YANKEE ATOMIC ELECTRIC COMPANY

Telephone (617) 872-8100
TWX 710-380-7619

1671 Worcester Road, Framingham, Massachusetts 01701

2.C.2.1
FYR 85-86

August 15, 1985

United States Nuclear Regulatory Commission
Wasington, D.C. 20555

Attention: Mr. John A. Zwolinski, Chief
Operating Reactors Branch 5
Division of Licensing

References: (a) License No. DPR-3 (Docket No. 50-29)
(b) YAEC Letter to USNRC, dated August 1, 1984 (FYR 84-83)
(c) USNRC Letter to YAEC, dated January 31, 1985
(d) USNRC Letter to YAEC, dated March 26, 1985

Subject: Reactor Coolant Pump Trip - Resolution of Outstanding Items

Dear Sir:

By letter dated August 1, 1984 [Reference(b)], YAEC provided an analysis of "Reactor Coolant Pump Operations During Small Break LOCA Transients" for the Yankee plant. The NRC reviewed the report and determined that additional information was needed to complete the review [Reference (c)]. At a meeting on March 13, 1985, the NRC staff and YAEC discussed our responses to these questions and concluded the following:

1. YAEC will provide written response to Questions 1 through 17.

2. YAEC will also respond to Questions 32, 40, and 44.

3. The remaining questions were resolved at the meeting and require no further action.

Enclosed please find the answers to Questions 1 through 17 and to Questions 32, 40, and 44. With the answers to these questions the RCP trip issue is considered to be resolved for the Yankee plant.

Very truly yours,

YANKEE ATOMIC ELECTRIC COMPANY

George Papanic Jr

G. Papanic
Senior Project Engineer- Licensing

GP/jo

8508200156 850815
PDR ADOCK 05000029
P PDR

A001
1/1

The question numbers match those of the enclosure in Reference 1.

Q.1

Assure that RCP trip will not occur for Steam Generator Tube Ruptures (SGTRs) up to and including the design basis SGTR.

A.1

The design basis SGTR accident with a double-ended guillotine break of one U-tube has been analyzed. The analysis was initiated at 102% power. At the Yankee plant, two pumps are powered on on-site power and two on off-site power. The on-site power pumps were tripped 90 seconds after reactor trip to account for turbine coastdown. The other two pumps were left running throughout the transient. The analysis shows that the subcooling drops to about 18°F at about 70 seconds and then increases slowly for the remainder of the transient (Figure 1).

Westinghouse analyses of credible non-LOCA events for the assessment of alternate RCP trip criteria (Reference 2) show that, except for the SGTR, the Reactor Coolant System subcooling margin will increase during the non-LOCA transients of concern, as shown in Figure 2 from Reference 2.

At the Yankee plant, two pumps are tripped 90 seconds after reactor scram for any event. The operators will be advised to keep the remaining two pumps running if the primary system coolant subcooling margin remains above 35°F. This will keep the pumps running for non-LOCA transients of concern. If the subcooling margin drops below 35°F, the operators will be advised to trip the remaining two pumps if the containment overpressure of 5 psig is reached. For most of the SBLOCAs analyzed in Reference 3, the containment isolation signal of 5 psig is reached in less than 2 minutes. For SGTRs, the containment will remain at nominal conditions; therefore, the operators will nct trip the remaining two RCPs.

Q.2

Assure that symptoms and signals differentiate between LOCAs and other transients.

A.2

The 35°F primary coolant subcooling margin will differentiate between non-LOCA transients and SGTRs and LOCAs.

To differentiate between SGTRs and LOCAs, the containment overpressure of 5 psig is used.

Q.3

Assure that adequate recovery procedures are developed and in place to handle upper vessel head steam bubble conditions following delayed RCP trip.

A.3

Adequate recovery procedures are developed and in place to optimize the handling of an upper vessel head steam bubble under any conditions. Operators are instructed to pump for overpressure based on thermocouples located above the core. Special cases are defined to handle PTS and SGTR events.

An upper-vessel head steam bubble alone is not a danger to the core. It does indicate a deteriorating situation and is an operational inconvenience, however. With this perspective, the recovery procedures first address prevention, then address the removal of a bubble as part of the handling of the larger problems.

Q.4

Assure that RCP operation is excluded in a voided system where pump head is more than 10% degraded unless analyses or tests can justify pump and seal integrity when operating in a voided system.

A.4

The adverse containment uncertainty on the RCS subcooling monitor is $30^{\circ}F$. Hence, tripping at $35^{\circ}F$ subcooling assures that the pump operation is excluded in a voided system. Additionally, the 5 psig containment overpressure pump tripping logic assures that the pumps are tripped for LOCAs.

Q.5

Assure that unnecessary challenges to the PORVs are not introduced by the RCP trip scheme.

A.5

The answers to Q.1 and Q.2 assure that the RCPs will remain in operation for credible non-LOCA events. Therefore, the operators can retain normal pressurizer pressure control and the PORVs will not unnecessarily be challenged.

Q.6

Provide guidance for detecting, managing and removing voids following delayed RCP trip with a partially voided system.

