GPU Nuclear

TECHNICAL DATA REPORT

PROJECT: TMI-1	TDR NO. 406	REVISION NO. 1
	BUDGET ACTIVITY NO. 128006	PAGE 1 OF 57
	DEPARTMENT/SECTION Safety Analysis	
	RELEASE DATE 3/28/83	REVISION DATE 5/8/83

DOCUMENT TITLE: SG Tube Rupture Procedure Guidelines

ORIGINATOR SIGNATURE	DATE	APPROVAL(S) SIGNATURE	DATE
Louis C. Lanese	5-8-83	R. L. Jackson	5/9/83
		APPROVAL FOR EXTERNAL DISTRIBUTION	DATE
		[signature]	5-13-83

*	DISTRIBUTION
*	R. W. Bensel
	D. J. Boltz
	T. G. Broughton
*	M. Campagna
*	P. R. Clark
*	J. J. Colitz
	M. J. Graham
	R. J. Glaviano
*	R. W. Keaten
*	G. Lehmann
*	T. Moran
*	S. Newton
*	M. J. Ross
	H. B. Shipman
*	D. G. Slear
	C. W. Smyth
*	R. J. Toole
	P. S. Walsh
*	R. F. Wilson

ABSTRACT:

This document provides technical guidelines for dealing with single and multiple tube ruptures. A significant improvement in procedures will result from reduction of the minimum subcooling margin and RC pump trip on loss of subcooling margin, waiver of fuel-in-compression limits, and revised RCP NPSH limits. Other benefits can be derived from revision of the RC pump restart criteria and from additional guidance regarding OTSG steaming and isolation. Finally, revised guidance is provided for preventing tube leak propagation. It is recommended that the tube-to-shell delta T be limited to 70F^{o} during tube rupture events with cooldown terminated if the limit approaches 100F^{o}.

*COVER PAGE ONLY

A0000030 7-81

GPU Nuclear

DOCUMENT NO.
TDR # 406

TITLE: SG Tube Rupture Guidelines

REV	SUMMARY OF CHANGE	APPROVAL	DATE
1	Minor editorial changes and correction of typographical errors on pages: 2,5,7,10,13, 19,20,22,A-1,A-3,A-4,B-1,A-2,B-2,6,18.	LCL	5-8-83
1	Revised cover page to show shell-to-tube delta T can be controlled below 100F.	LCL	5-8-83
1	Added a List of Tables pp i & iii	LCL	5-8-83
1	Included use of MFW as means to cool OTSG shell. p 5	LCL	5-8-83
1	Indicated that continuous steaming of OTSG is simplest means of meeting OTSG level, pressure and differentail temp. considerations pp. 5,6,10,18	LCL	5-8-83
1	Eliminated reference to RAC for determining when to isolate OTSG based on radiological conditions. p 6	LCL	5-8-83
1	Added Section 2.1.3.1 to discuss control when both OTSG's are isolated.(Also p 10).	LCL	5-8-83
1	Provided discussion and Figure for RCP NPSH limits. Section 2.1.6 and Figure 6. Ref 25,26. & Sections 4.2.1 and 4.2.4.	LCL	5-8-83
1	Revised explanation of ADV & TBV flow capability relative to OTSG flooding (incorrect in Rev 0) pp 11,16	LCL	5-8-83
1	Added Reference to B&W guidance which allows cooldown at 100F/hr during Tube Ruptures without a soak time even if cooldown rate is exceeded. p 11 & Ref 24	LCL	5-8-83
1	Section 4.2.3 revised to account for inability to start either RCP in the A loop.	LCL	5-8-83
1	Added Section 4.2.7.1 and revised 4.2.7 to address Steaming Isolation of OTSG considering the continuous steaming philosophy.	LCL	5-8-83
1	Simplified Section 4.2.8 on cooldown rate.	LCL	5-8-83
1	Added Section 4.2.9 on controlling OTSG shell-to tube differential temperature.	LCL	5-8-83

A0000038 7-82

GPU Nuclear

DOCUMENT NO. TDR # 406

TITLE

SG Tube Rupture Guidelines

REV	SUMMARY OF CHANGE	APPROVAL	DATE
1	Added Section B.7, B.8, and B.9, which were left out of Rev 0 inadvertently.	REL	5-8-83
1	Deleted redundant Section of Part 4 (guidelines)	REL	5-8-83
1	Added Section C.2 through C.6 to discuss the Guidelines Flow Diagram (Figure C-1) in words.	REL	5-8-83
1	Rewrote Appendix E on Process Computer Output and Alarms.	REL	5-8-83
1	Revised Figure 4 to show decay heat levels as a function of time.	REL	5-8-83

TABLE OF CONTENTS

There are a total of 57 pages in this report.

LIST OF FIGURES

LIST OF TABLES

1.0 INTRODUCTION AND BACKGROUND

In November 1981, primary to secondary side leaks were discovered in the tubes of both of the TMI-1 Once Through Steam Generators (OTSG). There are 15,531 tubes in each OTSG. The plant design basis for a steam generator tube rupture (SGTR) accident is the double ended offset severence of a single tube. Since extensive circumferential cracking was discovered in approximately 1200 of the 31,000 tubes, it became clear that a revised set of procedures for dealing with both single and multiple SGTRs should be developed.

This report describes a program which has been formulated to improve existing procedures and operator training by providing improved operator guidelines for dealing with tube leakage and tube rupture events. The guidelines development program will be described in detail, and the major revisions to the existing procedures which have been identified as part of the program will be discussed. The proposed guidelines will then be presented in terms of their overall scope, with a step by step discussion of required operator actions. The analytical evaluations which are the basis for the recommendations, consist of a series of simulations which are ongoing and will be documented in detail in a subsequent report. The guidelines in this TDR were tested at the B&W simulator training cycle beginning in January, 1983. The results of this training experience are discussed. Finally, the overall conclusions and major recommendations of the guidelines development program are documented.

2.0 TECH FUNCTIONS SGTR GUIDELINES DEVELOPMENT PROGRAM

Figure 1 shows the execution of the steam generator tube rupture guid development program. The plan has three main paths: Path 1 is the development of design basis tube rupture guidelines. Path 2 is the development of multiple tube rupture guidelines; and, Path 3, is a benchmark effort to compare the RETRAN and RELAP 5 computer codes. This last effort also includes an evaluation of the B&W ATOG analysis of a single tube rupture using MINITRAP. The purpose of this TDR is to explain paths 1 & 2. The benchmarking and comparison efforts are discussed in a separate TDR describing all of the tube rupture analysis work. None of the computer analysis of Path 3 has been used to justify the recommendations of this report. The analyses were an aid in conceptualizing the physical processes during a tube rupture.

2.1 Development of Design Basis Guidelines (Path 1)

The major activities involved in developing this part of the guideline were to:

1. Search existing industry events and procedures for lessons to be learned about handling tube ruptures.

2. Define allowable steam generator stresses during cooldown (either as cooldown rate or as tube/shell delta T).

3. Determine when OTSG's should be isolated and when they should be steamed.

4. Revise the minimum allowable subcooling margin.

5. Waive fuel in compression limits.

6. Relax RCP NPSH limits.

7. Redefine entry point conditions.

Each of these items are discussed in detail in the following sections.

2.1.1 Literature Search

Several tube rupture leaks have occurred at various operating reactors within the last four years. The experience gained from these events has offered us an opportunity to improve tube rupture guidelines. The major lessons learned from these events have been summarized in various documents from the NRC, INPO, and plant procedures and included in the B&W ATOG tube rupture guidelines (References 1-10). The lessons include the following:

1. Subcooling margin should be minimized to minimize primary to secondary leakage. Subcooling is maintained by keeping the RCS temperature below the saturation temperature with OTSG cooling.

Since the OTSG is in a saturated condition, it is always lower in pressure than the RCS if subcooling is maintained. Therefore, keeping subcooling margin at or near its minimum acceptable value reduces leakage.

In order to maintain the minimum subcooling margin, several plant limits have to be violated: fuel pin-in-compression limits and RCP NPSH limits. The former is acceptable to violate during emergency conditions, while the latter has been reevaluated to determine acceptable emergency operation of the pump.

2. RCP's should be maintained running for several reasons. Pump trip on loss of subcooling margin allows the operator to maintain forced flow for a leak size of up to several tubes while 1600 psig ESAS is much more restrictive. Forced RC flow provides several benefits during a tube rupture. First, they assure that steam voids do not form in the hot leg U bends or upper vessel head. Steam voids in these locations can interrupt natural circulation or prevent RCS depressurization. Second, RCP operation results in a lower primary to secondary differential pressure for a given subcooling margin (since core delta T is smaller with the RCP's running). Finally, with RCP's running, pressurizer spray is available and RCS pressure control is not dependent on the PORV or pressurizer vent.

   Main feedwater can be used if RCP's are running; with pumps off, emergency feedwater must be used, which is less effective in cooling the OTSG shell, thereby increasing tube to shell delta T. (i.e., tube tensile loads).

3. RCS pressure should be maintained low enough to prevent secondary side safety valves from lifting. HPI flow was not throttled sufficiently in the Ginna event of January 25, 1982 and the steam generator filled with water. Since the RCS pressure was above the SG safety valve setpoint, the safeties opened resulting in an atmospheric release of radioactivity. Moreover, the safeties were forced to pass liquid, which might cause the open failure of the valves.

4. RCS Degassing

   RCP NPSH limits at Ginna required shutting down of the reactor coolant pumps at low pressures. Shutting the pumps allowed noncondensible gases to collect in the top of the steam generator tube U bends. These trapped gases prevented RCS depressurization for many hours. An analogous situation might occur at the hot leg U bends. The TMI-1 design has always had capability of venting noncondensable gases from the U bends, however, which can be used if RCP's are not available.

5. BWST Inventory

The Oconee tube leak of September 18, 1981 resulted is a sustained (17 hour) leakage from the RCS to the OTSG's. This leakage caused the generator to fill. In order to prevent steam line filling, the operators at Oconee transferred water out of the OTSG's. In effect there was a once through cooling path from the BWST through the core and out of the OTSG's. This experience illustrated the need to assure adequate BWST Inventory for core cooling. Second, it highlighted the need to store radioactive water in the plant during a prolonged RCS cooldown.

6. Tube to Shell Delta T

A tube leak at Rancho Seco in May 1981 yielded evidence of the importance of controlling OTSG tube/shell delta T. The existing limits and precautions at TMI-1 is 100°F. However, before tube/shell delta T exceeded 100°F, the leaking tube was placed under tensile stress and the tube was pulled into a circumferential tear. Maintaining tube/shell delta T limits are important during tube rupture and are discussed in more detail below.

2.1.2 <u>Limiting OTSG Tube Stresses</u>

Steam generator tube stresses are generated as a result of tensile loads placed on the tubes. These tensile loads come from two load components. The first is the temperature differential between the tube and steam generator shell. As the RCS temperature decreases, tube temperature decreases. At some point the difference in temperature between the colder tubes and warmer shell is sufficient to result in tensile stresses that pull apart a leaking tube. This topic has been the subject of extensive analyses within GPUN in conjunction with B&W, EPRI, and MPR and the subject of a separate report (see Ref. 15).

The second load component is from OTSG pressure loading on the tubesheet which causes elongation of the shell. Isolation of the OTSG causes the tube/shell difference to increase while adding a tensile load on the tubes by elongating the shell via pressure loading. Structurally there are compensating effects involved in mitigating these two load contributors. Rapid depressurization eliminates the pressure induced stress but aggravates the delta T induced stresses. The optimum OTSG cooldown/depressurization rate has not been determined. However, it is known that isolating the OTSG at 1000 psig is not the best means of reducing stress. Cooldown/depressurization is the preferred method.

