GPU Nuclear TECHNICAL DATA REPORT	TDR NO. 406	REVISION NO. 3
	BUDGET ACTIVITY NO. 128006	PAGE 1 OF 81
PROJECT: TMI-1	DEPARTMENT/SECTION Safety Anal/Plant Cont	
	RELEASE DATE 3/20/83	REVISION DATE 12.2.83

DOCUMENT TITLE:
SG Tube Rupture Procedure Guidelines

ORIGINATOR SIGNATURE	DATE	APPROVAL(S) SIGNATURE	DATE
[signature]	12.2.83	[signature]	12/2/83
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		APPROVAL FOR EXTERNAL DISTRIBUTION	DATE
		[signature]	12.2.83

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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° during tube rupture events.

Revision 2 to this TDR included the following recommendations for procedural revisions, some of which have already been incorporated in EP 1202-5.

1. Isolate the OTSG's on a measured or projected dose rate of 50 mRem/hr whole body or 250 mRem/hr thyroid dose.

2. Stop the non-ES HPI pump if the RCS is cooling more than 100F/hr.

3. Priorities should be spelled out in EP-1202-5:
   a. Minimizing SCM has a priority over minimizing cooldown time.
   b. Keeping OTSG level below 95% is less important than control of the RCS cooldown rate.

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ABSTRACT (Cont'd)

* 4. Initiate the DHR system at 300F under tube rupture conditions.

* 5. Trip the reactor if pressurizer level cannot be
¢ maintained above 150 inches with two HPI pumps on.

* 6. Raise the unaffected OTSG level to 95% before raising the affected OTSG level to 95% unless incore temperatures are not decreasing and there is no OTSG heat transfer.

* 7. If RCPs are not tripped within two minutes of a loss of SCM, maintain 1 RCP in each loop running.

¢ Revision 3 of this TDR provides guidelines for possible reduction
¢ of offsite doses under specific plant conditions. These considerations
¢ address deviations from OTSG steaming and isolation criteria which can
¢ be evaluated as part of the long term response to a tube rupture. It
¢ also includes some additional guidance on the use of shell thermo-
¢ couples and address conflicts in requirements for RCP NPSH and DHR
¢ maximum pressure.

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* There are a total of 81 pages in this report including Figures, Tables and Appendices.

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REV	SUMMARY OF CHANGE	APPROVAL	DATE
3	Figure 6, Table 1 and Section 2.1.6 and 2.1.9 were revised to address simultaneous DHR and RCP operation, clarify #1 seal staging requirements and specify that NPSH curves are dependent on the number of operable pumps per loop.	LCL	12-2-83
3	Section 2.3.2 on issue resolution was updated. Additional work items have been noted.	LCL	12-2-83
3	Section 3.2 was updated to agree with the guidelines of Section 5.	LCL	12-2-83
3	Appendix E was amended to provide guidance in calculating shell temperatures if the weighted average is not available from the computer; or if only thermocouple voltages are available in the control room.	LCL	12-2-83
3	Appendix F was added to address long term dose reduction considerations. Under certain plant conditions variance from the OTSG steaming isolation criteria will allow a reduction in dose. This TDR recommends that those guides be appended to the tube rupture procedure.	LCL	12-2-83

REV	SUMMARY OF CHANGE	APPROVAL	DATE
2	Section 2.2.3. Expanded discussions on feed and bleed cooling to include ADV and TBV mass and energy relief capabilities relative to decay heat and leak rate.	/s/	5/12/83
2	Section 2.3. Discussed additional work which will be addressed in a future revision to TDR 406.	/s/	8/12/83
2	Section 4.2.2. Added note regarding actions to be taken if RCPs are not tripped within 2 minutes of a loss of SCM.	/s/	8/12/83
2	Section 4.2.3. Added note regarding loss of SCM after RCPs are restarted.	/s/	8/12/83
2	Section 4.2.6. Revised guidance on raising OTSG levels to 95%.	/s/	8/12/83
2	Section 4.2.7. Added isolation criterion on 250 mRem/hr.	/s/	8/12/83
2	Section 4.2.11. Added criterion for Core Flood Tanks Isolation.	/s/	8/12/83
2	Section 6.0. Added recommendations to isolate CFTs using criterion provided and initiation of DHR system at 300F.	/s/	8/12/83
2	Section 2.2.1. Clarified that emergency NPSH curves should be followed for both RC pump trip and restart.	/s/	5/12/83
2	Sections 3.2.4 and 4.2.3. Start one RCP per loop or both RCP's in the same loop.	/s/	5/12/83
2	Section 3.2.2. Revised to be consistent with the steaming criterion in Section 2.1.3.	/s/	8/12/83

REV	SUMMARY OF CHANGE	APPROVAL	DATE
2	Added detail to Table of Contents and reversed the order of Sections 4.0 and 5.0.	/s/	8/12/83
2	Added Tables 1 & 2 which provide tabular data on RCP NPSH requirements and on spray flow for various RCP pump combinations.	/s/	5/12/83
2	Section 2.1.3. Added recommendation for OTSG isolation if iodine release rate exceeds 250 mRem/hr or whole body dose rates 50 mRem/hr, correcting error in previous revision.	/s/	8/12/83
2	Section 2.1.6 and Figure 6. Revised emergency NPSH limits to account for calculated instrument errors during LOCA conditions.	/s/	5/12/83
2	Sections 2.1.8 and 4.2.2. Added discussion on the experience gained from the June 1983 Lynchburg simulator sessions.	/s/	8/12/83
2	Section 2.1.9. Added recommendation to allow DHR system initiation at 300F instead of 275F.	/s/	8/12/83
2	Section 2.1.10. Recommendation to trip reactor if 200 inch pressurizer level cannot be maintained with two HPI pumps running.	/s/	8/12/83
2	Section 2.2.2. Clarified guidance on isolation criterion with leaks in both OTSGs.	/s/	8/12/83
2	Section 2.2.2. Addressed raising OTSG level to 95% without causing an overcooling.	/s/	8/12/83
2	Section 2.2.2. Discussed why EFW should not be used to control OTSG pressure in an isolated OTSG (changing previous recommendation of Rev. 1).	/s/	8/12/83

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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.	/s/	5/8/83
1	Revised cover page to show shell-to-tube delta T can be controlled below 100F.	/s/	5/8/83
1	Added a List of Tables pp i & iii	/s/	5/8/83
1	Included use of MFW as means to cool OTSG shell. p 5	/s/	5/8/83
1	Indicated that continuous steaming of OTSG is simplest means of meeting OTSG level, pressure and differential temp. considerations pp. 5,6,10,18	/s/	5/8/83
1	Eliminated reference to RAC for determining when to isolate OTSG based on radiological conditions. p 6	/s/	5/8/83
1	Added Section 2.1.3.1 to discuss control when both OTSG's are isolated. (Also p 10).	/s/	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.	/s/	5/8/83
1	Revised explanation of ADV & TBV flow capability relative to OTSG flooding (incorrect in Rev 0) pp 11,16	/s/	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	/s/	5/8/83
1	Section 4.2.3 revised to account for inability to start either RCP in the A loop.	/s/	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.	/s/	5/8/83
1	Simplified Section 4.2.8 on cooldown rate.	/s/	5/8/83
1	Added Section 4.2.9 on controlling OTSG shell-to tube differential temperature.	/s/	5/8/83

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

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.