A.6

Adequate recovery procedures are developed and in place to optimize the handling of voids under any conditions. Operators are instructed to pump for overpressure based on thermocouples located above the core and

to establish subcooling using the secondary heat sinks. Special cases are defined to handle PTS, SGTR and loss of secondary heat sink cases.

Voids alone are not a danger to the core. However, they do indicate a deteriorating situation and present an operational inconvenience. With this perspective, the recovery procedures first address prevention, then address the removal of voids as part of the handling of the larger problems.

Q.7

Assure that containment isolation will not cause problems if it occurs for non-LOCA transients and accidents.

A.7

It is possible to postulate non-LOCA accident conditions which would lead to containment isolation. However, the objective of the proposed RCP trip strategy is to establish reasonable assurance for continued pump operation during non-LOCA events.

The subcooling of $35^{\circ}F$ and the 5 psig containment overpressure will allow continuous pump operation for most of the steam line breaks outside the containment, including a stuck open steam generator safety relief valve.

A small steam line break inside the containment was postulated. This represents a break in one of the pressure taps in the steam generator dome. The containment pressure calculation for the complete severance of the tap was analyzed with CONTEMPT LT-26. It was found that during this event, the containment overpressure of 5 psig will be reached in 10 minutes (Figure 3). Coupled with the subcooling monitor indication, the operators will know that it is a non-LOCA overcooling transient with containment spill and will proceed accordingly. See response to Question 10 below.

Q.8

Address the consequences of pump and/or pump seal failure in the analyses if their integrity cannot be assured.

A.8

The integrity of the Main Coolant Pumps (MCPs) at Yankee Nuclear Power Station is assured by the canned-rotor design.

Q.9

Evaluate the capability to continue RCP operation without essential water services.

A.9

MCP operation depends on an operable Component Cooling System for continuous operation. Without component cooling, MCP operation is limited by bearing temperatures to approximately 3 minutes of operation.

Q.10

Evaluate the capability to rapidly restore essential water services.

A.10

Adequate recovery procedures are developed and in place to rapidly restore essential water (component cooling) services. Procedures address partial or total loss of component cooling by first isolating nonvital equipment using essential water, possibly isolating the break in the process, and by cross-connecting to a virtually endless supply of cooling water via the fire mains.

Q.11

Assure that the pump trip devices are reliable and that pump trip will occur when required.

A.11

The MCP trip devices at Yankee are highly reliable, assuring that pump trip will occur when required. These devices are tested on a refueling basis and have proven their reliability through 25 years of operating experience. Manual trip circuitry is exercised frequently during shutdown and startup maneuvers.

The automatic trip relays are backed up by two manual trip mechanisms and operators are trained in recognizing the precursers and conditions requiring trip.

Q.12

Later in the report, YAEC concludes that Phase I was successful in complying with 10CFR50.46. Does this imply that operators are allowed to trip the RCPs at any time during an SBLOCA?

A.12

The operators will be advised to trip the pumps if the subcooling drops below 35°F and the containment overpressure exceeds 5 psig.

Q.13

Does the successful compliance of Phase I mean that operators have the option of tripping the RCPs during a SGTR or other non-LOCA upset event?

A.13

The operators will only trip the pumps if the subcooling drops below 35°F and the 5 psig containment overpressure is exceeded. These criteria will allow the pumps to run for SGTR and other non-LOCA upset events and be tripped for SBLOCAs.

Q.14

If RCPs are to remain on during SGTR events, how will operators distinguish between SBLOCA and SGTR events?

A.14

The containment overpressurization signal of 5 psig will allow the operators to distinguish between SBLOCA and SGTR events.

Q.15

How does Phase I address recovery of a partially voided system?

A.15

Phase I of Reference 3 does not address the recovery of a partially voided system. However, the answers to Q.3 and Q.10 address this issue.

Q.16

Are any recommended timing procedures for RCP restart analyzed as part of Phase II?

A.16

No, there are no recommended procedures for RCP restart analyzed in Reference 3.

Q.17

To identify the worst break size, the RCPs are tripped at the time of minimum inventory. Can the peak clad temperature be higher if the RCPs are tripped before minimum inventory is reached because then a longer core uncovery time occurs before the ECCS refills the core? If so, then justify why the worst break size can be based on an analysis assuming all RCP trips occur at minimum inventory time.

A.17

The minimum inventory trip setpoint was used for the break spectrum analysis only. This provided a consistent comparison between the breaks analyzed.

For the worst break size and location (1" break in the cold leg discharge) determined in the EM break spectrum, a pump trip time sensitivity was performed. The trip time for the two pumps powered by off-site power was varied. Table 1 presents the pump trip time and the PCT for the break analyzed.

TABLE 17-1

1" Cold Leg Discharge Break (EM)

Pump Trip Time (seconds)	Peak Clad Temperature (^{o}F)
92	850
211	870
693	800
1,000 (minimum inventory)	800

Clad heatup is nearly the same for early pump trip as for the minimum inventory pump trip. We expect similar results for the other breaks analyzed with the pumps tripped at minimum inventory only.

Q.32

In Figure 3.1.4-4, what is the cause of the large core void spike at 1800 s? The concern is that if this phenomenon is real, how accurately is the resulting temperature spike being predicted in Figure 3.4.1-7?