There are three limits for tube/shell delta T that presently apply to TMI-1. Plant "Limits and Precautions" (Ref. 22) limit delta T to 60°F during heatup and to 100°F during cooldown with one OTSG isolated. This value of 100°F assumed that tubes had no more than

40% through-wall cracks. In reference 23, B&W established 142°F for a cooldown using both OTSG. The 70°F value in this TDR is proposed as a guide in determining an acceptable cooldown rate. If delta T can be maintained at or below 70°F, the operator has optimized the plant cooldown rate. The 70°F limit more conservatively assumes that tubes in the OTSG are leaking below a detectable limit. A 70°F value limits propagation of these cracks.

Control of tube to shell delta T is accomplished in several ways. First by cooling the OTSG liquid (steaming) to allow the metal shell to cool. Second, by providing cool, main feedwater into the downcomer. If neither of these methods works, the RCS cooldown must be decreased until the OTSG shell cools sufficiently. If reducing the cooldown doesn't work, then the cooldown must be terminated.

2.1.3 Steaming, and Isolation Filling of the Leaking OTSG

Isolation of the leaking OTSG can result in the overfilling of that generator. It is preferable to prevent overfilling, however, to allow plant cooldown in an expeditious manner. If the OTSG fills, it becomes a large pressurizer (as evidenced by the Ginna event). The time it took to cool down this mass of hot water greatly extended the cooldown of the plant. Steaming also maintains some natural circulation flow in the hot leg. This flow cools the hot leg U bend and decreases the chances of steam void formation.

As discussed in Section 2.1.2, steaming and depressurization of the OTSG also reduces OTSG tube stresses. However, depressurization of the OTSG also increases leakage rate. As discussed in Section 2.2.2, the OTSG pressure should be below RCS pressure to promote flow through the hot leg. The optimum OTSG control results in 1) depressurization of the OTSG without causing large delta T's; 2) minimum RCS leakage; 3) promotion of natural circulation flow in the hot leg; and 4) positive leakage from the RCS into the OTSG to assure hot leg cooling in the absence of natural circulation. The optimum pressure control scheme to meet this criteria has not been determined analytically.

Meeting all four of these criteria will probably result in a nearly continuous steaming of the affected OTSG. Moreover, intermittent steaming of the OTSG's will result in release of all the noble gases transported into the OTSG from the RCS. Therefore, the TDR recommends continuous steaming of the OTSG's. The advantages of continuously steaming the affected OTSG's are:

1. All of the above OTSG control conditions are met.

2. The operator follows his normal cooldown procedures.

3. Plant response is symmetric.

Continuous steaming should result in a more rapid cooldown than intermittent steaming because of tube/shell delta T limitations. Cooldown at 100 F/hr using the unaffected OTSG will result in a 70 F delta T limit in 1-2 hours. From this time on, the OTSG would have to be steamed. Similarly, the OTSG would have to be steamed to maintain natural circulation.

Although it is highly desirable to prevent steam line filling, there are certain circumstances which dictate that the OTSG should be filled. The Engineering Mechanics Section of GPUNC has established the capability of the steam lines to sustain the water hammer and dead load effects of flooding the steam lines (Ref 11). This analysis shows that the loading is acceptable without pinning (except for the dead load effects during a design basis earthquake). Since this combination of events is extremely remote, the procedures have been modified to allow filling of the OTSG.

The guidelines have the operator fill the OTSG's only under two circumstances. The first condition is that BWST level decreases below 21 ft. At this level, there is still sufficient inventory to flood both steam lines and put about 30,000 gallons of water into the containment building (Ref 12). This amount of water is sufficient to provide adequate NPSH in an LPI to HPI "piggyback" mode of core injection from the reactor building sump (Ref. 13).

A second reason to fill the OTSG is for radiological considerations. The OTSG should be isolated if offsite doses are approaching levels which would require declaration of a Site emergency. It should be noted that a Site Emergency may already have been declared based on OTSG leakage rate. Nevertheless, this level provides a rationale for deciding that release rates are high enough to warrant OTSG isolation.

Consideration was given to defining isolation conditions based on RCS activity levels, meteorology and steam line radiation levels. RCS activity level cannot be correlated to offsite releases, since offsite dose will be affected by the location of the tube leak in the OTSG, availability of the condenser and plateout and decontamination factors. It is also undesirable to isolate the OTSG based on assumed, meteorological conditions. The most desirable approach is to isolate based on actual releases occurring during the event.

2.1.3.1 Steaming, Isolation and Filling with Both OTSG's Leaking

Isolation and steaming of the OTSG's must be addressed for leaks in both OTSG's. Once RCS temperature is below 540 F, a choice has to be made regarding OTSG isolation. Both OTSG's should be steamed unless either the BWST level of offsite release criteria is reached. If both OTSG's are steamed, the all steam loads from both OTSG's should be isolated except for the TBV's/ADV's. All other steaming, isolation and filling criteria should be followed.

The basic guidance for leaks in both OTSG's is to:

1. Steam both OTSG's.

2. Isolate all steam loads except for those required for decay heat removal.

3. Steam at least one OTSG until the radiological isolation criterion cannot be met steaming only one OTSG.

4. Follow all other OTSG steaming, filling and isolation criteria as written.

2.1.4 Minimum Allowable Subcooling Margin

A primary goal during a tube rupture is to minimize offsite dose. Minimizing leakage from the RCS is the first line of defense. Leakage from the primary to secondary is determined by the size of the leak, and by the differential pressure between the RCS and OTSG. Primary to secondary differential pressure is controlled by fixing the degree of RCS subcooling. Once secondary pressure is fixed cold leg temperature is determined. For any time, decay heat is fixed. RCS flow (which is determined by OTSG level or RCP operability) then determines hot leg temperature. Reactor coolant pressure or HPI flow then fixes the degree of subcooling. Since the operator controls OTSG and RCS pressure and HPI flow, he is in control of the subcooling margin, hence primary to secondary delta P. Figure 2, illustrates the effect of subcooling margin on primary to secondary leakage.

Figure 3 illustrates the relative effects of a cooldown with RCP's off using 50°F and 25°F subcooling. Even at the maximum cooldown of 100°F/hr, the integrated leakage differs by a factor of two.

2.1.5 Waive Fuel in Compression Limits

Fuel pin in compression limits are specified to assure that fuel pins are always in compression above 425°F in order to prevent detrimental orientation (i.e., radial orientation of hydrides) (Ref. 14). These limits require a high subcooling margin for RCS pressures ranging from 1350 psi to 550 psi. In correspondence dated January 20, 1983, (Ref. 14) B&W confirmed that violation of these limits during tube rupture events is acceptable. When these limits are violated it is important that the pressure and temperature versus time be recorded so the effects on cladding can be evaluated. The evaluation must determine whether clad ballooning or incipient cracking has been induced.

2.1.6 Reactor Coolant Pump NPSH Limits

RCP NPSH requirements place limitations on the minimum subcooling margin. At low RCS pressures RC pump NPSH limits approach 100°F of subcooling. However, general centrifugal pump test data have shown that NPSH requirements are substantially reduced at water temperatures above 250°F. A review of TMI-1 test data on the subject reactor coolant pumps indicates a single loop flow of 98,500 gpm with two loops in operation with one pump per loop. The pumps' manufacturer (Westinghouse) has provided required NPSH at various pump suction temperatures (Reference 25) for the flow associated with two pump operation. The NPSH available, as indicated by the saturation meter monitored in the hot leg, is then calculated by considering the total pressure drop from the hot leg to the pump's suction (Reference 26). The resulting NPSH requirements for 2 pump operation (one per loop) are shown in Figure 6. Also shown is the 4 pump operation NPSH curve which has considered the changed flow distribution in the coolant loops. In addition, normal NPSH curve and 25F° subcooling curve are shown for comparison purposes.

The emergency NPSH limits are intended for operation of RCP's during abnormal and emergency conditions such as small break LOCA, SG tube rupture, station blackout and secondary side upset events.

2.1.7 Procedure Entry Point Condition

The use of an emergency tube rupture procedure should be limited to situations where normal limits (e.g. fuel pin-in-compression and RCP NPSH) are being waived. The guidelines' entry point condition is chosen as 50 gpm. A leak rate of this magnitude would be expected from the complete separation of one tube (as opposed to 385 gpm for a double-ended offset of one tube). Less likely, (but more serious) would be leakage of this extent from a number of tubes. Both situations warrant entering the emergency procedure. Below this limit, plant cooldown should be achieved within normal limits unless additional equipment failures occur.

2.2 Development of Multiple Tube Rupture Procedure Guidelines (Path 2)

The treatment of multiple tube ruptures relied on several sources of information. The Ginna tube leak exceeded the single tube flow for a B&W plant and also resulted in a loss of subcooling. Therefore, that event legitimately represented a multiple tube rupture. The Oconee tube leak with a delay in getting onto decay heat removal prompted analysis of water inventories required to assure a source of water for HPI cooling.

Besides plant operating experience, this TDR investigated the following aspects of multiple tube ruptures:

1. Revision of the RCP trip and restart criteria.
2. OTSG steaming and level control.
3. Establishment of criteria for going on feed and bleed cooling.
4. Cooldown/depressurization.

2.2.1 Revision of RCP Trip and Restart Criteria

Based on initial small break LOCA analyses received from PWR vendors in 1979, NRC concluded in NUREG 0623 that delayed trip of reactor coolant pumps during a small break LOCA can lead to predicted fuel cladding temperatures in excess of current licensing limits. At the time of RC pump trip the liquid that was previously dispersed around the primary system through pumping action now collapsed down to low points of the primary system such as the bottom of the vessel and steam generators. This separation results in significant uncovery of the reactor core if system voiding is high enough, due to an insufficient amount of liquid being available to provide acceptable core cooling. Unacceptable consequences would result from delayed reactor coolant pump trip only for a range of small breaks LOCA (.025 to 0.25 ft) and a range of trip delay times after accident initiation. Based on these findings, a meeting of utility vendors and owners was held with NRC in September 1979. At this meeting it was agreed that the 1600 psig ESAS signal provided timely Control Room indication for manual action to prevent possible voiding scenarios.

GPU had B&W reevaluate these LOCA scenarios assuming RCP's were tripped on loss of subcooling margin. The conclusion of that reanalysis was that loss of subcooling was an acceptable alternative to pump trip on 1600 psig ESAS. In March 1983, the NRC Staff required Utilities to reevaluate their pump trip schemes (Ref. 17). GPUNC provided an evaluation of the pump trip criterion and a schedule for implementing this criterion by May 1, 1983.

The advantages of maintaining RCP's are that during Steam Generator Tube Ruptures in which minimum subcooling margin is maintained, continuous RC pump operation assures expeditious cooldown with a minimum primary to secondary differential pressure. This change in criteria for RCP trip will allow RCP's to be operated for a greater spectrum of tube ruptures (including ruptures beyond the design basis) and to reduce the offsite doses for those events. Figure 3 illustrates the reduced leakage possible with RCP's on. Similarily, restart of RCP's has a great advantage. During tube ruptures, primary to secondary differential pressure decreases rapidly since OTSG pressure is high. Leakage flow is exceeded by HPI flow and subcooling margin should normally be restored within 20-60 minutes after larger tube ruptures. Restarting RCP's provides pressurizer spray, and prevents void formation in the hot legs U bends and reactor vessel head.

2.2.2 OTSG Steaming and Level Control

The concepts of OTSG steaming are nearly the same for multiple tube ruptures as for single tube ruptures. The OTSG pressure should be controlled to prevent lifting of safety valves (i.e. stay below 1000 psig). Level should be maintained below 95% on the operate range. There are several other issues to be considered for multiple tube ruptures, however. First, large tube ruptures may result in RCP trip. The OTSG's should be steamed to maintain natural circulation in the affected loop. Natural circulation flow will minimize the chances of drawing a bubble in the hot leg U bend. Continuous steaming of the OTSG allows all of these considerations to be accomodated.

It is important to recognize that a large tube rupture with loss of subcooling is a LOCA condition. Therefore, it is required to raise OTSG level to 95% to assure that liquid level in the tube region is high enough to allow water to flow into the core during boiler condenser cooling.