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2.0 TECH FUNCTIONS SGTR GUIDELINES DEVELOPMENT PROGRAM

Figure 1 shows the execution of the steam generator tube rupture guideline 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. Develop emergency RCP NPSH limits.

7. Redefine entry point conditions.

* 8. Factor in experience from use of the guidelines on the B&W simulator.

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

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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 in 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. Shell-to-Tube 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 100F°. However, before tube/shell delta T exceeded 100F°, 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 Limiting OTSG Tube Stresses

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 60F° during heatup and to 100F° during cooldown with one OTSG isolated. This value of 100F° assumed that tubes had no more than 40% through-wall cracks. In reference 23, B&W established 142F° for a cooldown using both OTSG. The 70F° value in this TDR is proposed as a guide in determining an acceptable cooldown rate. If delta T can be maintained at or below 70F°, the operator has optimized the plant cooldown rate. The 70F° limit more conservatively assumes that tubes in the OTSG are leaking below a detectable limit. A 70F° value limits propagation of these cracks.

\+ Control of shell to tube 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, Isolation and 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.

+ 4. Cooldown at low pressure/temperature can be accomplished more quickly, allowing DH system operation sooner.

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.

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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. The existing site emergency limits are 50 mRem/hr whole body and 250 mRem/hr thyroid dose measure or projected at the site boundary (Ref. 33). Section 2.1.9 discusses the length of time required to cool the plant down to DHR system conditions. This length of time defines the integrated does allowed by the guidelines (i.e. release rate for the specified period of time).

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 or offsite release criteria is reached. If both OTSG's are steamed, then 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.

If the BWST level reaches 21 feet, then both OTSGs must be isolated. If the dose criteria is reached, one OTSG should be isolated and the doses reevaluated. If the dose criteria still cannot be met, then the second OTSG should be isolated.

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 50F° and 25F° subcooling. Even at the maximum cooldown of 100F°/hr, the integrated leakage differs by a factor of two.

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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 425F° 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 100F° of
+ subcooling. However, general centrifugal pump test data have shown
+ that NPSH requirements are substantially reduced at water tempera-
+ tures above 250F°. 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' manu-
+ facturer (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
* margin monitor 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 (actually,
¢ either one pump or one per loop) are shown in Table 1 and Figure 6.
+ Note that the curves should be used as if the two loops of the RCS are
¢ independent, i.e. either one or two RCPs are operating. Also shown
¢ is the 4 pump operation (actually two pumps in one loop or four pumps)
+ NPSH curve which has considered the changed flow distribution in the
+ coolant loops. In addition, the normal NPSH curve and the 25F° sub-
+ cooling curve are shown for comparison purposes. RCP operation below
¢ 300 psig is only allowed if pressure differential across the No. 1
¢ pump seal is maintained as 275 psi or greater (Ref. 32).

+ 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. Pump
* limits and precautions must be adhered to while following the
* emergency NPSH limits (e.g., the pump should be tripped on high
* vibration).

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)

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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.

2.1.8 * Simulator Experience

* Steam generator tube rupture procedures were exercised during the January and June 1983 simulator sessions. The experience gained from these two sessions has been factored into this TDR. The principal lessons learned were that:

* 1. Controlling plant cooldown rate with 2 or 3 HPI pumps running is very difficult at best. Raising OTSG level to 95% during this plant condition may not be possible.

* 2. Prioritization of plant control parameters was not obvious to the operator in certain situations. The two situations which were encountered were:

* a. Minimizing subcooling margin has priority over minimizing the cooldown time, and;
* b. Steaming to control OTSG level is less important than RCS cooldown rate.

* 3. Plant response after RCPs are restarted was unexpected to the operation. Second pump restarts may be required before subcooling margin stays above 25F°.

* 4. Criteria for isolating core flood is required. Core flood tanks should be isolated in a subcooled system before they initiate.

* 5. Additional guidance is required if the RCPs are not tripped within two minutes of a loss of subcooling margin.

* These items are discussed in more detail in Section 4.0.

2.1.9 * Emergency Limits for Decay Heat System Initiation

* Plant experience indicates that a large portion of time during cooldown occurs below the temperature of 350F. Simple analyses, assuming only one ADV for a loss of offsite power, using the CSMP computer code (Ref 31) indicate that the RCS can be cooled down below 300F in less than 4 hours, and cooldown to 275F can be accomplished in about ten hours. 275F is the normal DHRS initiation temperature. GPUNC has evaluated the capability of the DHR system to operate at a temperature of 300F (Ref 32) and concluded that it is within the design capabilities of the system. Therefore we recommend that the tube rupture procedure allow initiation of the DHR system at 300F instead of 275F.

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¢ As shown on Figure 6, the DHR system cannot be operated simultaneously
¢ with reactor coolant pumps above a certain temperature. Therefore, a
¢ decision will be required to evaluate the desirability of either: (1)
¢ steaming the OTSGs but maintaining forced RCS; or (2) stopping
¢ steaming but tripping the RCPs with the resultant potential for
¢ steam formation in the hot legs. This decision will be dependent
¢ on the amount of radioaactivity release at that time. It is
¢ technically acceptable to trip RCPs if the result is termination
¢ of significant releases.

2.1.10 * Reactor Trip on Low Pressurizer Level

* Preventing a loss of subcooling margin has many advantages in
* contolling the plant. Before the spring of 1983, EP 1202-5 required
* the plant to be tripped if level could not be maintained above 100
* inches with two HPI pumps running. This may not be a sufficient
* level to prevent voiding of the pressurizer after a reactor trip.
* Emptying of the pressurizer causes a loss of subcooling margin and
* the subsequent tripping of the RC pumps. In order to prevent this
* situation, the reactor should be tripped if level cannot be
* maintained above 150 inches or higher. This is sufficient volume
* (about 600 cubic feet) to prevent pressurizer voiding.

* There is a disadvantage to this recommendation since the safety
* valves will lift after the reactor trips. However, this situation is
* considered acceptable when weighed against the plant control
* advantages of having RCP's running. Also, only a certain window
* of break sizes will result in reaching 150 inches and not 100
* inches with full HPI flow. Outside of this window, both levels would
* be reached.

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.

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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^2) 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 advantage of maintaining RCP's is 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. Reducing
* the allowable subccoling margin is not intended to reduce RCP
* equipment protection. RCP's should be tripped if emergency NPSH
* requirements are not met, and should not be started until NPSH
* requirements are re-established. If applicable NPSH pump vibration
* limits are exceeded, then the RCP's should be tripped. Pumps should
* be restarted as indicated in the TMI-1 Small Break LOCA
* Procedure (EP 1202-6B, Attachment 1). As noted in Section 2.3,
* bumping criterion requires additional clarification.

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 guidelines for OTSG steaming are nearly the same when either one or both OTSGs are affected. 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 accommodated.