A.32

Case 1BS was used to study the cause of the phenomena observed in these three cases. The sudden drop in core level is seen to occur shortly after 1,800 seconds, as seen in the Case 1BS results (Figure 4). In the vicinity of 1,800 seconds, vessel inventory reaches its minimum, as

shown in Figure 5. However, the drop in core vessel inventory is fairly gradual and constant in the time prior to minimum inventory. The vessel inventory depletion over the time interval from 1,500 to 1,800 seconds corresponds to a time period of high break flow (Figure 6). The high break flow is due to low temperature in the cold leg break volume, which is receiving cold ECCS water (135^{o}F) under fairly stagnant conditions. A manometer effect causes the core level to drop shortly before 1,800 seconds, as shown in Figure 7. Due to the manometer effect, hot water from the plenum moves to the break, causing break flow to be substantially reduced, around 1,830 seconds. The reduction in flow breaks the siphoning of liquid from the vessel. By 1,850 seconds, the vessel inventory gains mass from the intact loops, and core level is recovered (Figure 8). Hot water from the reactor downcomer and the mixing of ECC injection water with the hotter water in the loops limit the manometer effect in the reactor vessel.

In summary, the sudden core level drop in the three cases is caused by a short-term manometer effect which amplifies the effect of increased break flow.

Q.40

In Figure 4.1.3-6, what caused the sudden decrease in collapsed liquid level at 2060 to 2250 s?

A.40

The explanation provided for Question 32 concerning the sudden decrease in the collapsed liquid level in Figure 3.1.4-6 of Reference 3, is also applicable to Question 40.

Q.44

In Figure 5.1.1-6, what phenomenon causes the sudden drop in the collapsed liquid level from 1300 s to 1670 s?

A.44

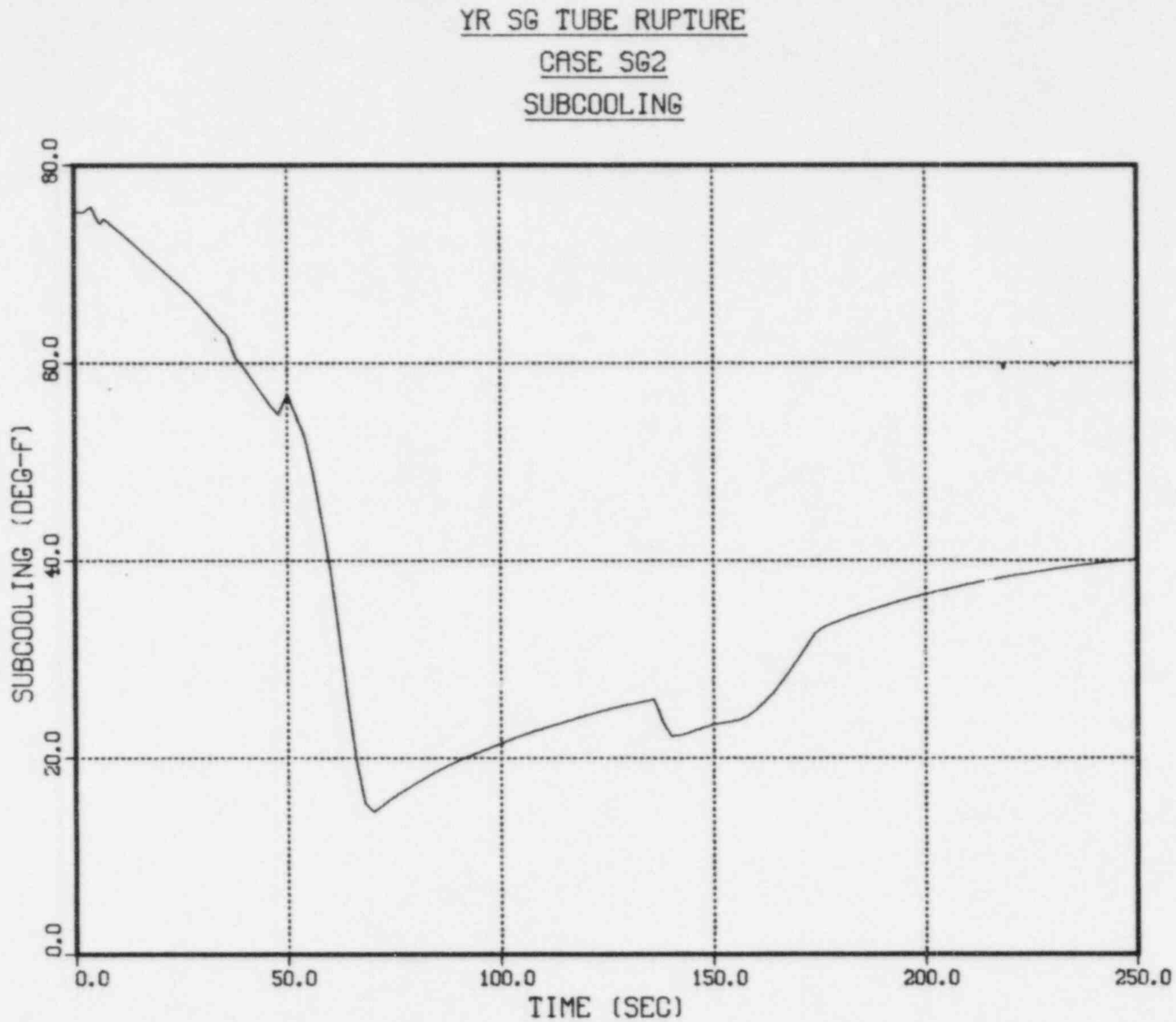
The explanation provided for Question 32 concerning the sudden decrease in the collapsed liquid level in Figure 3.1.4-6 of Reference 3, is also applicable to Question 44.

