason the operator cannot maintain Make-up Tank and Pressurizer levels above their respective low level alarm setpoints, "TRIP" the reactor, "INITIATE" Safety Injection manually (push buttons on panel 3), & then "Close" MU-V12.

3.0 FOLLOW UP ACTION

3.1 Safety Injection Not Initiated.

3.1.1 Initiate unit shutdown & cooldown per 2102-3.1 and 2102-3.2 respectively.

3.2 Safety Injection Manually Initiated (HPI and LPI).

3.2.1 Verify that the Makeup Pumps & Decay Heat Removal Pumps start satisfactorily.

3.2.1.1 Close MU-V12 and MU-V18.

3.2.2 Bypass the SAFETY INJECTION by DEPRESSING the Group Reset Pushbuttons & "THROTTLE" MU-V16A/B/C/D as necessary to maintain 220" pressurizer level and not exceed 250 GPM/HPI flow leg.

3.2.3 If MU pump flow drops below 95 GPM, trip excess MU pumps.

APPENDIX C

CONFERENCE CALL BETWEEN DR. DONALD ROY AND MR. LELAND ROGERS ON MARCH 28, 1979 AT ABOUT 7:30 P.M. DR ROY IS AT THE B&W OFFICES IN LYNCHBURG AND MR. ROGERS IS IN THE TMI-2 CONTROL ROOM.

Roy: Won't eventually that steam bubble from the B loop come over into the A loop and get into the pump and aren't you gonna have a lack of water in the complete system eventually?

Rogers: No, the pressurizer is full of water.

Roy: But will that be enough to fill the B loop when you start the pump?

Rogers: There is certainly a chance not knowing how much of a steam bubble condition we do have there that we will come up with a saturated condition, yes.

Roy: Aren't you safer to fill the whole system solid?

Rogers: Cannot fill the whole system! There is no way to fill it! The system is designed to move water in a loca. We haven't had a loca and we can't operate it that way. What we got is . . . . We have been in hot functional test. We had a similar condition where we had the legs on both loops filled with hotter temperature water than what we had in the system and it took us something like four days to get out of that thing to try and cool it down to where we could get that bubble condition out of there. We've got a similar condition here. The only way we can do it is to force that bubble out of that B loop at this point.

Roy: Then you've got no choice.

Appendix D

"Recollections" prepared by W. Dornsife in April 1979
Times are approximate

0705 - Received call from C. Deller, PEMA duty officer indicating that TMI has a site emergency and to call plant to get details

0706 - Called Maggie to inform her and verify number to call at plant site - only number we had was thru plant switchboard 944-4041

0707 - Called plant site - had difficulty getting through switchboard to Unit 2 control room - finally gave switchboard my home number to have control room directly call me

0710 - Shift supervisor called back to my home number. He told me the plant had suffered a transient and RB [reactor building] radiation level was high initiating the site emergency - things sounded very confused at this point in time - I tried to get a status of important safeguards without very much success - they did tell me that reactor was shutdown and RB pressure was about 1 or 2 psi - SI [safety injection] had been initiated and was cooling core - They informed me that they had sent out monitoring teams and there was no detectable radiation levels outside the plant. I then heard in background the announcement to evacuate the Unit 2 fuel handling and auxiliary buildings. At this point a health physics type got on the phone and things sounded extremely confused and finally he hung up saying he would call back.

0720 - Called office - talked to Diane - told her briefly about what had happened and I was on my way in to the office - told her first technical type who arrives in office should call Unit 2 control room immediately. I arrived in office about 0750. Tom was there with open line established to plant control unit. Plant had declared a general emergency about 0730 due to high radiation levels in the reactor building. There still were no releases outside plant. Met Ed monitoring teams were out and around.

I talked to plant to get a later status. As I recall, they said they were being core cool by feeding with makeup pumps and bleeding out through pressurized electromechanical relief valve. From the information that I was getting it sounded as if plant conditions were stabilized (In reality the core was probably being uncovered at this time and fuel damage was continuing).

For the next hour or so we kept getting plant status reports periodically. (The open line was not manned continuously by Met Ed. They would come to the phone when ready to report). Things seemed to remain the same with still no release occurring. At about 0900 I was asked by Middendorf or Duncan to go brief the Lt. Governor and attend a press briefing that was scheduled for about 1000. I called back to plant to get more details on what had initiated accident and what the present status was in order to brief Lt. Governor and Governor. At this point, Gary Miller, plant superintendent, came on the line and briefed me on what had occurred. His briefing was as follows (based partly on notes and partly on recollection):

> At 4:00 a.m. a turbine trip from 98% power occurred - reactor shutdown automatically - violation of tech specs in that aux feed was valued out temporarily S/G may have boiled dry - electromechanical relief valve lifted but did not reseat - indication in control room (elec signal to valve) indicated that it had reseated - block valve upstream is now closed - High pressure safety injection was initiated - all safeguard system operated as designed - pressurizer may have gone solid and low pressure in primary probably caused flashing and bubbles in primary - may have temporarily lost main coolant circulation - even currently stabilized and cooling normally on A S/G - possible primary to secondary leak in B S/G - B S/G has been isolated - 100 ppm Boron in primary may have been diluted

by secondary to primary feedback thru tube leakage - there has probably been a slight amount of failed fuel no speculation as to amount RB dome monitor reading 600 R/hr - RB pressure or approximately 1 psig - fence post dose $<$ 1 mr/hr - wind blowing to west currently sending monitoring team to Goldsboro