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. If level is not raised to 95%, then EFW flow must be at a high enough flow rate to penetrate the tube bundle sufficiently to provide adequate heat transfer. A flow rate of 450 gpm total (225 gpm/OTSG) has been verified as acceptable by B&W (Ref 29). This flow is the minimum available after a seismic event and worst case single failure, coincident with a small break LOCA in which boiler condenser cooling is required. It is important to recognize that with two HPI's available, boiler condenser cooling is not required. Procedures should therefore allow the operator to raise OTSG level to 95% tempered with the need to control the RCS cooldown rate. During tube rupture events with both HPI pumps available, , the unaffected OTSG level should be raised first while the affected OTSG level should be prevented from boiling dry (maintain a minimum level of 30"). The operator can control 1 OTSG instead of trying to raise level in both OTSG's simultaneously. For the case with only one HPI pump, if incore temperatures are not decreasing and the OTSG is not removing RCS heat, then there will not be a cooldown rate control consideration; moreover, the plant may be in a condition that requires boiler condenser cooling. Therefore, OTSG levels must be raised to the 95% level simultaneously in this situation (Ref. 30).

Section 2.1.3.1 discusses steam generator isolation, 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

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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 within two hours of
* reactor trip assuming that there is no energy relief out of the
* ruptured tube (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 RCS pressure above 1000 psig and neither OTSG available requires the
* opening of the TBV or ADV's to control OTSG pressure below 1000 psig
* and level below the upper tube sheet. Either the ADV or TBV have
* sufficient steam capacity at high OTSG pressure to remove decay
* heat. The TBV's also have sufficient capacity to prevent OTSG
* flooding. However, if the leak rate is large enough, the steaming
* rate required to control level in the OTSG may result in an
* unacceptable RCS cooldown rate. In this case, cooldown rate must be
* controlled and the OTSG allowed to flood. As discussed in Section
* 4.2.2.1, this situation does not seem likely (at least at high
* decay heat levels). As decay heat decreases, steaming can be
* terminated when RCS pressure goes below 1000 psig and is controlled
* by the PORV and HPI.

* 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 until an
* OTSG heat sink can be restored or until the plant can be put on decay
* heat removal.

* Until OTSG level is above the upper tube sheet, pressure in the OTSG
* will remain below 1000 psig, since the RCS temperature is below
* 540F°. With level less than 600 inches, however, the operator still
* must steam to keep pressure below 1000 psig; therefore he should not
* have to steam to control level on the affected OTSG. When level goes
* above 600 inches, pressure in the OTSG is determined by the steam
* pressure in the steam line. If the lines are leak tight, then
* compression of the steam bubble can cause a pressure increase above
* 1000 psig. In this case, the operator would steam the OTSG to reduce
* pressure. However, if there are steam leaks in the system (e.g.,
* through steam traps.) then the lines could fill with water before
* OTSG pressure increased. Therefore to prevent this situation the
* OTSG's must be steamed to preclude this possibility.

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 instituted. 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°/hr cooldown without having to steam either OTSG.

2.3 Additional Work Requirements

2.3.1 * Analyses

As noted in Section 1.0, there is a program of ongoing computer analysis work simulating single and multiple steam generator tube ruptures. The effort includes the plant states listed in Sections 3.1.1 and 3.1.2. This list does not reflect two specfic detailed analysis efforts which are being undertaken as part of the tube rupture quantitative development effort. The first analysis is a simulation of the vessel head region during natural circulation cooldown. This analysis effort will help in evaluating the effect of a vessel head bubble on the RC pressure response. It will also aid in evaluating the benefit of the reactor vessel head vent.

The second analysis effort is being performed in conjunction with Babcock and Wilcox Company. Detailed analyses are being performed to provide improved guidelines for OTSG filling after a loss of subcooling margin, with one, two and three HPI pumps available. The intent of the analyses is to assure that the OTSG's are filled without violating cooldown or shell to tube differential temperature limits, while still meeting core coolability requirements. This effort was considered after the January 1983 simulator training session and further defined after the June 1983 simulator session.

2.3.2 * Issue Resolution

A number of issues were identified which require further effort to resolve. These are the following items:

A. Operator guidance for identifying two phase natural circulation cooling (boiler-condenser). B&W provided this guidance to the B&W Owners Group (Reference 37) and the information is presently being evaluated by GPUN.

B. Acceptability of excessive cooldown rates for very short time intervals. B&W has provided their response to this concern (Reference 39). Deviations of no more than 15F are allowed at any given time.

C. Importance and technical basis of fuel-in-compression limits. This effort is still underway.

D. Viability of DHR system initiation at temperatures above 300°F. This effort is still on-going.

E. Identification of pump vibration limits for various pump combinations. This effort still requires resolution.

F. Determine viability of ATOG RCP "Pump Bump" criterion. The bumping criterion would allow running pumps without adequate subcooling or NPSH margin as long as a steam generator heat sink is available. Determine whether the ATOG criterion anticipates that NPSH will be reestablished since the heat sink is available. The existing bumping criterion in TMI-1 emergency does not require that a heat sink be established and would allow continuous RCP operation in violation of NPSH limits.

ATOG requires that RCP protective limits be observed for all conditions except under certain inadequate core cooling situations. Therefore, the operator would only operate the RCPs if normal limits allowed.

G. As noted in Section 2.1.9, simultaneous DHR and RCP operations are not allowed above 275F. The basis for the DHR pressure limit and for the RCP seal staging pressure must be re-evaluated for emergency conditions.

H. Reactor coolant pump shaft vibration data were taken during the plant cooldown of 10/6/83 (Reference 38). The RCS was depressurized at a constant temperature of 400°F from a subcooling margin of 180F° to 60F°. Shaft vibration increased to 26 mils. A plant test will be developed and executed to determine if the pump response at this temperature is typical or anomalous. If typical, then shaft vibration will have to be monitored by the operator when normal NPSH limits are violated.

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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 * Assumed Plant Conditions

The following assumptions apply to the development of guidelines as they apply single tube leak/ruptures.

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.

In addition the revised guidelines 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.

* The consideration of items 1 and 2 are equivalent to an assumption of
* loss of offsite power.

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.

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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 the tube rupture guidelines (with a written discussion of those guidelines). This section of the report provides descriptive text of the guidelines shown in that diagram and described in Section 5.0. 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, reducing primary to secondary leakage by minimizing primary to secondary differential pressure; third, minimize stresses on the OTSG tubes by limiting shell/tube delta T; and finally, minimizing releases by allowing the leaking OTSG to flood if offsite doses are large enough (approaching levels at which a Site Emergency would be declared).

The following sections discuss the most important control considerations recommended by the guidelines.

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 operator used to be allowed (before 1983) to let the OTSG fill anytime that RCS pressure was below 1000 psig. The guideline has the operator steam the OTSG: first, to prevent lifting of the OTSG safety valves; second, to prevent the generator from filling; third, to maintain OTSG level on scale, fourth, to promote natural circulation; and finally, to control shell/tube differential temperature. Since meeting these criteria will mean nearly constant steaming, the guidelines require the operator to steam the OTSG unless specific isolation criteria are met.

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3.2.2.3 Shell-to-Tube Delta T

Plant limits and precautions require maintaining the OTSG tube temperature within 100F° of the shell temperature. Detailed analyses (Ref. 36) show that a shell to tube delta T of 70F° acceptably limits stresses and minimizes the chances of increasing the leak from a pre-existing through wall crack.

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

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. Tripping of RCPs on loss of subcooling assures that the core remains covered for all LOCA conditions. At the same time, the chance of pump trips during tube ruptures is reduced.