REFERENCES

1. Zwolinski, John A., USNRC to J. A. Kay concerning, "Reactor Coolant Pump Trip Request for Additional Information - Yankee Nuclear Power Station," NYR 85-19, Docket No. 50-29, February 5, 1985.

2. Kemper, R. M., et al., "Evaluation of Alternate RCP Criteria," Westinghouse Report Prepared for the Westinghouse Owners Group, 4497Q:1D/081283, August 1983.

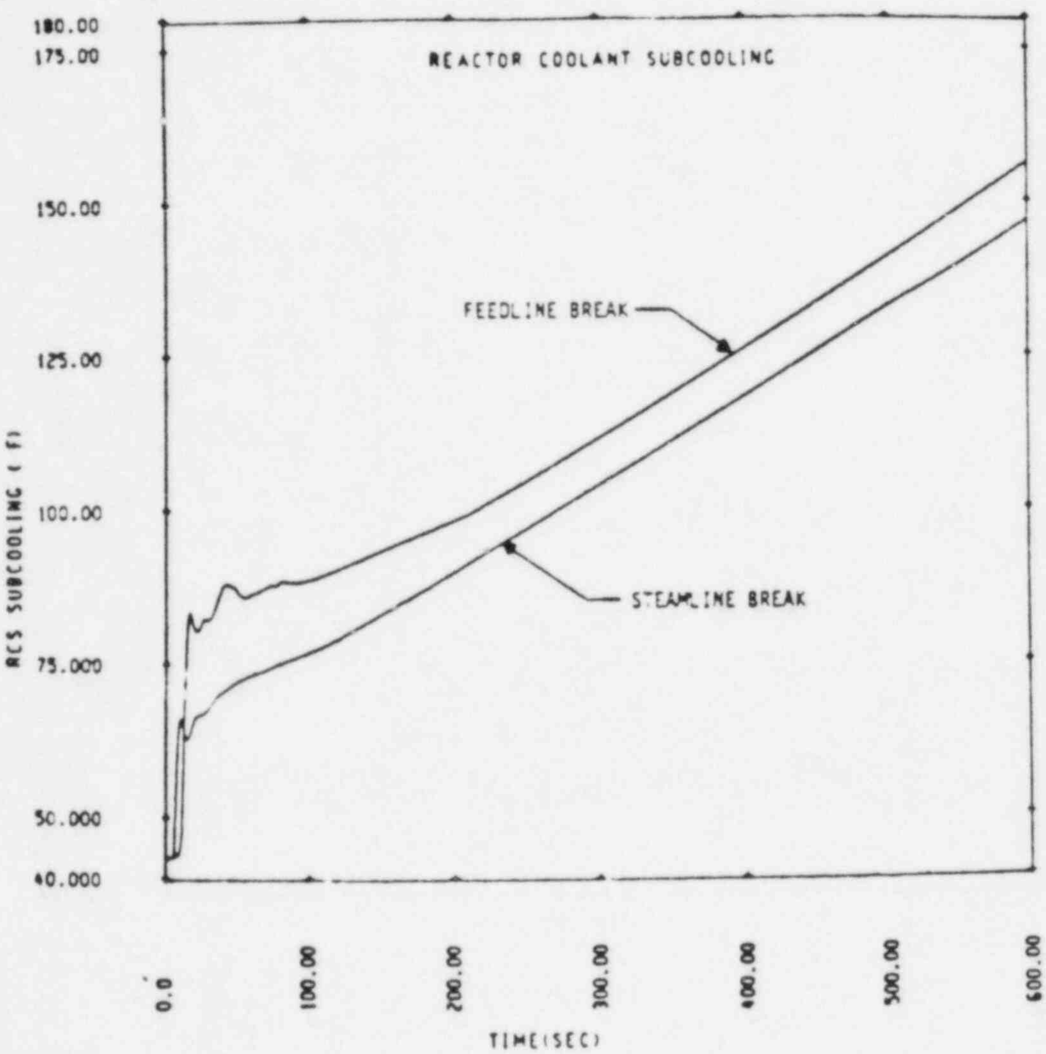
3. Loomis, James, et al., "Reactor Coolant Pump Operation During Small Break LOCA Transients at the Yankee Nuclear Power Station," YAEC-1437, July 1984.