Section 2.1.3.1 discusses steam generator isolating steaming and filling criteria when both OTSG's are leaking. This discussion also applies when the RCS subcooling margin has been lost.

2.2.3 Criteria for Feed and Bleed Cooling

Analyses of multiple tube ruptures indicate that existing plant procedures for establishing feed and bleed cooling are correct. Feed and bleed cooling should be initiated when the OTSG heat sink is not available. If both steam generators are isolated during a tube rupture, the PORV should be opened with full HPI turned on. An

additional complication for tube ruptures, however, is the potential to flood the OTSG's and force open the safety valves under this condition. If RCS pressure is below 1000 psig, the PORV is capable of removing decay heat (even with liquid relief) (see Figure 4). Therefore, the operator can control RCS pressure by throttling HPI. Moreover, with RCS pressure below 1000 psig the OTSG safety valves will not lift.

If RCS pressure stays above 1000 psig, however, the operator must take action to prevent safety valve lifts. A situation with pressure above 1000 psig and neither OTSG available requires the opening of the TBV or ADV's to control level. Either the ADV or TBV have sufficient steam capacity at high OTSG pressure both to remove decay heat. The TBV's also have sufficient capacity to prevent OTSG flooding. As decay heat decreases, steaming can be terminated when RCS pressure goes below 1000 psig and is controlled by the PORV and HPI.

2.2.4 Cooldown/Depressurization

Analyses of multiple tube ruptures demonstrated that subcooling margin should be regained in 20-60 minutes (see Figure 5). RCP's can be started and a forced flow cooldown institutes. Even if RCP's are not available, the cooldown during a multiple tube rupture can be accomplished within the single tube rupture guidelines. If equipment failures prevent a normal natural circulation cooldown, then the plant would be cooled down with feed and bleed cooling. This maneuver would probably require initiation of feed and bleed cooling in the HPI/LPI "piggyback" mode. Existing plant procedures give correct guidance about when to initiate this mode (BWST level below 3 ft.).

Guidance from B&W on PTS/Brittle Fracture limits requires a "soak time" to allow the vessel wall to reach the RCS temperature. However, B&W has also recommended that the "soak time" is not required during tube rupture events in which a rapid cooldown is necessary (Reference 24).

Steam releases during multiple tube rupture events can be minimized by judicious use of the EFW, HPI and TBV's. Full HPI flow, in conjunction with throttled EFW flow allows a 100°F/hr cooldown without having to steam either OTSG.

3.0 DISCUSSION OF MAJOR REVISIONS TO EXISTING PROCEDURES

The development of the design basis guidelines discussed in Section 2.1 identified a number of areas which were investigated to determine where specific changes should be incorporated into the new guidelines. This section further explains what areas of the guidelines should be revised.

3.1 Basic Plant State

3.1.1 The following assumptions apply to the single tube leak/rupture guidelines.

1. Subcooling margin (SCM) is maintained.
2. Only one OTSG is affected.
3. Condenser is available.
4. Reactor Coolant Pumps (RCP's) remain on.
5. Decay heat is removed by the intact OTSG until the Decay Heat Removal (DH) system can take over.
6. The affected OTSG can be steamed to maintain less than 95% level (Operating Range) and less than 1000 psig.

Contingencies

The revised guideline will have provisions to deal with the following circumstances:

1. RCP's not available.
2. Condenser not available.
3. High radiation releases offsite.
4. Tube leaks in both OTSG's (but one OTSG remains capable of removing decay heat).
5. Steam lines associated with leaking OTSG flood.

3.1.2 Tube Rupture Guidelines For Loss of Subcooling

Tube leaks in this category generally go beyond the licensing basis, or are otherwise remarkable due to plant conditions (aside from the tube leak) or equipment malfunction.

The following conditions were assumed in developing guidelines for this category of tube rupture event.

1. More than one tube leak.

2. SCM is lost.

3. RCP's are unavailable.

4. Pilot-operated relief valve (PORV) and Reactor Coolant System (RCS) high point vents are available.

5. Unaffected OTSG can be steamed.

Contingencies

The revised guideline will have provisions to deal with the following additional circumstances:

1. Both OTSG's are affected.

2. Both OTSG's affected, but one OTSG remains capable of RCS heat removal and either a) the PORV is unavailable or b) RCS pressure stays above the main steam safety valve setpoint due to void formation in the RCS.

3. Neither OTSG is capable of removing decay heat, and either a) the PORV is available, or b) the PORV is unavailable.

3.1.3 Revised Equipment Limits & Operating Practices

During the course of the analyses leading to the guidelines provided in Section 4.0. It became apparent that certain normal equipment limits and operating practices should be adjusted to effectively deal with a tube leak/rupture. These changes will help accomplish the following:

1. Mitigate or prevent further OTSG damage.

2. Maximize the cooldown rate to cold shutdown.

3. Minimize SCM (thus minimizing primary to secondary leakage).

4. Maximize RCS pressure control options.

An Event Tree showing the various possible developments of an OTSG tube leak appears as Appendix D to this report.

3.2 Discussion of Guidelines

Appendix C provides a logic diagram of (with a written discussion of those guidelines) the tube rupture guidelines. This section of the report describes the guidelines shown in that diagram. The symptoms of the tube rupture procedure define the entry point conditions when the emergency procedure is used. This procedure need only be entered for situations where a rapid depressurization of the plant is warranted. When such conditions warrant, then the plant should be shut down and cooled down as expeditiously as possible and certain normal plant limits (RCP NPSH, normal tube/shell delta T and fuel in compression limits) are waived.

3.2.1 Immediate Actions

The tube leak in question may not be large enough to cause a reactor trip. In such a case, the operator begins a load reduction as rapidly as possible without causing a reactor trip (10%/min.). Avoiding a reactor trip prevents lifting of the OTSG safety valves.

3.2.2 Followup Actions - Subcooling Maintained and RCP's Available

Once the load reduction is initiated, the operator has several major goals to achieve while bringing the plant to a cold shutdown condition. First, he must prevent lifting of the OTSG safety valves; second, isolate the affected OTSG to prevent unnecessary radioactivity releases; third, minimize primary to secondary leakage by minimizing primary to secondary differential pressure; and, fourth minimize stresses on the OTSG tubes by limiting tube/shell delta T. Finally, the operator will minimize offsite dose by allowing the leaking OTSG to flood if offsite doses are large enough (approaching levels at which a Site Emergency would be declared).

The major differences between the existing plant procedure and the proposed new procedure would be the following:

3.2.2.1 Maintain a Minimum of 25F° Subcooling

Minimizing subcooling margin means that primary to secondary differential pressure is also minimized, which reduces leakage and offsite doses making the event more manageable.

3.2.2.2 Steaming/Isolation Criteria for the Affected OTSG

The present procedure allows the operator to let the OTSG fill anytime that RCS pressure is below 1000 psig. The revised procedure has the operator steam the OTSG for these purposes: First, to prevent lifting of the OTSG safety valves. Second, to prevent the generator from filling.

3.2.2.3 Tube-to-Shell Delta T

Plant limits and precautions require maintaining the OTSG tube temperature within 100°F of the shell temperature. A tube to shell delta T of 70°F limits stresses and minimizes the chances of increasing the leak size.

3.2.3 Followup Actions (Automatic Reactor Trip has Occurred)

All of the followup actions discussed above still apply when the tube leak is large enough to cause an automatic reactor trip. In addition, the following procedure changes would apply.

3.2.3.1 RCP Trip With a Loss of Subcooling Margin

Present plant procedures require RCP trip on initiation of 1600 psig ESAS. Rupture of one or a few OTSG tubes will likely result in RCS depressurization to the HPI setpoint, but may not result in a loss of SCM.

3.2.4 Followup Actions for Loss of Subcooling

The third section of the tube rupture procedure is entered when RCS subcooling is lost. Here, the operator must treat LOCA, as well as tube rupture symptoms. He is then able to pursue the followup tube rupture actions. All of the guidance for followup actions without loss of subcooling apply.

The objective in this portion of the procedure is to maintain natural circulation (if possible), reestablish subcooling margin, restart a reactor coolant pump, and return to the section of the procedure for forced flow cooldown. If the affected OTSG cannot be steamed for either radiological or equipment reasons, then EFW is used to control OTSG pressure. Essentially, EFW is used as a pressurizer spray to keep the leaking generator slightly lower in pressure than the RCS. The benefits in controlling pressure are:

1. safeties will not lift.

2. the steam generator will not control RCS pressure.

3. there will not be backleakage into the RCS of water or steam from the OTSG.

4. leakage from the RCS to the OTSG will be small since differential pressure will be small.

5. the small flow through the hot leg may prevent void formation in the hot leg.

If subcooling is regained in the RCS, then HPI is throttled, RCP's are started and the operator continues the cooldown. If unable to start an RCP in the loop containing the pressurizer, then start both RCP's in the opposite loop.

The reasons for restarting RCP's are similar to the reasons for not tripping them on low RCS pressure. If subcoolilng margin is lost immediately after RCP restart, it implies that the increased RCS flow has caused voids in the system to collapse, thus dropping RCS pressure. Allowing 2 minutes for SCM recovery prevents incessant "pump bumping," but keeps the RCS out of the fuel damage region.

If subcooling cannot be restored, the operator cools the plant down on natural circulation unless the OTSG heat sink is lost (for example, due to loss of natural circulation in the unaffected loop). With no steam generator heat sink, the operator must put the plant in a feed and bleed cooling mode. Feed and bleed cooling is initiated by isolating the OTSG's, assuring full HPI is on and opening the PORV. With RCS pressure below 1000 psig, water relief out of the PORV is sufficient to keep the core cooled (See Figure 4). If the OTSG heat sink is restored, the feed and bleed mode is terminated and a natural circulation cooldown is reinitiated.

If RCS pressure stays above 1000 psig during feed and bleed cooling (e.g., the head bubble prevents depressurization or the PORV fails closed) then the secondary side safety valves have to be protected from challenge. The operator controls OTSG pressure with whatever means are available (turbine bypass, EFW or Atmospheric Dump Valves.). When the OTSG is about to fill, the operator opens the ADV and leaves it open. This action minimizes the chances that safety valves will be forced to relieve water and/or steam and fail open. The steaming capacity of an ADV at 1000 psig exceeds decay heat levels within several minutes after reactor trip. HPI capacity exceeds the capacity of one ADV. Therefore, the RCS pressure can be controlled at 1000 psig in this mode without lifting safety valves. Subcooling margin can be regained and the plant cooled down in this mode until an OTSG heat sink can be restored or until the plant can be put on decay heat removal.

4.0 TUBE LEAK/RUPTURE GUIDELINES

4.1 Scope

The guidelines will deal with tube leaks in excess of 50 gpm. Primary-to-secondary tube leak rates less than 50 gpm will be handled in accordance with "Guidelines for Plant Operations with Steam Generator Tube Leakage," TDR 400 (Ref. 16).

4.2 Guidelines & Limits

This section provides plant specific technical guidelines for tube rupture events which can be used to generate plant Emergency Procedures.

4.2.1 Subcooling Margin Requirements

Control Reactor Coolant System (RCS) subcooling margin (SCM) between 25°F and 50°F. Maintain SCM as close to 25°F as possible consistent with the RCP NPSH curve of Figure 6 and while waiving fuel pin-in-compression limits.

This will minimize primary to secondary differential pressure, thus minimizing the leak rate.

4.2.2 Reactor Coolant Pump Trip Criterion

Trip Reactor Coolant Pumps (RCP's) when SCM is lost.

4.2.3 Reactor Coolant Pump Restart Criteria

When the required subcooling margin (25°F) has been established, restart 1 RCP per loop. If unable to start an RCP in one loop, start both RCP's in the opposite loop.

Note: If subcooling margin is lost immediately after pump restart and does not return within 2 minutes, the RCP's must be tripped again and not restarted until SCM is regained.

4.2.4 Reactor Coolant Pump NPSH for Emergency Operations

The attached curve (Figure 6) depicts the RCP NPSH limit to be used during a cooldown with a tube leak.