3.2.4 Followup Actions for Loss of Subcooling

The second 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 one reactor coolant pump per loop (if possible), and return to the section of the procedure for forced flow cooldown. If one pump can not be started per loop, then both RCP's in one loop are started. This will maximize the pressurizer spray flow for the given RCP availability.

Even if the OTSG must be isolated, steaming is still required to keep OTSG pressure below 1000 psig. Pressure control is required to prevent the MS safety valves from lifting.

If subcooling is regained in the RCS, then HPI is throttled, RCP's are started and the operator continues the cooldown. The desired RCP configuration is to start one pump in each loop. If the operator is unable to start an RCP in each loop then he should start both RCP's in one loop.

The reasons for restarting RCP's are similar to the reasons for not tripping them on low RCS pressure. The RC pump flow may cause voids in the system to collapse, dropping pressurizer level. The guidelines require the operator to wait 2 minutes for SCM to recover to prevent incessant "pump bumping".

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If subcooling cannot be restored, the operator cools the plant down
6 on natural circulation unless the OTSG heat sink is lost (e.g. 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) after about
+ 2 hours. 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
6 from challenge. The operator controls OTSG pressure below 1000 psig
* with whatever means are available (turbine bypass, or Atmospheric
* Dump Valves.). When the OTSG tube region is filled with water, 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.

4.0 SIMULATOR TRAINING EXPERIENCE *

4.1 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.

* As indicated in Section 4.2.2, the January 1983 training cycle was not
* effective in training operators on the basic concepts for treating
* tube ruptures. The June training cycle was successful in
* communicating concepts.

4.2 Results *

4.2.1 January 1983 Training *

Of all the guidelines proposed in this TDR at that time, the two changes most useful (and obvious) to the operators were:

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 100F°/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.

4.2.1.1 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.

4.2.2 June 1983 Training *

* A number of items were identified during the B&W operator training
* simulator sessions held from June 6 to June 29, 1983. The experience
* gained from using a revised tube rupture procedure EP 1202-5. These
* items will be discussed below.

4.2.2.1 Control of RCS Cooldown Rate

* Section 5.2.1 discussed the difficulties in controlling
* cooldown rate while raising OTSG level to 95%. At the time,
* it was the author's belief that steaming of the OTSG's was
* causing the excessive cooldown rate. However, further
* discussion with B&W (Ref 29) indicated a different
* explanation. The B&W HPI model calculates flow by
* iteratively solving two equations of the form:

* $$P_d = 2840 - \left(\frac{W}{N}\right)^2 K \qquad (1)$$

and

* $$W = \left[(P_d - P_{RCS}) A\right]^{1/2} \qquad (2)$$

* where:

* P_d = pump discharge pressure
* P_{RCS} = RCS pressure
* W = flow, lbm/sec
* N = number of HPI pumps running
* A = coefficient to account for number of HPI valves open.

* The HPI flow is overpredicted for TMI-1 with three HPI pumps
* running and/or low RCS pressure. The difference results from the
* physical arrangement at TMI-1, in which two pumps discharge into
* a common header. The cavitating venturies at Unit 1 also reduce
* the maximum flow of the HPI pumps compared to the valve
* calculated by the simulator.

* As a result of this understanding, subsequent simulator drills
* were run with only two HPI pumps available and control of
* cooldown rate was improved.

* The HPI initiation rule has been reemphasized to the operator, namely that "full" HPI means the full flow from two HPI pumps. It is acceptable to secure the third HPI pump when the RCS is saturated, and the OTSG heat sink is available or if cooldown rate is 100F°/hr. or more.

* A second consideration in controlling cooldown rate was in raising the OTSG level increase to 95% after a loss of subcooling margin. Operations believes that it is an unnecessary burden on the operator to control cooldown rate while raising level on both OTSG's simultaneously. Therefore, the leaking generator level will not be raised until the unaffected OTSG has been raised to 95% unless incore thermocouple temperatures are not decreasing and the OTSGs are removing decay heat. The 95% level is important in establishing boiler-condensor cooling during small LOCA's in which only one HPI is available. However, the RCS cooldown is only a concern if both HPI pumps are running. Therefore the two concerns are mutually exclusive. Operator training and EP 1202-5 have been revised to have the operator control one OTSG level at a time as long as incore temperatures are decreasing. If RCS temperatures are not affected by the secondary side cooldown (i.e., no secondary side heat removal) then both OTSG's should be raised to the 95% level simultaneously.

* During the simulator session of June 11, 1983 the operators were faced with a large (about 1400 gpm) tube rupture. They attempted to control OTSG level below 95% on the affected generator. However, the cooldown rate was too high even with the unaffected OTSG isolated. The simulator response to the event was partially responsible, but the procedure also needed to be more explicit. The simulator leak model currently uses the orifice equation to predict leak flow (Ref. 28). This model would initially overpredict the break flow and would account for the very rapid flooding of the OTSG's compared to results of the RETRAN and RELAP5 computer codes. RETRAN and RELAP5 (Ref. 35) analyses to date do not predict such a response. Nevertheless, the operator needs to recognize that cooldown rate control takes precedence over OTSG level control and EP 1202-5 was subsequently revised to prevent this conflict in plant control requirements.

4.2.2.2 Plant Stabilization Before Cooldown *

* The procedure used during the training session had the operator initiate plant cooldown and then establish minimum subcooling margin. The simulator sessions showed that the RCS could not be depressurized fast enough to maintain a minimum subcooling margin. Therefore, training material was revised to emphasize the need to stabilize the plant and reach the minimum allowable subcooling margin. Plant cooldown should then be initiated. The procedure was modified so that all four RCP's can be left on until 500F instead of 540F. Based on the TMI-1 ATOG (Ref. 28), this change provides a difference of about 47% in the spray flow (see Table 2). Thus pressurizer spray flow is maximized for as long as possible. Finally, the

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* operator is given the option of using the pressurizer vent if he is still unable to reduce pressure sufficiently to maintain a minimum subcooling margin.

* 4.2.2.3 RCP Restart Criteria

Section 5.2.1 observed that the RCP restart criteria on 25F subcooling margin (SCM) was very useful to the operator. Several areas required clarification or expansion, however. First, RCP restart should not be attempted unless pump emergency NPSH limits are met. The only exception is that NPSH requirements are waived, with pump restart allowed during certain inadequate core cooling conditions as specified in plant procedures. Pump "bumps," however, should be attempted even if NPSH requirements are not met. Second, loss of subcooling may occur after the RCP's are restarted. Collapse of steam voids in the RCS may cause voiding of the pressurizer, resulting in a loss of SCM. RCS temperatures may be hotter in the tube region than in the core if natural circulation has been lost. Mixing of this hotter water with the cooler core water will cause a decrease in the SCM. Several pump starts may be required before subcooling margin stabilizes above 25F°. Allowing two minutes of RCS flow is helpful in eliminating both steam voids and temperature gradients so that successive restarts will be successful.

* 4.2.2.4 Core Flood Tank Isolation

In several simulator runs, the operations were faced with tube rupture or small break LOCA conditions in which the RCS was subcooled, but core flood tanks initiated, providing cooling water which was not required, since SCM was maintained. Most significant was that CFT initiation delayed RCS depressurization. Neither the LOCA nor tube rupture procedure provides any guidance about isolation of the core flood tanks. Therefore, this TDR has been revised to provide guidance about when to isolate the core flood tanks (see Section 5.2.11).