RCS Subcooling for SGTR

at the Yankee Plant

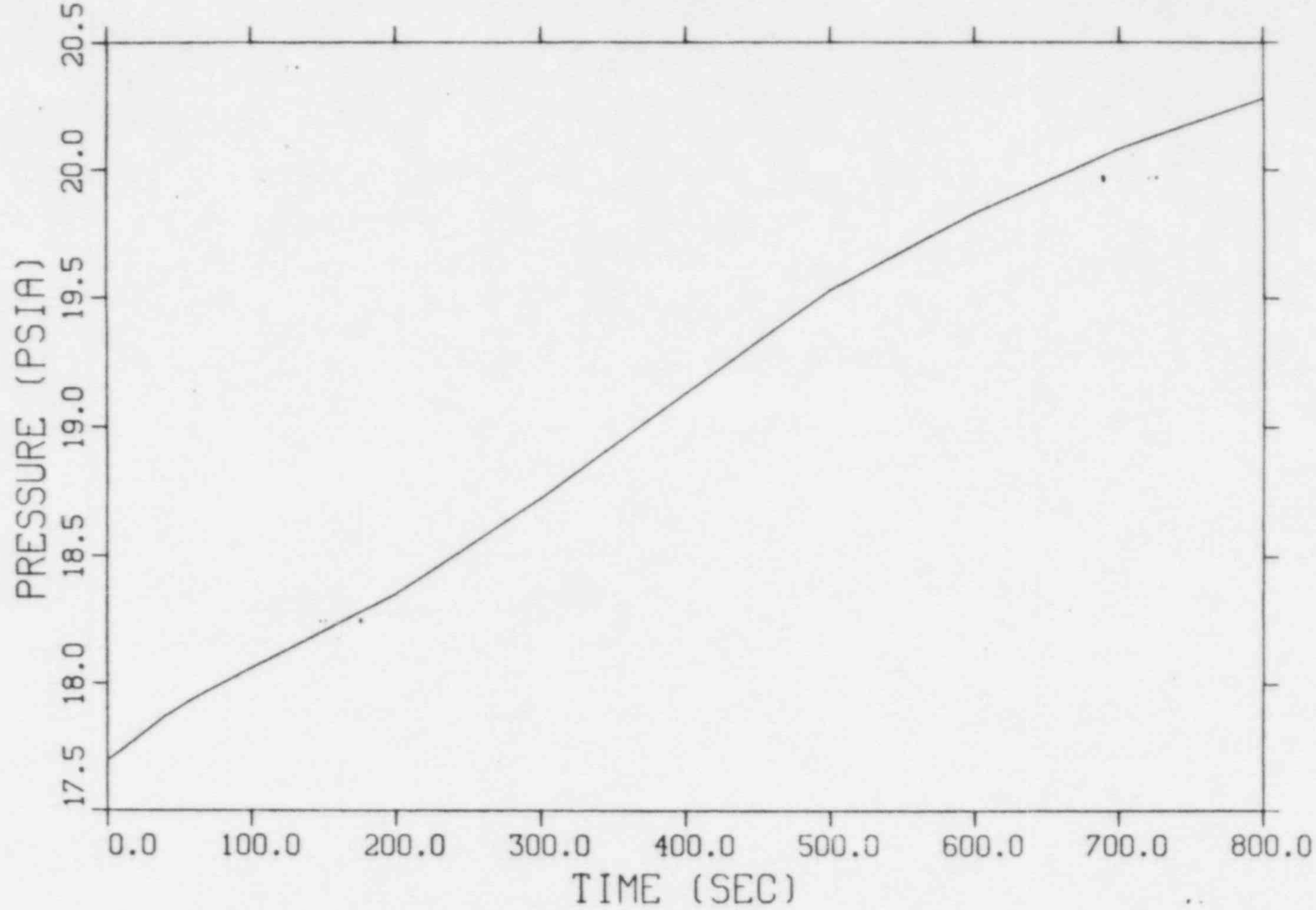
Figure 1



Typical RCS Subcooling Transients

(Reference 2)