4.2.5 High Pressure Injection Throttling Criteria

Throttle HPI when SCM requirements are met and pressurizer level comes back on scale. (Note that the other HPI throttling criteria remain unchanged.)

4.2.6 OTSG Level

Take both OTSG's to 95% on the Operating Range if SCM is lost.

4.2.7 OTSG Isolation/Steaming Criteria

When the leaking OTSG is identified, close all steam valves except the ADV's and TBV's.

Note: Do not close MS-V1D until an alternate source of gland steam is available.

When RCS T_{hot} is less than 540°F the affected OTSG must be isolated if:

(a) Borated Water Storage Tank level is below 21 ft., or

(b) Offsite dose projections approach the level requiring a Site Emergency (50 $_{m}$Rem).

Note: If both OTSG's are leaking and isolation is required based on offsite dose projections, first isolate the OTSG with the higher leakage. If such a distinction cannot be made, isolate one OTSG and reevaluate offsite dose projections.

4.2.7.1 Pressure Control of an Isolated OTSG

Steam the affected OTSG(s) only:

1. To keep OTSG pressure below 1000 psig,

2. If the plant is on feed and bleed cooling and RCS pressure is above 1000 psig.

Note: (a) If the OTSG must be steamed and is below 600 inches (i.e., below the Emergency Feedwater nozzels) use Emergency Feedwater to spray the OTSG steam space.

(b) If the OTSG must be steamed and is filled above the EFW nozzles (600 inches) open the Turbine Bypass Valves or Atmospheric Dump Valve on the affected OTSG.

4.2.8 Cooldown Rate During a Tube Leak Event

The cooldown rate shall be limited to a maximum of 1.67F°/min (100F°/hr) whether on forced or natural circulation.

Note: Steaming of the OTSG's may not be required if OTSG level is being increased using EFW.

4.2.9 OTSG Shell-to-Tube Differential Temperature Limit

Maintain OTSG differential temperature less than 70F°. If this limit is approached, then:

1. Reduce the cooldown rate in half.
2. Continue steaming on the affected OTSG.
3. Supply MFW thru the startup control valve at about .05 x 10^6 lbm/hr (if MFW is not being used).

If the differential temperature approaches 100F°, stop the cooldown and maintain RCS temperature constant. Remove decay heat by steaming the OTSG(s) with the high differential temperature. Resume the cooldown when the differential temperature drops below 70F°.

4.2.10 Cooling Mode When Both OTSG's are Unavailable for RCS Heat Removal

Use HPI "feed and bleed" to cool the RCS when both OTSG's are unavailable. Open the Pilot Operated Relief Valve (PORV), RC-RV-2, to provide a cooling water flow path to the Reactor Building Sump.

4.2.11 Guideline Flow Chart

Appendix C includes a flow chart and explanatory text showing the logic path of the tube rupture guidelines.

5.0 SIMULATOR TRAINING EXPERIENCE

5.1 Introduction

Most of the guidelines proposed in this TDR were incorporated into a lesson plan for the annual requalification training of the TMI-1 licensed operators at the B&W simulator. A draft revision to TMI-1's EP 1202-5, incorporating the guidelines, was also prepared.

These documents were then used to inform the licensed operators of the changes contemplated for EP 1202-5, and to demonstrate the combined effects these changes would have. During the classroom session, each guideline was described and the reasoning behind it was explained. During the simulator session, their combined effect was illustrated by running a large tube leak scenario twice.

For the first simulator run, the then-existing revision of EP 1202-5 was used to deal with the leak. For the second run, the draft version was employed. It became apparent that the new guidelines made plant control easier.

5.2 Results

Of all the guidelines proposed in this TDR, the two changes most useful (and obvious) to the operators are these:

1. Reactor Coolant Pump (RCP) trip as a followup to low subcooling margin (SCM) rather than following automatic HPI from a low RCS pressure ESAS signal;

2. HPI throttling when SCM requirements are met and pressurizer level is back on scale, rather than waiting for pressurizer level to reach 100".

Another useful (but less obvious) change is the RCP restart criterion based on regaining SCM rather than various combinations of primary and secondary pressures. This and Item 1 above may be considered under the same general heading of increased RCP availability.

The exercise of the draft EP 1202-5 was useful in critiquing the contemplated changes. Simulator experience also showed that it is not possible to raise OTSG level to 95% with full HPI on, while steaming the OTSG and maintaining a 100°F/hr cooldown. The difficulty was created by the steaming of the OTSG's in this situation. HPI and throttled EFW flow can provide a plant cooldown at near 100°F/hr if the OTSG's are not steamed.

During the simulator session, B&W revised the simulator to allow leakage of more than 2 tubes and to allow leakage in both OTSG's.

5.3 Comments

This material was presented to two of seven groups by Tech. Functions personnel. The remaining five groups received it from B&W training personnel who taught the material using the same lesson outline. B&W did not endorse the material. Comments from trainees indicate that the training was of dubious value. It will be necessary to repeat the training for all personnel.

6.0 CONCLUSIONS AND RECOMMENDATIONS

The ability of the plant to handle beyond design basis events can be substantially increased and the RCS leakage can be reduced for design basis leaks with the adoption of the following changes/additions to tube rupture procedures.

1. Reduce minimum subcooling margin to 25°F
2. Replace the existing RCP trip criteria with trip on loss of subcooling.
3. Adopt the steam generator isolation and pressure/level control guidelines of this guideline.
4. Provide the RCP NPSH limits of Figure 6 for use during emergency conditions.
5. Waive fuel pin-in-compression limits.
6. Control plant cooldown to limit the tube/shell delta T to 70°F.
7. Revise procedure entry point conditions to be leakage greater than 50 gpm.
8. Incorporate criteria for initiation of feed and bleed cooling into the tube rupture procedure.
9. Adopt criteria for opening TBV's/ADV's if RCS pressure stays above 1000 psig during feed and bleed cooling.
10. HPI throttling should be allowed when subcooling is regained and pressurizer level is on scale.

It is further recommended that these changes be implemented prior to restart of TMI Unit 1.

7.0 REFERENCES

1. U.S. Nuclear Regulatory Commission. NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R. E. Ginna Nuclear Power Plant. NUREG-0909. U.S. NRC.

2. U.S. Nuclear Regulatory Commission. Safety Evaluation Report Related to the Restart of R. E. Ginna Nuclear Power Plant. Docket No. 50-244. NUREG-0916. May 1982. U.S. NRC.

3. C. Y. Cheng. Steam Generator Tube Experience. NUREG-0886. U.S. NRC.

4. Rochester Gas & Electric Corp. "Procedure E-1.4. S/G Tube Rupture." June 23, 1982. Ginna Station.

5. "Procedure No. E1.0. Safety Injection Initiation." February 23, 1982. Prarie Island Nuclear Generating Station.

6. "Procedures E-1.3 and E-1.4 S/G Tube Rupture." June 3, 1982. Prarie Island Nuclear Generating Station.

7. Duke Power Co. "EP/0/A/1800/17. Steam Generator Tube Rupture." Oconee Nuclear Station.

8. INPO. "SOER 82-16 Draft." January 4, 1983.

9. INPO. "Analysis of Steam Generator Tube Ruptures at Oconee and Ginna" INPO 82-030. November 1983.

10. Babcock & Wilcox "Abnormal Transient Operating Guidelines. Three Mile Island Nuclear Station - Unit One." June 1981.

11. GPU Nuclear Corp. "Acceptability of Intentionally Loading Main Steam Lines During an OTSG Tube Rupture Event." Document No. 1101X-5320-A18. July 30, 1982 - Engineering Mechanics section.

12. L. C. Lanese "TMI-1 OTSG Tube Rupture Basis for Design Leak Rate." August 31, 1982. SAPC/118.

13. M. Sanford "NPSH Requirements for Piggyback Safety Injection Operation - TMI-1." December 27, 1982. MSS-82-584.

14. J. Veenstra. "Fuel Pin Compression Limits During an SGTR Event." TMI-83-009. January 20, 1983.

15. S. D. Leshnoff. "Mechanical Integrity Analysis of TMI-1 OTSG Unplugged Tubes." TDR 388. GPUNC. March 9, 1983.

16. P. S. Walsh. "TDR 400 Draft Guidelines for Plant Operation with Steam Generator Tube Leakage." PA-893. February 15, 1983.

17. Darrell G. Eisenhut, to H. D. Hukill. March 4, 1983. Docket No. 50-289. March 4, 1983. U.S. Nuclear Regulatory Commission.

18. N. K. Savani, Dunn, B. M., Jones, R. C., Response to GPU letter dated August 28, 1980 (TMI-1/E 1203). Document Identifier 12-1121718-00. September 26, 1980. Babcock & Wilcox Co., Lynchburg, Virginia.

19. H. D. Hukill (GPUN) to D. G. Eisenhut (NRC). "RCP Trip on 25°F Subcooling Margin." March 31, 1983. 5211-83-017

20. G. J. Sadauskas. "TMI-1 Saturation Margin Monitor Loop Error Analysis." Calc. No. 1101X-322B-009 Rev. 1. February 28, 1983. GPUNC.

21. G. L. Lehmann. "OTSG Leakage and Operating Limits." TDR 417. GPUNC. March 1983.

22. TMI Unit 1. "Plant Limits and Precautions. OP 1101-1. Rev. 14." February 26, 1982. GPUNC.

23. "Determination of Minimum Required Tube Wall Thickness for 177-FA Once Through Steam Generators." BAW-10146. October 1980.

24. J. Veenstra to D. G. Slear. "Thermal Shock Clarification For Small Break Operating Guidelines." September 9, 1982. TMI-82-071.

25. M.H. Kostrey, Memo to L.C. Lanese "TMI-1 Reactor Coolant Pumps NPSH limits Westinghouse Pumps" MC-1752, March 30, 1983.

26. GPUN Calculation # 1101x-5450-013 "Minimum Subcooling Margin for RCP Operation." by L.C. Pwu April 15, 1983.

APPENDIX A

TMI-1 SGTR PROCEDURE GUIDELINE ANALYSIS

COMPARISON OF GUIDELINES AND EP 1202-5 REV. 16 TO
THE REQUIREMENTS OF VARIOUS SOURCE DOCUMENTS

A.0 SCOPE

The purpose of this Appendix is to compare the guidelines of this TDR to the current revision (16) of EP 1202-5, OTSG Tube Leak/Rupture, and guidelines, requirements, commitments or recommendations from various sources. The sources reviewed were the TMI-1 Anticipated Transient Operating Guidelines (draft of 15 May 1981; hereinafter referred to as ATOG), "Clarification of TMI Action Plan Requirements" (referred to as NUREG 0737), and the Safety Evaluation Report related to restart of Ginna (Ref. 2) (referred to as NUREG 0916) and the INPO draft Significant Operating Event Report of 04 January 1983 concerning steam generator tube leaks (referred to as SOER).

With respect to NUREG 0737 and 0916, only those requirements or commitments directly related to a tube leak emergency procedure were considered. With respect to ATOG, only the followup guidance for tube leaks was considered, and then only if it differed from the guidance in the latest approved tube rupture procedure (EP 1202-5 Rev. 16). With respect to the SOER, only the recommendations related to procedures were considered.

The results of this comparison work are summarized in Table A-1.

TABLE A-1
Comparison of TDR 406 Guidelines to Other Sources

Requirement	Source				Addressed By		Comments
	ATOG	0737	0916	SOER	1202-5	406	
Run RCP's with low RCS pressure	X		X	X		X	
RCP restart	X		X	X	X	X	No specific guidance provided for RCP restart with a "solid" pressurizer.
Subcooling margin	X			X	X	X	
HPI throttling	X			X	X	X	
Steam line flooding	X			X	X	X	ATOG does not recognize TMI-1 capability to flood steam lines without damage.
Cooldown of damaged OTSG			X	X		X	Guidelines provide for continued steaming of affected OTSG for cooling except when OTSG isolation is required.
Specify entry threshold	X			X	X	X	Symptoms.
Method for plant cooldown following SGTR	X		X		X	X	
Plant cooldown following SGTR with stuck open SG safety valve	X				X	X	ATOG refers to excessive heat transfer section EP 1202-5 and guidelines provide means for minimizing probability of lifting a SG relief valve.