* 4.2.2.5 RCP Trip Criterion

Questions arose regarding the actions to be taken if RCPs were not tripped within 2 minutes of a loss subcooling margin. Clarification was provided using the guidance of the TMI-1 ATOG (Ref. 28) which requires the operator to keep the RCPs in each loop running if the two minute time limit is exceeded. If RCPs are subsequently tripped, the RCS may be voided enough to uncover the core. RCPs should be run for at least 7000 seconds to assure that the core will not uncover (based on Appendix K assumptions). If pump damage may occur, then one pump in each loop should be tripped. If either of the running pumps fails, the tripped pump in that loop should be started. For simplicity, the guidelines in this TDR call for 1 RCP to be run in each loop. This provides sufficient flow to cool the core (Ref. 28).

+ Rev. 1
* Rev. 2

5.0 TUBE LEAK/RUPTURE GUIDELINES *

5.1 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).

5.2 Guidelines & Limits *

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

5.2.1 Subcooling Margin Requirements *

Control Reactor Coolant System (RCS) subcooling margin (SCM) between
+ 25F° and 50F°. Maintain SCM as close to 25F° 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.

5.2.2 Reactor Coolant Pump Trip Criterion *

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

Note: * If RCP's are not tripped within 2 minutes of loss of 25F° SCM, then
* run 1 RCP in each loop.

5.2.3 Reactor Coolant Pump Restart Criteria *

When the required subcooling margin (25F°) 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. Subcooling may be lost several times
* before the pumps can be left running.

5.2.4 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.

5.2.5 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.)

+ Rev. 1
* Rev. 2

5.2.6 OTSG Level *

If SCM is lost:

* a) Raise level on the unaffected OTSG to 95% while leaving level
* control on the unaffected OTSG at 30 inches.

* b) Raise level on the affected OTSG to 95%.

*NOTE: If incore thermocouple temperatures are not decreasing and there is
* no heat transfer to the OTSG's, then both OTSG levels must be raised
* to 95% simmultaneously.

5.2.7 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-V10 until an alternate source of gland steam is available.

When RCS T_{hot} is less than 540F° 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 mRem/hr whole body or 250 mRem/hr thyroid).

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.

5.2.7.1 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. OTSG level must be
* controlled below 600 inches on the wide range indication.

* If the OTSG must be steamed open the Turbine Bypass Valves or
* Atmospheric Dump Valve on the affected OTSG.

5.2.8 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.

5.2.9 <u>OTSG Shell-to-Tube Differential Temperature Limit</u> + ¢

\+ 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
\+ $.05 \times 10^6$ 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°.

5.2.10 <u>Cooling Mode When Both OTSG's are Unavailable for RCS Heat Removal</u> *

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.

5.2.11 <u>Core Flood Tank Isolation</u> *

* Isolate the Core Flood Tanks if:

* 1. Subcooling margin can be maintained above 25F°, and

* 2. RCS pressure is below 700 psig.

5.2.12 <u>Guideline Flow Chart</u> *

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

\+ Rev. 1
* Rev. 2
¢ Rev. 3

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 25F°

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 70F°.

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.

* 11. Core flood isolation criteria be incorporated into emergency
* procedures dealing with LOCA, tube rupture and steam line breaks
* and in operating procedures dealing with dorced and natural
* circulation cooldown.

* 12. Decay heat removal system initiation under emergency conditions
* be allowed at 300F.

¢ 13. The additional considerations for dose reduction provided in
¢ Appendix F during tube ruptures being appended to the tube
¢ rupture procedure.

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

+ Rev. 1
* Rev. 2
¢ Rev. 3

*

TABLE 1

* Tabular Values of RCP Emergency NPSH Requirements

FOR 2 RCP PER LOOP OPERATION *

	INDICATED TEMP.[2] (F)	ALLOWABLE INDICATED PRESS.[3] (PSIG)
¢	94.4	203.5 (1)
¢	194.4	207.2 (1)
¢	294.4	251.0 (1)
*	344.4	310.8
*	394.4	413.9
*	444.4	567.4
*	544.4	1187.1

FOR 1 RCP PER LOOP OPERATION *

	INDICATED TEMP. (F)	ALLOWABLE INDICATED PRESS. (PSIG)
*	94.4	115.5 (1)
¢	194.4	199.2 (1)
¢	294.4	243.0 (1)
*	344.4	302.8
*	394.4	405.9
*	444.4	559.4
*	544.4	1178.6

NOTE:

¢ 1. RCP operation will not be limited by NPSH requirements. Rather the requirement to maintaina 275 psi differential on the #1 pump seal may be limiting. Instrument error is not included in the 275 psig value.

¢ 2. Instrument error of 5.6F° for a small break LOCA condition has been subtracted from the actual reading on the temperature instruments (TI 959A and 961A).

¢ 3. An instrument error of 94.9 psi has been added to the actual reading for the wide range pressure instrument (949A, B) based on errors generated from a small break LOCA environment.

+ Rev. 1
* Rev. 2

TABLE 2(1)

Pressurizer Spray Flow for Various Pump Combinations

NUMBER OF RC PUMPS RUNNING

SPRAY LOOP	OPPOSITE LOOP	SPRAY FLOW (% FULL FLOW)
2	2	100
2	1	92
2	0	84
1 (Spray line next to running pump)	2	60
1 (Spray line next to running pump)	1	53
1 (Spray line next to idle pump)	2	50
1 (Spray line next to running pump)	0	41
1 (Spray line next to idle pump)	1	38
1 (Spray line next to idle pump)	0	26
0	2	20
0	1	0

As a rule of thumb tripping one pump in each loop will provide a good balance between the spray flow rate and the heater capacity. It will also provide good forced circulation for cooldown.

NOTE: The following table will give general guidance for the effects of running various pumps. This table was calculated for normal operating conditions.

NOTES:

1. Reproduced from TMI-1 ATOG (Ref. 28), "Best Methods for Equipment Operation", Table 6.

2. Entire table was added with Rev. 2.

+ Rev. 1
* Rev. 2
¢ Rev. 3

7.0 REFERENCES

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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.

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

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

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

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* Rev. 2
¢ Rev. 3

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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-1/8 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 20°F Subcooling Margin." March 31, 1983. 5211-83-017

* 20a. W. Drendall. "Saturation Margin Monitor Inaccuracy (Non-Accident Conditions)". Calc. No. C-1101-665-5350-002. June 14, 1983.

* 20b. G. J. Sadauskas. "TMI-1 Saturation Margin Monitor Loop Error Analysis." Calc. No. 1101X-3223-009 Rev. 2. June 14, 1983.. GPUNC.

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

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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. L.C. Pwu. GPUN Calculation # 1101x-5450-013, Rev. 2 "Minimum Subcooling Margin for RCP Operation." August 5, 1983.

* 27. L.C. Pwu. GPUN Calculation # 1101X-5450-015, "Saturation Margin Monitor Adjustment on Elevation Difference Between the Pressure Tap and the Tops of the Hot Leg." July 29, 1983.