Figure 2

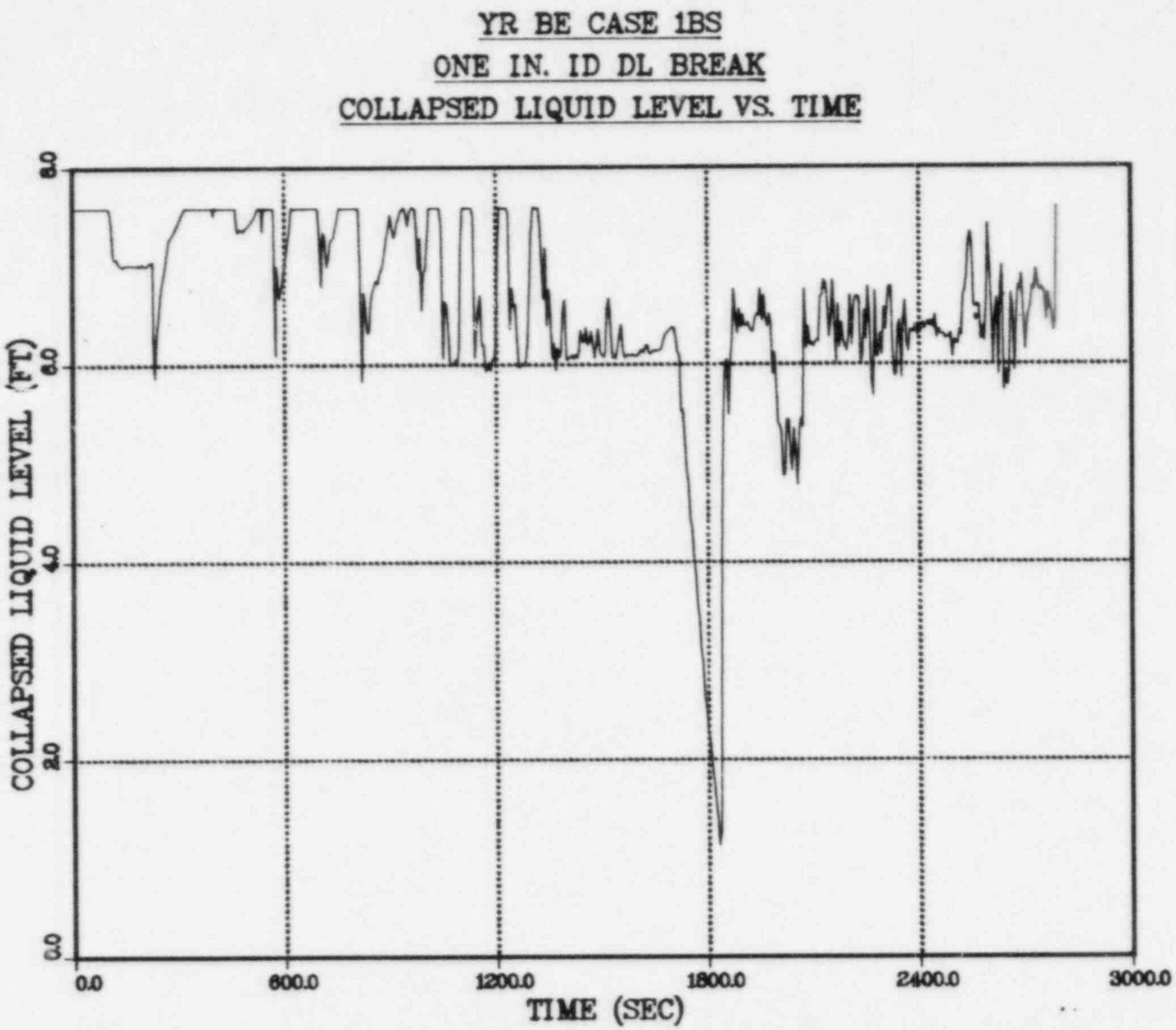


<u>Containment Overpressurization</u>

<u>Due to Rupture of 15/16-Inch</u>

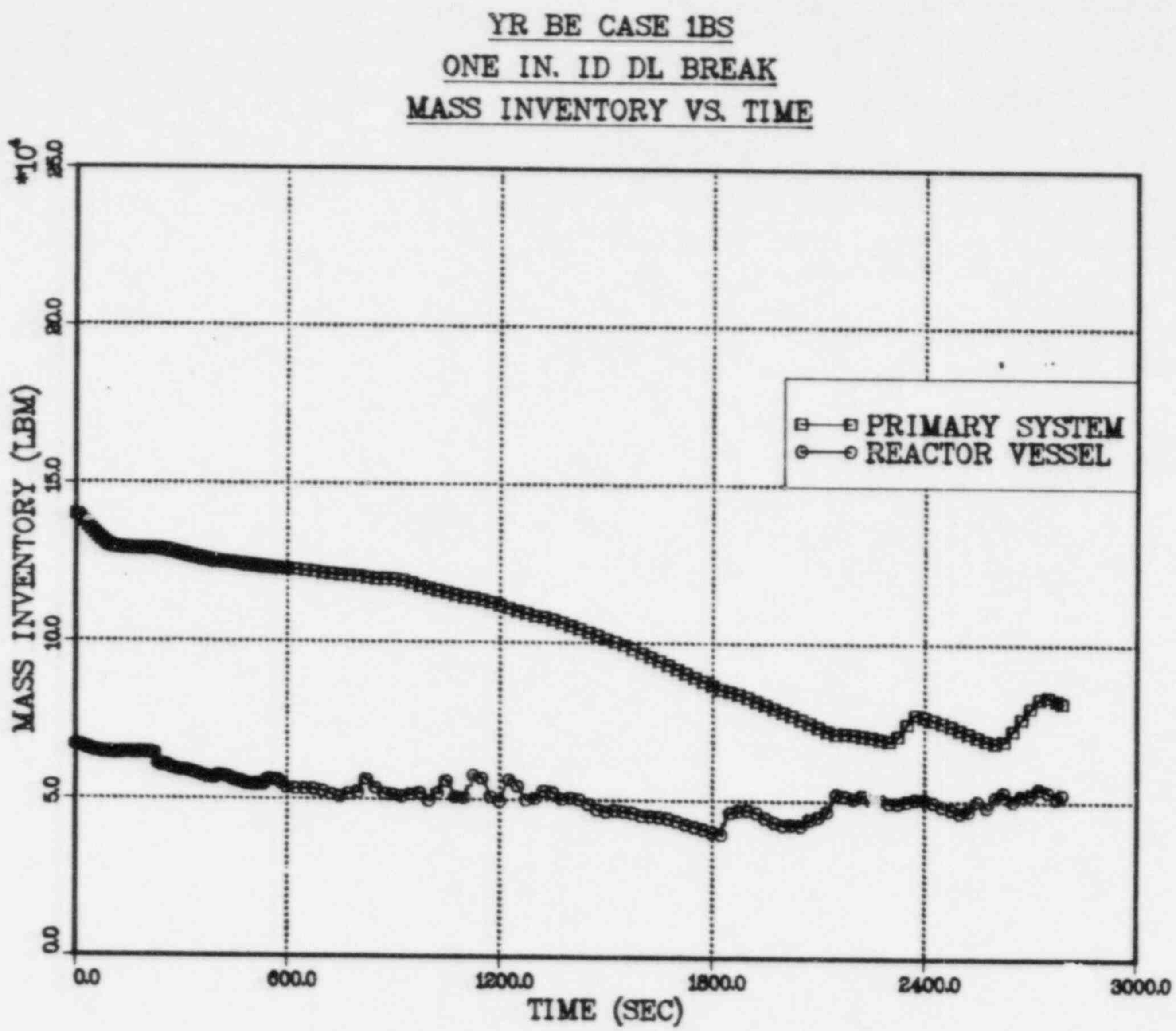
<u>Pressure Tap on the Steam Generator Dome</u>