TABLE A-1
(continued)

Requirement	Source: ATOG	Source: 0737	Source: 0916	Source: SOER	Addressed By: 1202-5	Addressed By: 406	Comments
Affected SG pressure control	X				X	X	
OTSG tube to shell differential temperature	X					X	
Criteria for using ADV's in preference to main condenser			X				Guidelines presume that steaming to condenser is always preferable to steaming to atmosphere.
Consider multiple tube ruptures		X				X	Guidelines prepared in consideration of these two cases.
Consider tube leaks in both OTSG's		X				X	
HPI on inadequate SCM	X				X		Not specifically stated in TDR 406, but it is an implicit requirement of the HPI throttling criteria.
Consider excessive primary to secondary heat transfer	X				X		Overfeeding considered by EP 1202-5.
Consider loss of offsite power	X					X	Guidelines make no particular distinction between offsite power available or unavailable, but they do provide guidance if the equipment disabled by LOOP is unavailable.

TABLE A-1
(continued)

Requirement	Source: ATOG	Source: 0737	Source: 0916	Source: SOER	Addressed By: 1202-5	Addressed By: 406	Comments
OTSG level control	X				X	X	
Primary pressure control without pressurizer spray	X					X	
Isolation of affected OTSG	X				X	X	

A.1 Interpretation

A.1.1 "Requirement" Column

These are paraphrased descriptions of guidelines, requirements, commitments, or recommendations from source documents.

A.1.2 "Source" Columns

These columns define the origin of the requirement considered in this comparison.

A.1.3 "Addressed By" Columns

These columns define the document which answers the requirements. If a mark appears in the 1202-5 column without a corresponding mark in the TDR 406 column, it means that the guidance in EP 1202-5, Rev. 16 should be retained in the revision that incorporates the guidelines of Section 4 of this TDR. If a mark appears in both columns, it generally means that the guidelines in this TDR supercede the guidance in EP 1202-5, Rev. 16. If a mark appears in the TDR 406 column alone, it denotes ew guideline to be incorporated into the revised EP 1202-5.

A.1.4 Comments

This column provides additional information if necessary.

APPENDIX B

PROCEDURE CHANGE SAFETY EVALUATIONS

B.0 PROCEDURE CHANGE SAFETY EVALUATIONS

The purpose of this Appendix is to address the safety implications of the key changes required to implement the tube rupture guidelines described in this TDR.

1. RCP trip on loss of subcooling margin (SCM)
2. Change in SCM
3. Tube/shell delta T of 70°F during emergencies
4. Revised RCS NPSH curve
5. Relaxation of fuel pin in compression limits
6. OTSG isolation criteria
7. RCP restart criteria
8. HPI throttling at 0" instead of 100 inches.
9. Leaving ADV open when no OTSG heat sinks and RCS is above 1000 psig.

Each of these items is addressed below:

B.1. RCP Trip on Loss of Subcooling Margin

In a letter dated March 4, 1983 to H. D. Hukill (Rev 17), the NRC superceded the actions required in IE Bulletins 79-05C and 79-06C. The staff has instead concluded that "the need for RCP trip following a transient or accident should be determined by each application on a case-by-case basis considering the Owner's Group input." For several years, the B&W Owner's Group has supported the concept of RCP trip on loss of subcooling margin. In Reference 19, GPUNC informed the NRC of their reasons for revising the trip criterion to loss of subcooling margin. The safety evaluation for this change has been transmitted separately to the TMI plant staff for review and approval.

B.2. Change in Subcooling Margin

GPUNC has evaluated the instrument error (Ref. 20) associated with the subcooling margin monitor and alarm. Under normal containment conditions, the loop error is 5°F. Under the temperature and radiation environment of a small break LOCA, this error is no worse than - 18.76°F. The basis for the original SCM was 5°F real subcooling plus 45°F string inaccuracy. Therefore safety margins are not decreased by this change.

A complete safety evaluation has been prepared and transmitted to the site separately for review and approval.

B.3. Change in Tube to Shell Delta T

The existing emergency limit for tube/shell delta T at TMI-1 is 100°F. The limit is being revised to reduce tensile stress on leaking OTSG tubes. Previously the limit of 100°F was based on stresses to intact tubes. Indeed, the limit has been increased to 150°F by Babcock & Wilcox; and, it is valid in the absence of degraded OTSG tubes.

The more restrictive 70°F limit increases plant safety limits by reducing the likelihood of propogating a crack. This analytical work is documented in Reference 15. This change can be made under the provisions of 10 CFR 50.59 because it does not affect technical specifications. Since tube stresses are reduced, plant safety limits are increased and the two additional criteria of 10 CFR 50.59 are also met. Namely, there are no new accidents introduced into the plant that have not been previously analyzed. Since tube to shell delta T has not been explicitly addressed in the FSAR, existing plant safety margins have not been decreased. In fact, plant safety margins have been increased since the allowable delta T has been decreased.

B.4. RCP NPSH Limits

Reduced NPSH limits bring the pump closer to a point of cavitation. However, NPSH requirements have been reduced for lower temperatures as determined by the pump manufacturer Westinghouse in Reference 20. Margins have been modified based on safety margins identified by the pump manufacturer, therefore, the probability of pump cavitation has not been increased and plant safety margins are protected. Neither are technical specifications affected. The operation of reactor coolant pumps at low RCS pressures does not introduce any new accident or transient other than those already analyzed in the FSAR. Pump operation is allowed at these lower RCS pressures but at a higher subcooling margin. Since real plant subcooling margin is still being maintained as discussed in item 2, there is no reduction in plant safety. Operation of the reactor coolant pumps increases plant safety margins with respect to thermal shock, increased DNB ratios, and improved capabilities for degassing the reactor coolant system under tube rupture conditions when minimum subcooling margins are being maintained.

B.5. Fuel Pin and Compression Limits

As addressed in Reference 14 B&W has recommended that fuel pin in compression, limits be waived during certain plant transient conditions including steam generator tube rupture events. Fuel pin in compression limits have been established in order to maintain cladding integrity. Waiver of these limits does not reduce pin integrity although reanalysis by B&W may be required when fuel compression limits have been waived. Since cladding integrity will have to be addressed each time these limits are violated, the demonstration of acceptable clad integrity will be made. No new accidents or transients will be introduced then have been previously analyzed in the FSAR. Similarly, plant safety margins will not be reduced, namely, cladding integrity will not be challenged.

B.6. OTSG Isolation Criteria

The existing steam generator tube rupture procedure, EP-1202-5, allows the operator to isolate the affected steam generator anytime RCS pressure is below 1000 psi. The revised criteria would allow steaming of the OTSG until BWST level is 21 ft. or radiation limits approach site emergency limits. Steaming of the OTSG introduces the potential for increasing offsite radiation doses; however, these limits will be maintained within the requirements of 10 CFR Part 20. It should be noted that the isolation of the steam generator on high radiation is keyed towards maintaining Part 20 limits. Steaming of the generator when possible increases the chances of preventing major offsite releases since flooding of an OTSG can result in liquid relief out of the steam safety valves with the possibility of safety valve failure. The value of BWST level at 21 ft. is sufficient to assure a source of water for the ECCS pumps. The value of 21 ft. allows sufficient inventory to flood both steam lines and allow the plant to be placed on feed and bleed cooling in the recirculation mode from the RB building sump (Ref. 12, 13). It should be further noted that the doses associated with a steam generator tube rupture were increased when the requirement for maintaining subcooling margin was introduced into the plant procedures following the TMI-2 accident. At that time the issue was addressed in writing to the NRC staff (Ref. 21) justification for the change was that Part 20 limits were being maintained. This criterion is still being maintained with the change in OTSG isolation criteria.

These changes can be made under 10CFR50.59 because safety margins are not decreased. Technical Specifications are not affected by this change. No new accidents or transients are introduced which have not been previously analyzed since this guidance is intended to deal with events which are beyond the design basis of the plant (i.e., tube rupture without condenser and RCP's).

B.7 RCP Restart Criteria

The RCP Restart Criterion assures that the pumps are not restarted until the core is adequately subcooled. (Note that there are other RCP restart unrelated to this criterion). Reference 19, and Sections B.1 and B.2 demonstrate that the core is adequately subcooled with RCP's running and a 25°F subcooling margin. No Technical Specifications are affected. No new accidents or transients are introduced into the plant; no safety margins are decreased and no accident consequences are increased. Allowing an earlier pump restart gives the operator greater control over the plant since forced flow is preferable to natural circulation cooling. This change can therefore be made under the provisions of 10CFR50.59.

B.8 HPI Throttling at 0 inches Indicated Level

The safety aspects of throttling HPI on 25F° subcooling margin are addressed in section B.2. Core coolability is not dependent on the pressurizer level at which HPI is throttled i.e., core cooling is only

dependent on an indication that the core coolant is subcooled. The basis for requiring pressurizer level is so that the existing pressurizer heaters are covered with water so that they can be energized. Energizing the heaters before they are covered causes them to burn out. On the other hand, there is no need to refill the pressurizer to the 100 inch level at full HPI flow. In fact, this flow rate is undesireable for two reasons. Rapid filling of the pressurizer causes an RCS pressurization during conditions when pressurizer sprays are unavailable. Insurges to the pressurizer compress the steam space. Pressurize pressure must be reduced either by sprays (if available) or pressurizer venting (vent line or PORV). Controlling the HPI flow minimizes the insurge rate, and hence, the pressurization. This reduced pressurization provides more margin to the 100F° subcooling curve thereby minimizing challenges to the thermal shock/brittle fracture limit.

This change does not represent an unreviewed safety question because:

1. No change to the Technical Specifications is required

2. No new accidents are introduced to the plant (the operator is still required to cover the pressurizer heaters before energizing them), and

3. The consequences of previously analyzed accidents/transients is not increased. It is less likely that the operator will violate the 100F° subcooling margin. Core coolability is not dependent on established pressurizer level, but only an adequate subcooling margin.

B.9 <u>ADV's Open when RCS is above 1000 psig with no OTSG Heat Sinks</u>

This TDR provides guidance for certain situations well beyond the design basis. One such situation is the case where the plant is one feed and bleed cooling, but RCS pressure is above 1000 psig. This condition can result in liquid relief out of the OTSG safety valves. Opening the ADV's is the preferred course of action because it minimizes the chance of an uncontrolled blowdown though the OTSG safety valves. This condition is well beyond the plant design basis. Plant Tech Specs are not affected by this procedural step. Therefore the change can be made under the provisions of 10CFR50.59.

Beyond the consideration of whether this change can be made under the provisions, of 10CFR50.59, it is believed that opening the TBV's/ADV's is prudent and reduces the risk of an uncontrolled release to the environment.

APPENDIX C

GUIDELINES FLOW CHART

C.0 GUIDELINES FLOW CHART

The flow chart in this section shows the major milestones and decision points on the path from operation at full power through the development of an OTSG tube leak/rupture to inspection and repair of the damage. The flowchart is not meant to be an exhaustive treatment of all actions required to reach cold shutdown, rather it is the framework upon which a procedure can be constructed.

C.1 INTERPRETATION

Diamond boxes are decision points. The path taken out of a diamond depends on the answer to the question posed in the diamond. Boxes enclosed by a single line represent steps that take seconds or minutes to execute. Boxes enclosed by double lines represent tasks that may require minutes to hours to accomplish. For the sake of simplicity, certain steps that will be required in the procedure have been omitted (e.g., confirming reactor trip, or making radiation surveys of the secondary plant).