* 28. B&W Abnormal Transient Operating Guidelines (ATOG). 74-1124158-00. Lynchburg, VA, April 24, 1983.

* 29. Telecon with Ralph Rosser (B&W) by L. C. Lanese. June 13, 1983.

* 30. Babcock & Wilcox Co. "Evaluation of SB LOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W - Designed Operating NSSS". B&W Doc. ID 77-1141270-00. Lynchburg, VA. February, 1983.

+ Rev. 1
* Rev. 2

* 31. L. C. Pwu. GPU Calc. No. 1101X-5450-014. "TMI-1 Cooldown Rate using Atmospheric Dump Valves", August 3, 1983.

* 32. J. P. Logatto. "TMI-1 Decay Heat Removal System". June 7, 1983. MSS-83-504.

* 33. N. G. Trikouros to R. J. Toole. "Thyroid Dose Limit for OTSG Isolation during SG Tube Rupture", SAPC #140, dated July 13, 1983.

* 34. TMI Unit 1 OP-1103-6, Rev. 24. "Reactor Coolant Pump Operation". Step 3.3.2.

* 35. J. R. White. RELAP5 Analysis of Steam Generator Tube Rupture Transients in a Generic Lower Loop Plant. R.P2420-4. Nuclear Safety Analysis Center. Palo Alto, CA. 1983.

¢ 36. G. Lehmann "TMI-1 OTSG Hot Testing Results & Evaluation". TDR 488. October 25, 1983. GPUNC.

¢ 37. R. L. Black "Supplement to ONS-3 Final ATOG". July 6, 1983. ESC-293. Babcock & Wilcox Co., Lynchburg, VA.

¢ 38. L. C. Lanese "TMI-1 90F/hr Cooldown Test of October 5 1983". October 10, 1983, SAPC # 197. GPUNC, Parsippany, NJ.