<u>Figure 3</u>



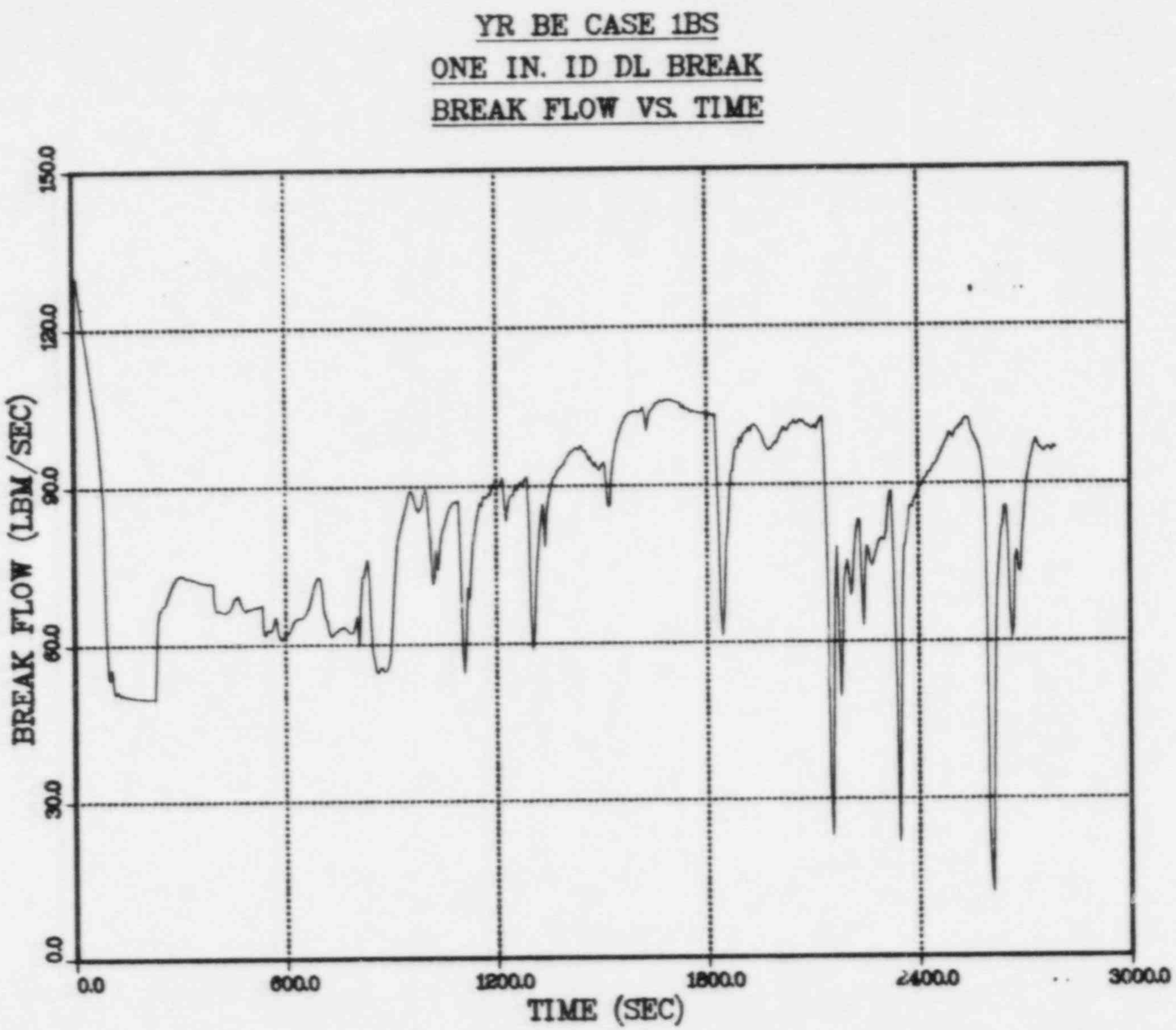
Collapsed Liquid Level in the Core

Figure 4



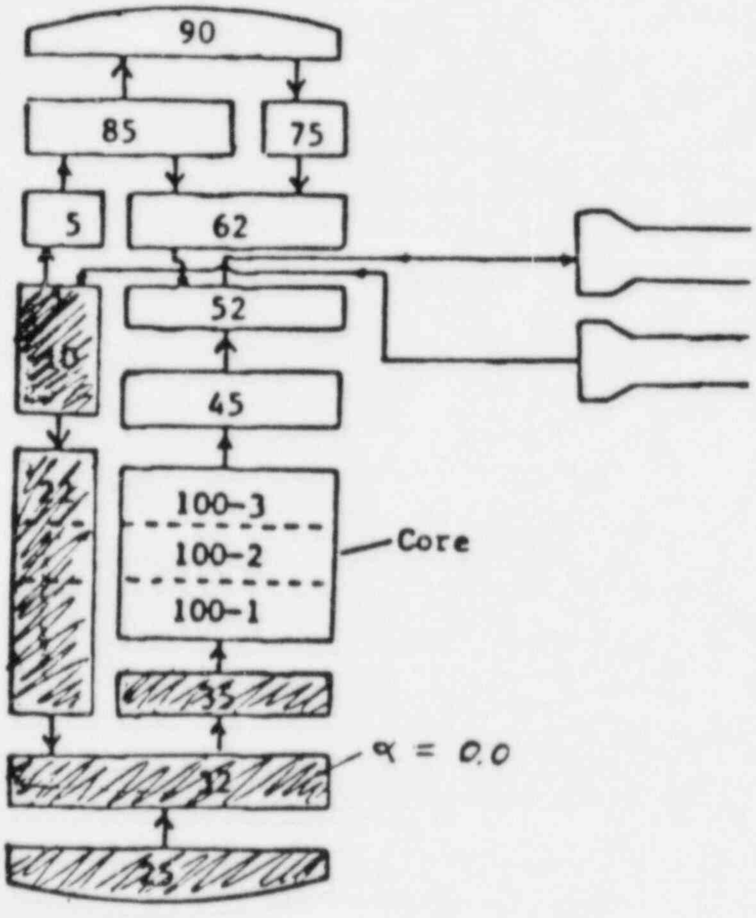
RCS and Vessel Mass Inventory

Figure 5



Break Flow