The decision points immediately following a double-line box are meant to force the operator into a "thought-loop" so that if conditions change, the operator may select an alternate, more appropriate cooldown path. For instance, while cooling down on forced flow with a tube leak in excess of 50 gpm, the operator should continually inquire as to whether the Reactor Coolant System pressure and temperature are within the capability of the Decay Heat Removal System. If so, when the operator should obviously change the RCS heat removal mode from steaming the OTSG's to using the DHRS. If not, then the operator should continue to ask whether the RCS conditions are suitable for forced flow cooling via OTSG's, i.e., is subcooling inadequate, are the OTSG's available/OK for use, are the RC pumps available. If the answers to these questions always no, yes, and yes, then continued forced flow cooldown is acceptable. If any of the answers change, then the thought flow breaks out of the loop and presents the operator with new criteria for selecting an alternate cooldown mode.

This "thought loop" philosophy should be incorporated into the procedure revision.

C.2 PROCEDURAL OBJECTIVE

The objective of the tube leak procedure is to expeditiously cool down and depressurize the plant so as to minimize primary to secondary leakge and thus, it is hoped, offsite doses. The process involves recognition of the event, shutting down the plant, and cooling down the plant to the point where the Decay Heat Removal System can remove core heat.

C.3 ENTRY POINT

The procedure will be entered when a primary to secondary leak is encountered that requires the plant to be shut down. The symptoms of a tube leak requiring shutdown are described in TDR 400 (Ref. 16).

C.4 PLANT SHUTDOWN

The rate of plant shutdown from 100% power will be determined in part by the magnitude of the RCS depressurization due to the leak. If the leak is small (the Makeup System is able to keep up withit), then the plant can be shutdown at a rate commensurate with equipment capabilities and, to a certain extent, the leak rate. When the reactor and turbine are off line, the plant is ready to enter the cooldown phase.

However, if the leak results in RCS depressurization to the trip setpoint, the reactor and turbine will be off line immediately. The ensuing transient will have to be dealt with and the plant status will have to be evaluated prior to the cooldown.

C.4.1 Preparation for Cooldown

If the shutdown transient results in a loss of subcooling margin, HPI must be immediately actuated and the Reactor Coolant Pumps (RCP's) must be immediately tripped. The OTSG's must then be evaluated for suitability as heat sinks for the RCS.

If the shutdown transient does not result in a loss of subcooling margin, the OTSG's must still be evaluated for suitability as RCS heat sinks.

If neither OTSG can be used because of high offsite doses or low BWST level, then the cooldown will proceed directly using the HPI "feed and bleed" method.

For the balance of the discussion in this section, assume that HPI "feed and bleed" is unnecessary.

If the RCP's are off, Emergency Feedwater flow to the OTSG's must be confirmed. The ICS will automatically control OTSG level at 50% on the Operating Range if the RCP's are off. If subcooling margin is <25F, the operator must manually raise the level to 95% to promote two-phase natural circulation in the RCS.

Since a forced circulation cooldown is the most preferred mode, the RCS conditions should be evaluated for RCP restart. If subcooling margin is regained and the RCP NPSH limits are met, 2 RCP's should be restarted. If the pumps cannot be restarted, the cooldown must proceed by natural circulation.

C.5 PLANT COOLDOWN

During the cooldown, RCS conditions must be continuously evaluated to ensure that the cooldown mode is appropriate and to determine whether conditions are suitable for the Decay Heat Removal System.

Regardless of cooldown mode, the following items, may be encountered while cooling down.

C.5.1 HPI Throttling

The existing HPI throttling criteria are unchanged with are exception: HPI may be throttled when subcooling is regained and pressurizer level comes on scale.

C.5.2 OTSG Steaming

The affected OTSG may be steamed for RCS heat removal purposes, but it must be steamed to avoid lifting the Main Steam safety valves, prevent premature Steam line flooding, keep OTSG pressure less than RCS pressure, and control OTSG tube to shell differential temperature.

C.5.3 OTSG Tube to Shell Differential Temperature

It is necessary to minimize tube to shell differential temperature to minimize tensile stresses on the OTSG tubes. As noted above, steaming is are way to accomplish this; another is to decrease the cooldown rate; a third is to use main Feedwater to cool the lower downcomer.

C.5.4 OTSG Pressure Control When RCS Pressure is Greater Than 1000 psig

During a natural circulation cooldown or an HPI feed and bleed cooldown, RCS pressure may stay high. Emergency Feedwater can be used to quench the steam space; if the OTSG is flooded, inventory can be relieved via the Turbine Bypass Valves or the Atmospheric Dump Valves.

C.5.5 Cooldown Rate

The cooldown rate should be limited to less than 1.6 F/hr to avoid reactor vessel brittle fracture concerns. It may not always be possible to observe this limit due to the effects HPI cooling and the occassional necessity to steam the damaged OTSG.

C.6 EXIT POINT

The operators exit the procedure when the RCS heat sink becomes the Decay Heat Removal System.

APPENDIX D

SIMPLIFIED EVENT TREE

D.0 SIMPLIFIED EVENT TREE

The event tree on the following page shows possible combinations of circumstances that were considered that resulted in the guidelines presented in this TDR.

The guidelines explicitly stated in section 4, when incorporated into a revised OTSG Tube Leak/Rupture Emergency Procedure, will enhance the capability of TMI-1 to deal with an OTSG tube leak. The purpose of this section is to describe the features of the revised procedure. The discussion which follows assumes that the logic presented by the flowchart depicted in Appendix D is adopted for the revised procedure.

APPENDIX E

PROCESS COMPUTER OUTPUT

E.0 PROCESS COMPUTER OUTPUT AND ALARMS

E.1 Scope

The process computer will have the following information available with alarms as noted:

Subcooling margin
OTSG Tube to Shell Differential Temperature

E.1.1 Subcooling Margin Alarm

Subcooling margin will be computed for each hot leg and the average of the five highest incore thermocouples. The process computer should trigger an alarm state if:

SCM < 25°F

E.1.2 OTSG Tube to Shell Differential Temperature

Calculate shell temperature as follows for each OTSG if all shell thermocouples are operable:

$$T_{shell} = 0.242\ T_1 + 0.176\ T_2 + 0.201\ T_3 + 0.143\ T_4 + 0.238\ T_5$$

Tables E.1.4.1 and E.1.4.2 define acceptable substitutes for various failed thermocouples and combinations thereof.

Limiting the alarm state to conditions when T_{cold} is < 535 inhibits the alarm during normal operations.

Table E.1.4.1 Shell Thermocouple Substitution

Failed T/C	Substitute T/C
T_5	T_4
T_4	T_5
T_3	T_2
T_2	$0.5\ (T_1 + T_3)$
T_1	T_2
T_4 & T_5	No Calc
T_3 & T_2	T_1
T_3 & T_1	T_2
T_2 & T_1	T_3
T_1 & T_2 & T_3	No Calc

Wide Range T_{cold} should be used in determining OTSG tube to shell differential temperatures. Normally, use the wide range input from TE-1-5A&B and TE-3-5A&B, although TE 959 and TE 961 can be used in certain cases. Table E.1.4.2 defines the data sources.

For each Loop, Calculate Tube to Shell ΔT as Follows:

$\Delta T_{T-S} = T_{shell} - T_{cold}$

ΔT_{T-S} should trigger an alarm state if

$T_{cold} < 535°F$ <u>and</u> $\Delta T_{T-S} > 70°F$

Table E.1.4.2 Wide Range T_{cold} Input

RC-P-1 A	B	C	D	T_{cold} A Loop	B Loop
0	0	0	0	Avg A	Avg B
0	0	0	X	Avg A	TE 4 5B
0	0	X	0	Avg A	TE 2 5B
0	0	X	X	Avg A	Avg B
0	X	0	0	TE 4 5A	Avg B
0	X	0	X	TE 4 5A	TE 4 5B
0	X	X	0	TE 4 5A	TE 2 5B
0	X	X	X	TE 4 5A	Avg B
X	0	0	0	TE 2 5A	Avg B
X	0	0	X	TE 2 5A	TE 4 5B
X	0	X	0	TE 2 5A	TE 2 5B
X	0	X	X	TE 2 5A	Avg B
X	X	0	0	Avg A	Avg B
X	X	0	X	Avg A	TE 4 5B
X	X	X	0	Avg A	TE 2 5B
X	X	X	X	Avg A	Avg B