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FIGURE 1

STEAM GENERATOR TUBE RUPTURE GUIDELINE DEVELOPMENT-ACTIVITY NETWORK

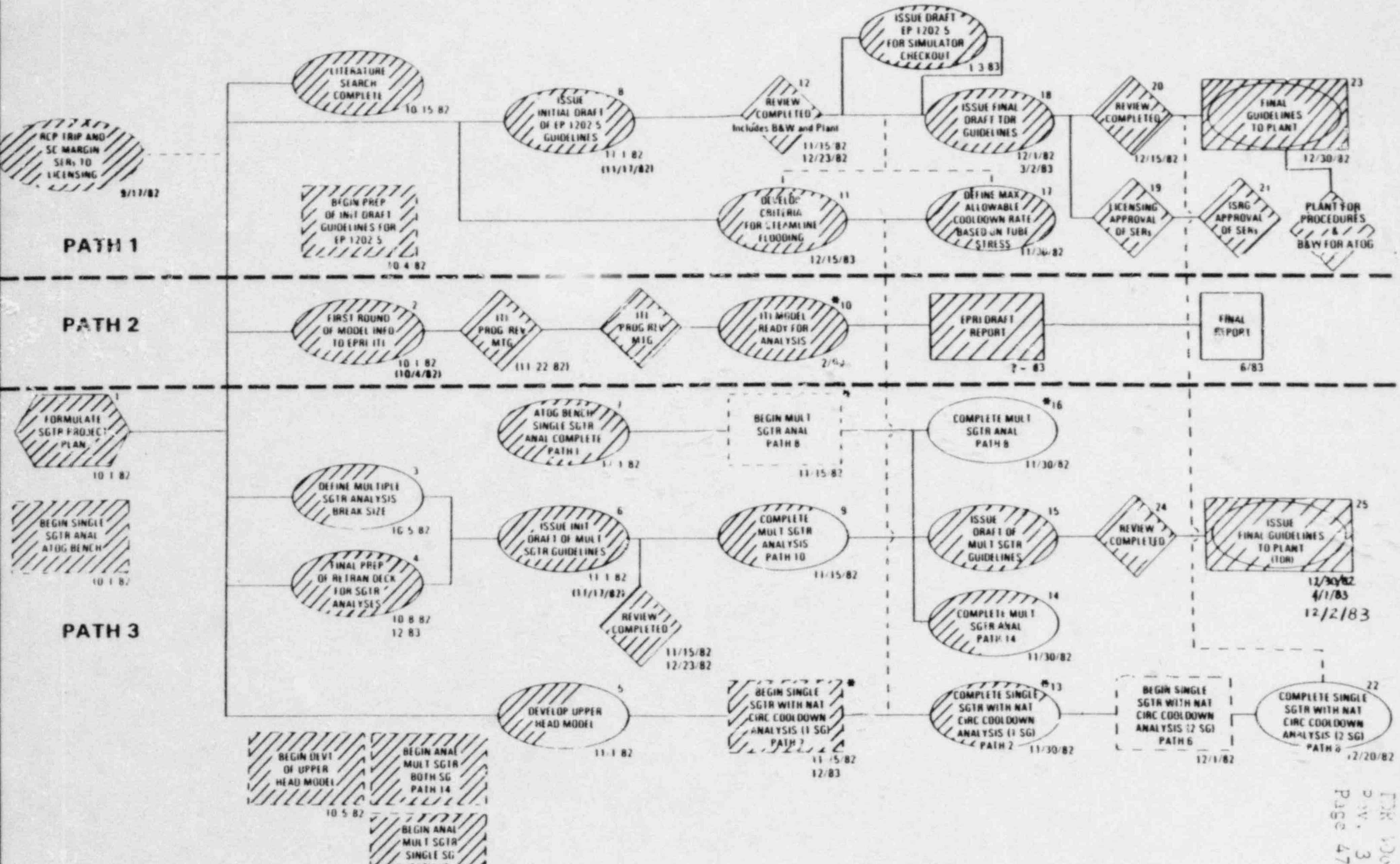


FIGURE 2

Break Flow for Single Ruptured Tube

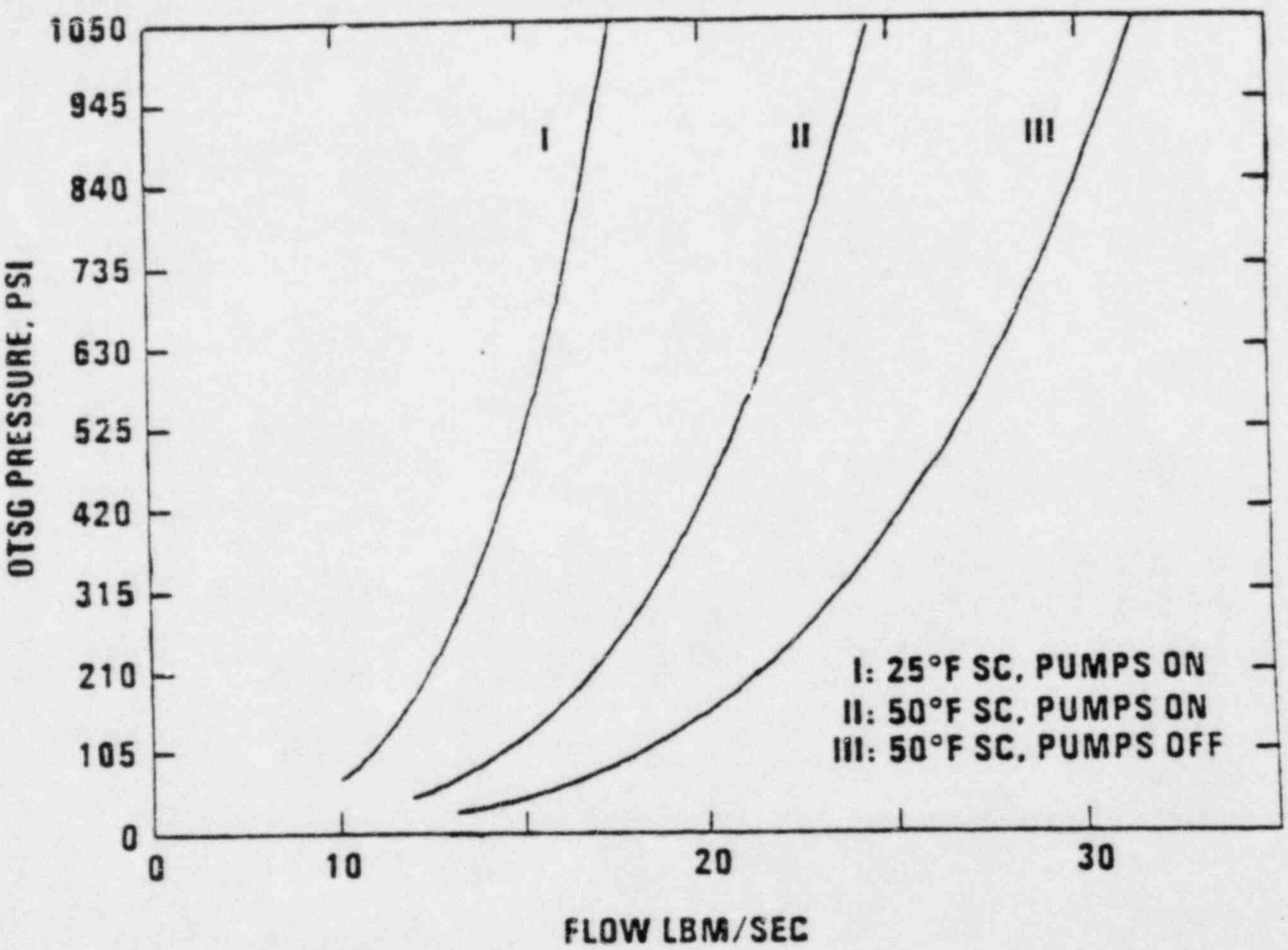


FIGURE 3

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

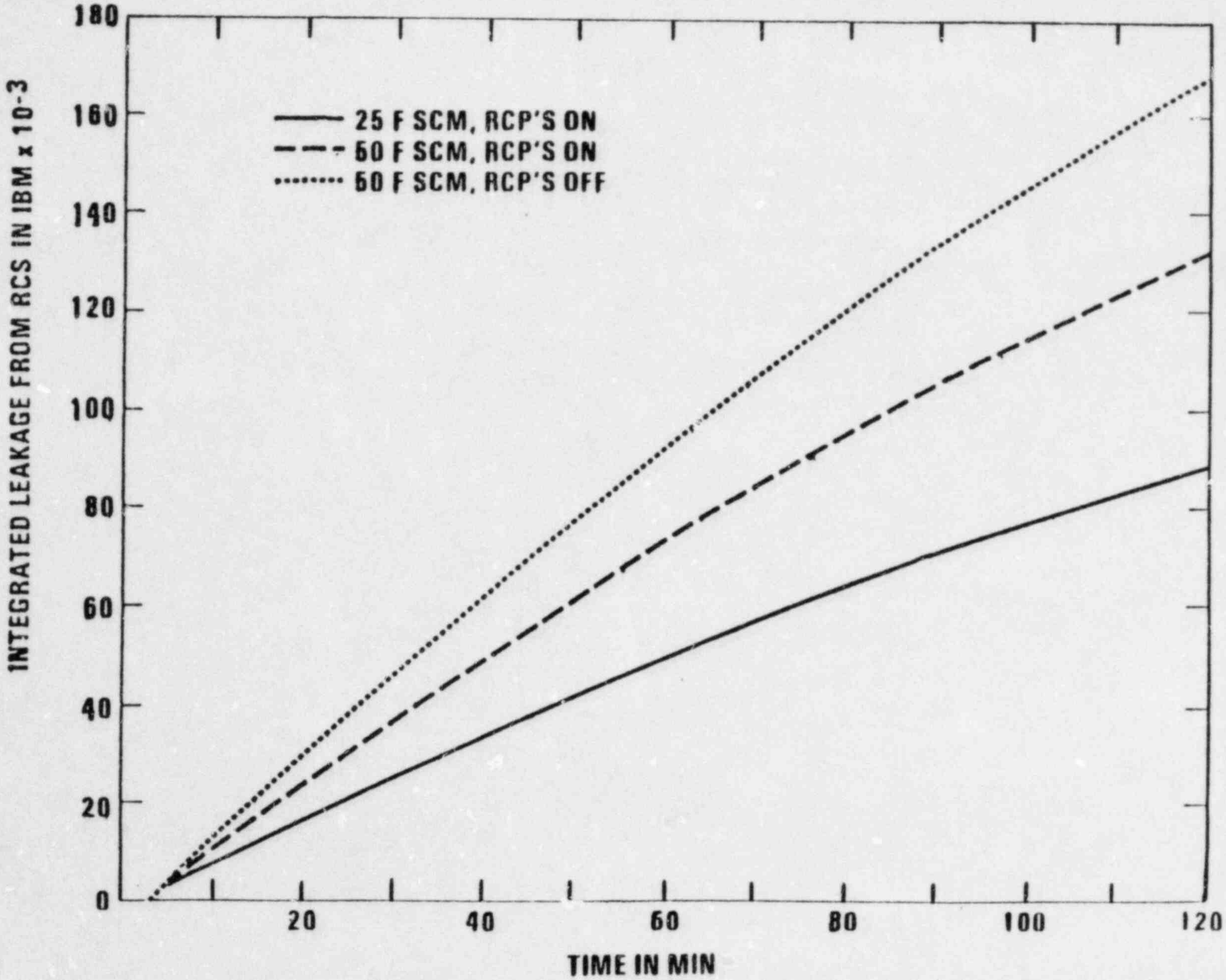
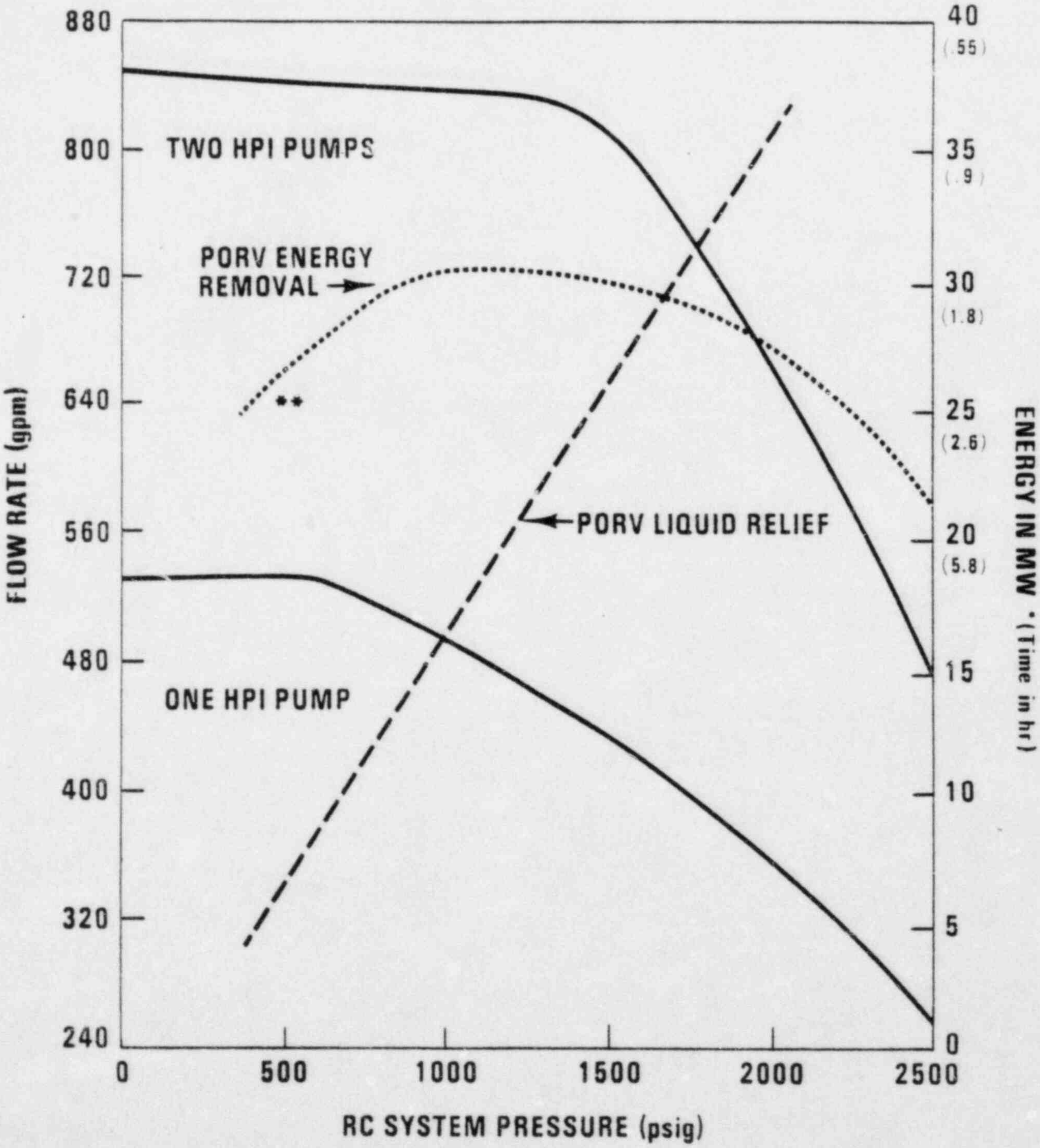