Figure 6

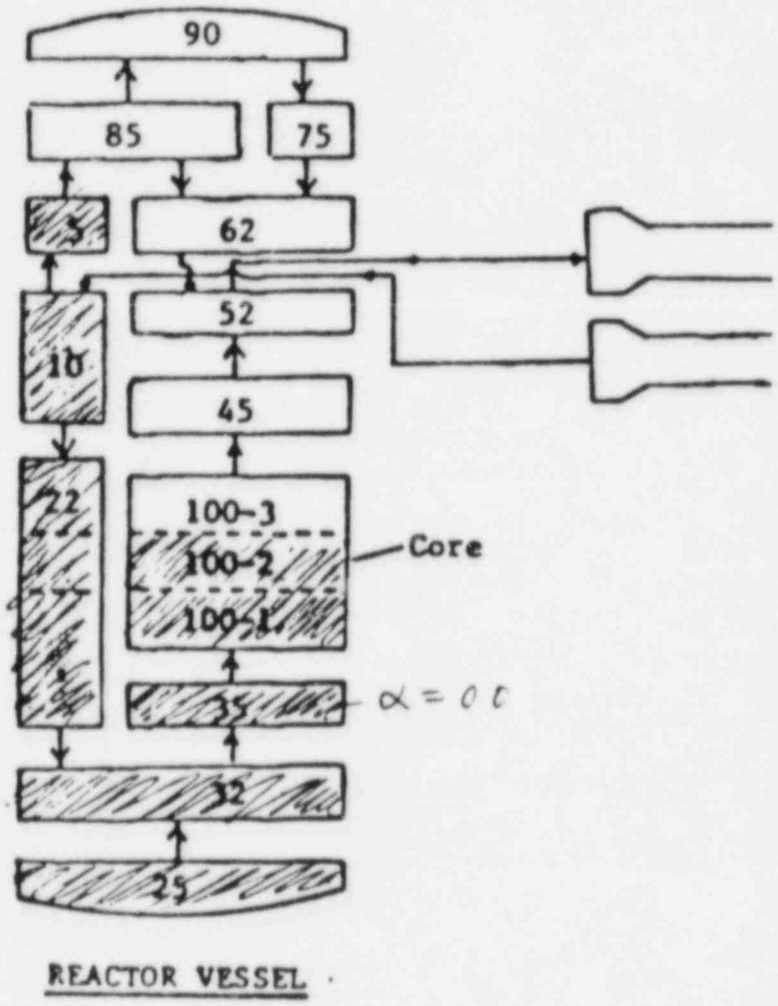


REACTOR VESSEL

Water-Filled Volumes of the Reactor Vessel

1800 Sec.

Figure 7



Water-Filled Volumes of the Reactor Vessel

1850 Sec.

Figure 8