0 = Pump Running

X = Pump Off

Avg A = (TE 4 5A + TE 2 5A)/2

Avg B = (TE 4 5A + TE 2 5A)/2

TE 959 May be substituted for TE 2 5A

TE 961 May be substituted for TE 4 5B

FIGURE C-1

SINGLE & MULTIPLE TUBE RUPTURE GUIDELINES

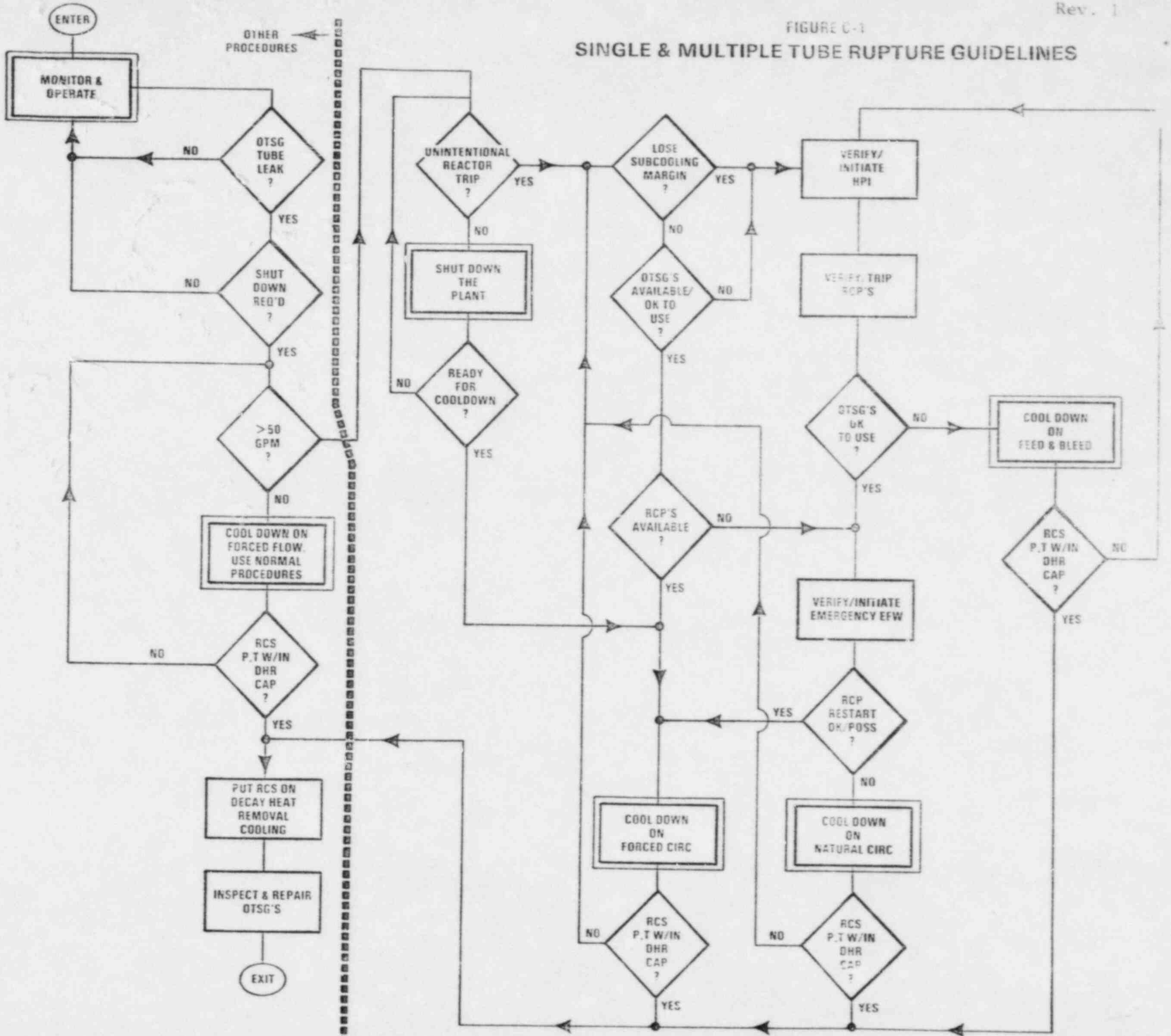


FIGURE 1

STEAM GENERATOR TUBE RUPTURE GUIDELINE DEVELOPMENT-ACTIVITY NETWORK

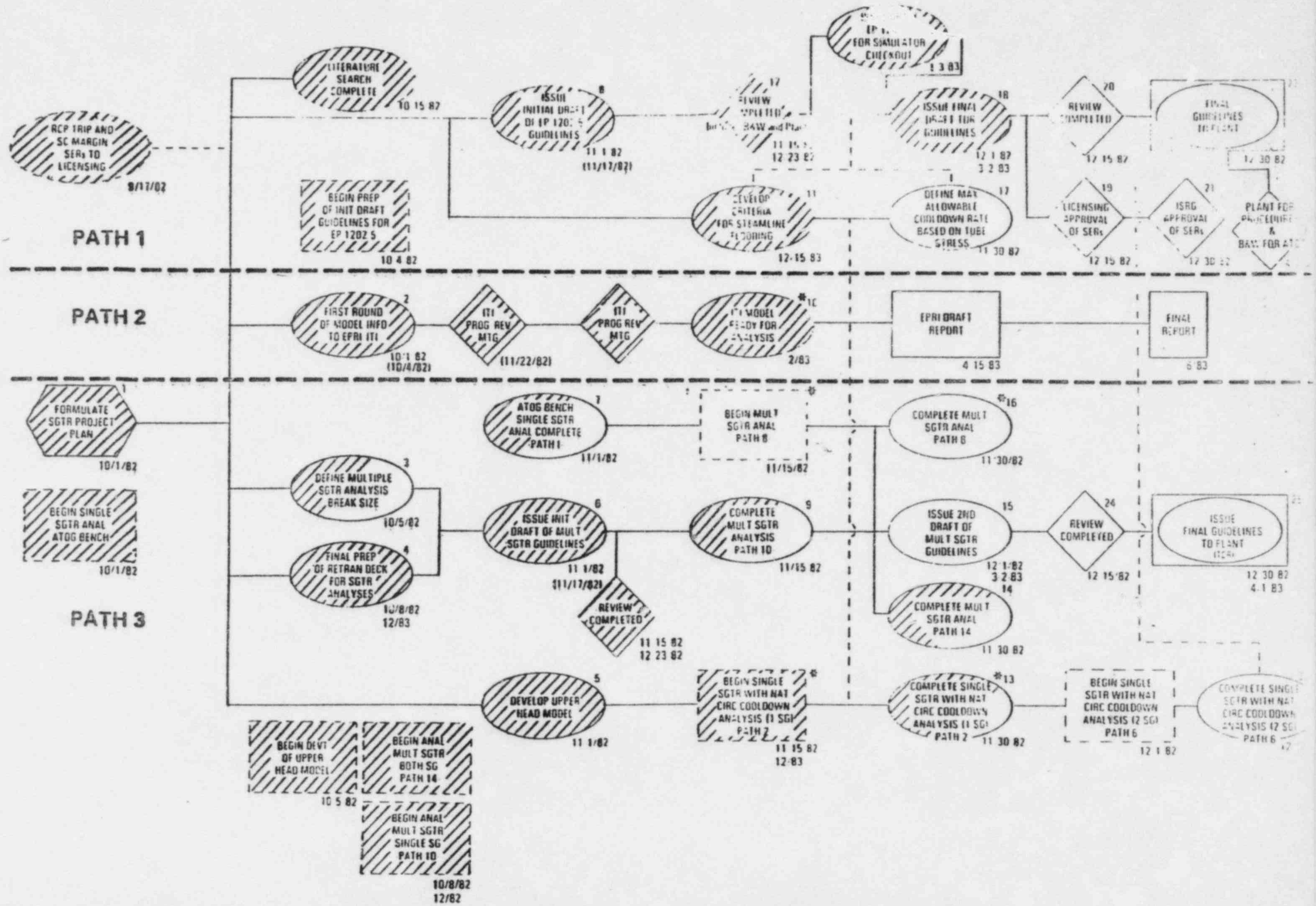


FIGURE 2

Break Flow for Single Ruptured Tube

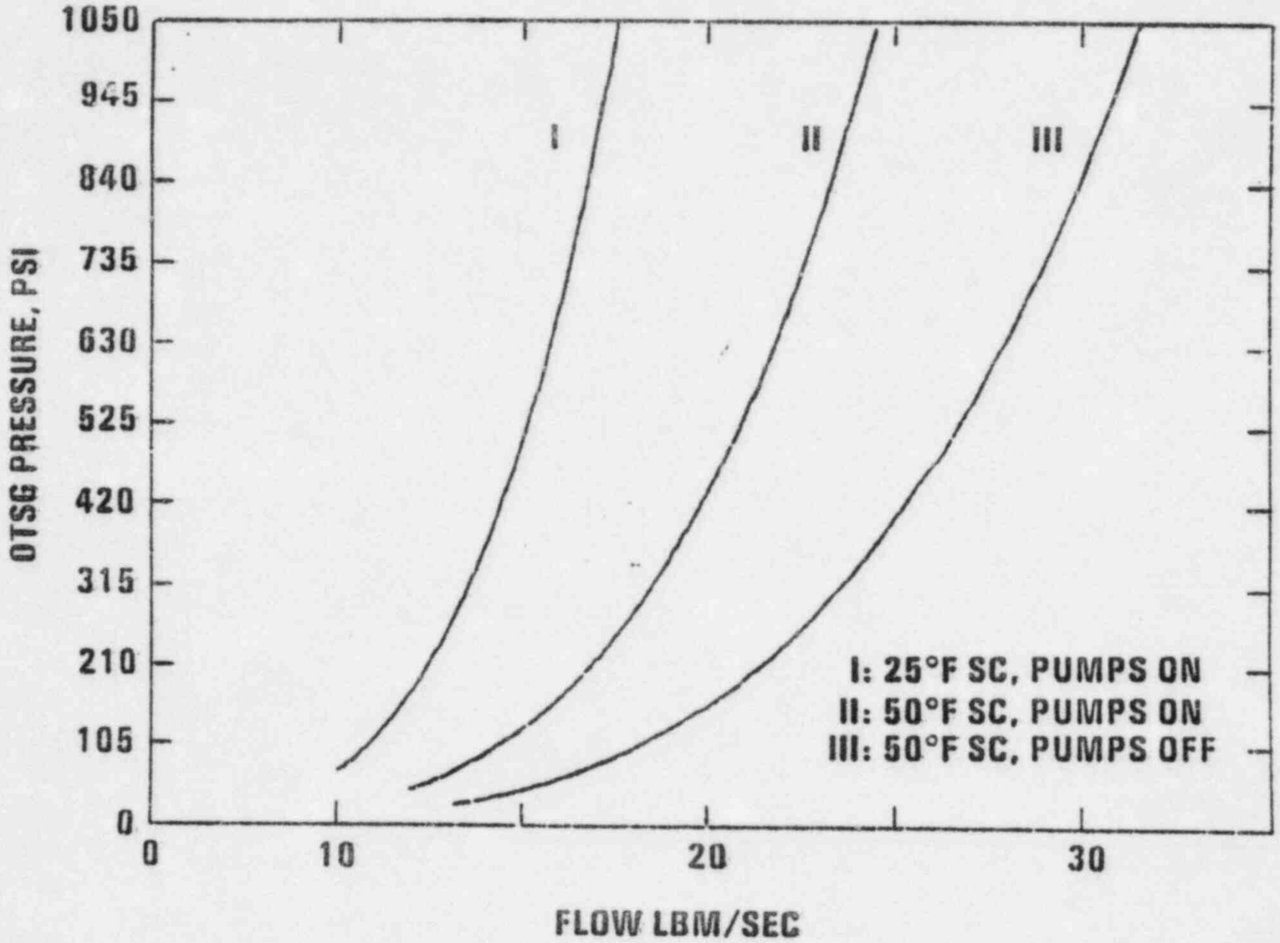


FIGURE 3

TDR # 406
Rev 1

Effect of RC Pump Operation on Integrated System Leakage for Single Ruptured Tube

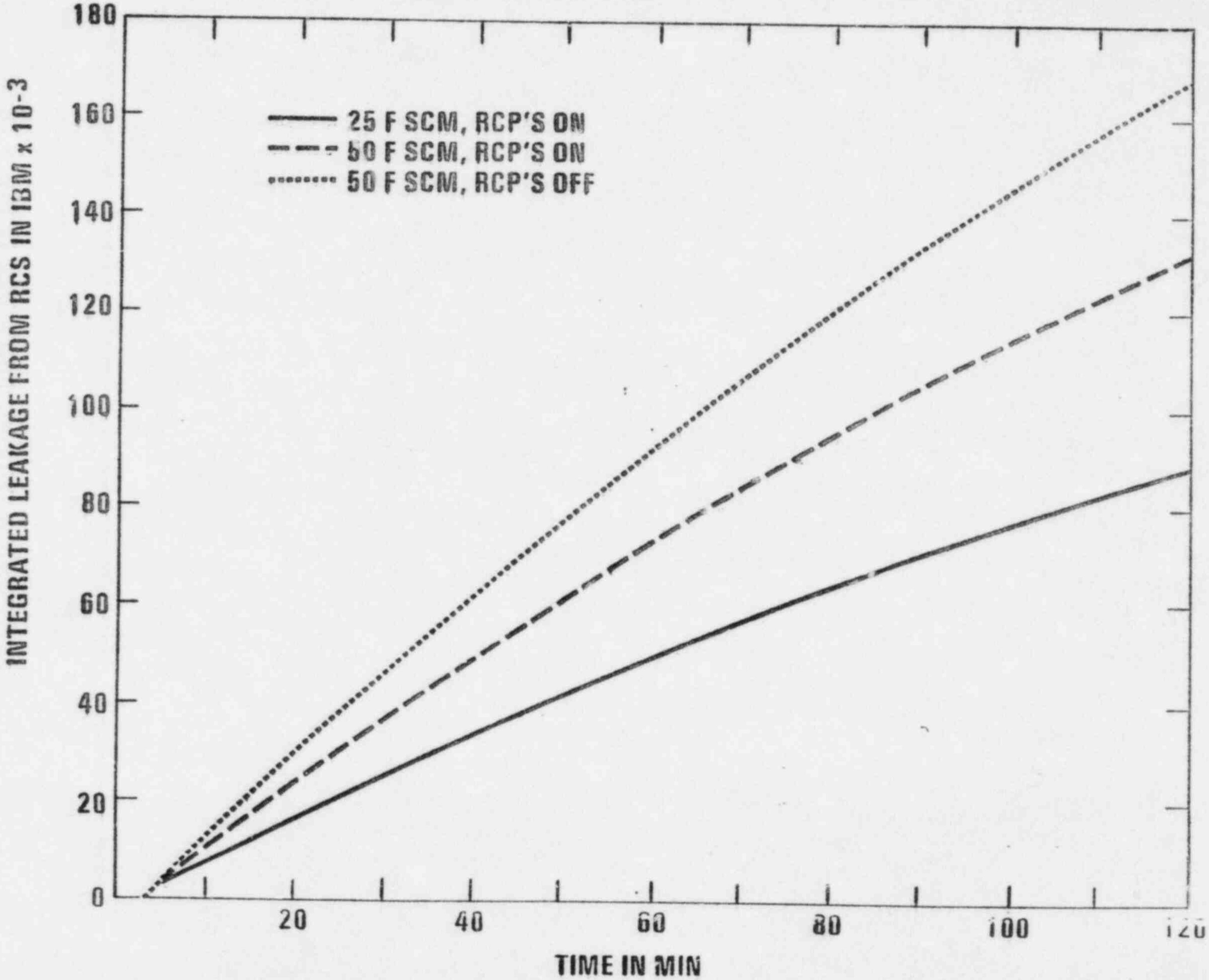
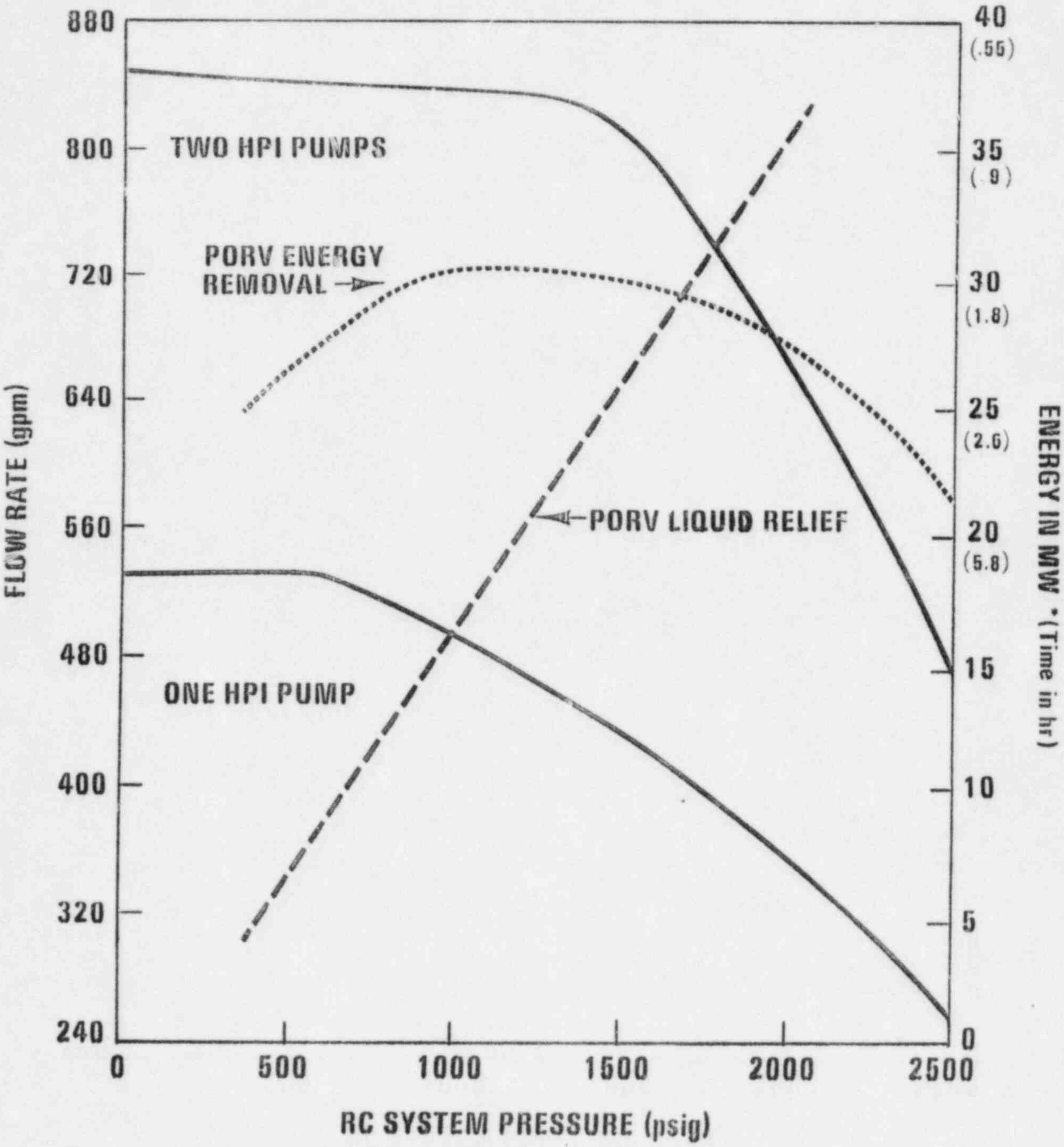


FIGURE 4

Mass and Energy Capabilities of HPI and PORV



*1.0 ANS
2535 Mw (t)

FIGURE 5

Time Behavior of Subcooling Margin for a Spectrum of Ruptured Tubes

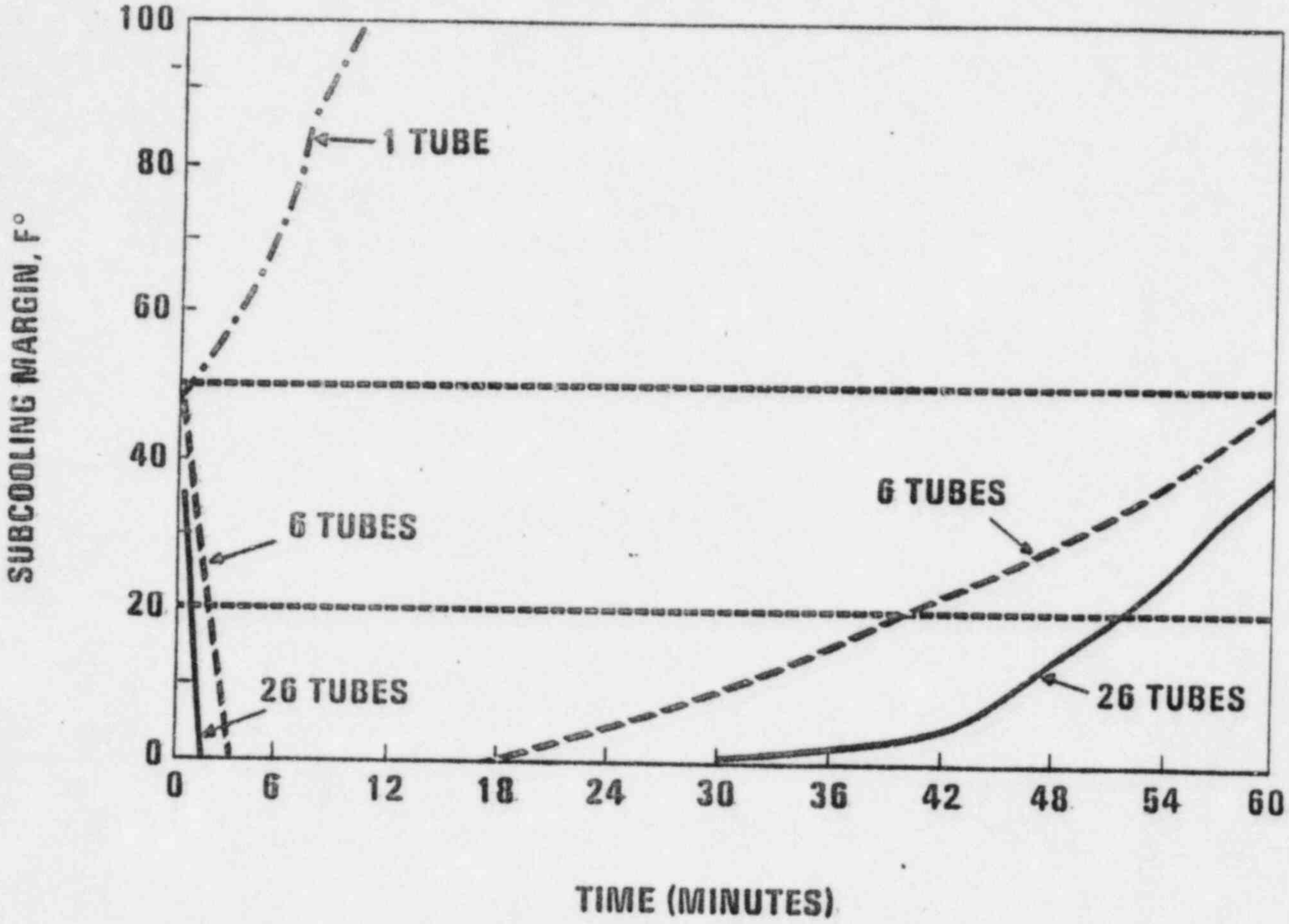


FIGURE 6

RCP NPSH Curves

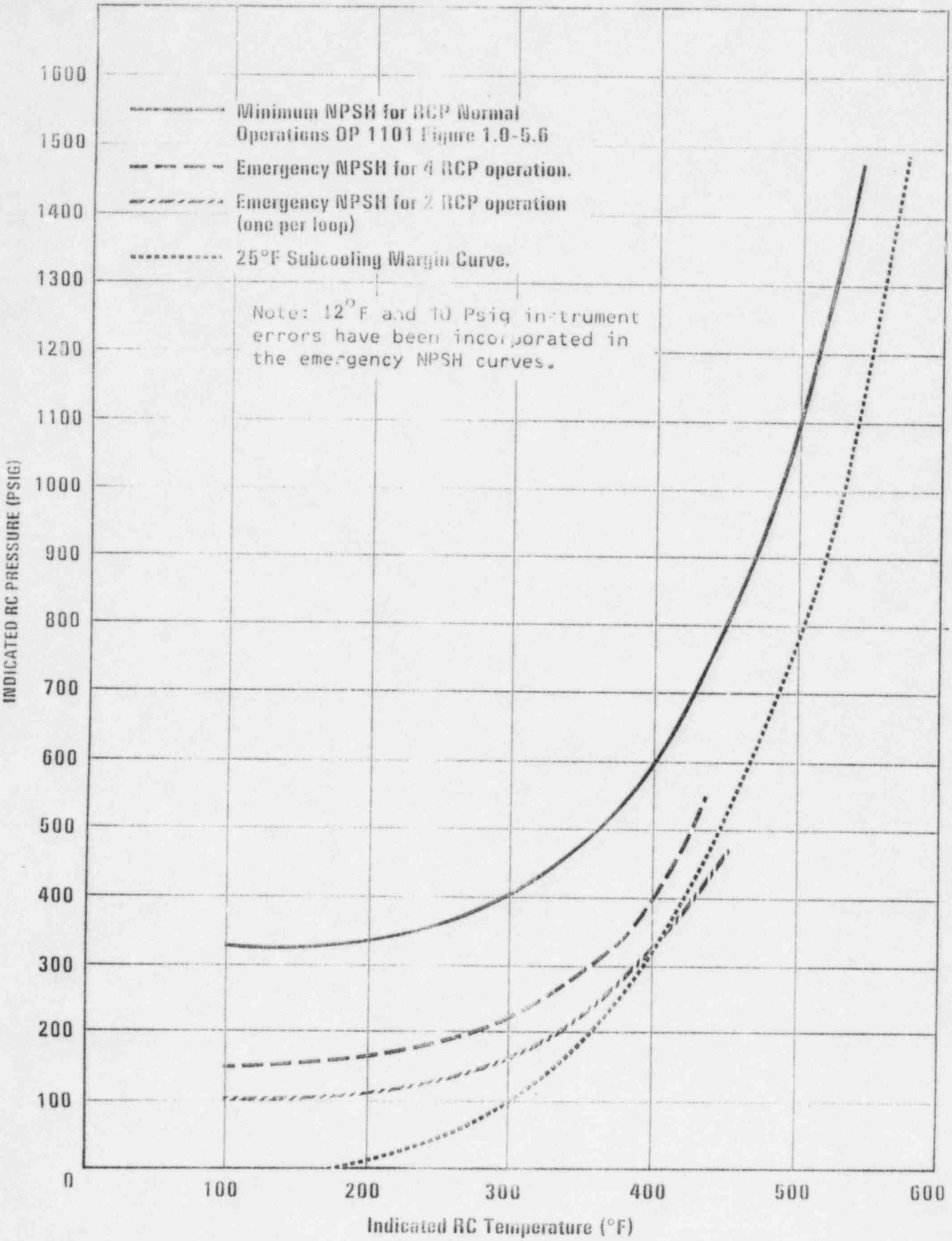


FIGURE D-1

TDR # 406
Rev 1

SIMPLIFIED OTSG EVENT TREE