FIGURE 4

Mass and Energy Capabilities of HPI and PORV



**Energy relief is product liquid relief capacity and enthalpy of 100F subcooled water.

*1.0 ANS
2535 Mw (t)

FIGURE 5

Time Behavior of Subcooling Margin for a Spectrum of Ruptured Tubes

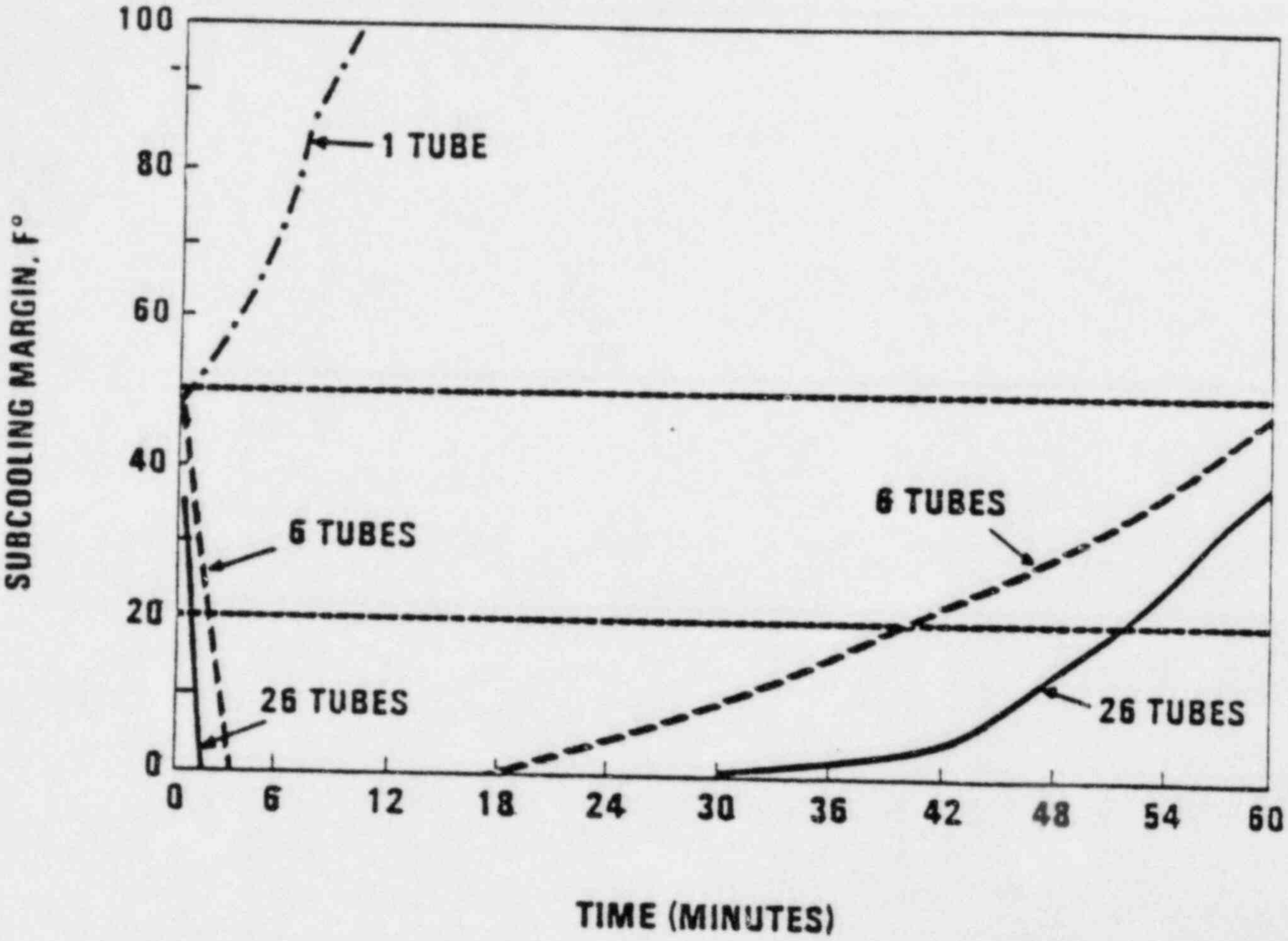
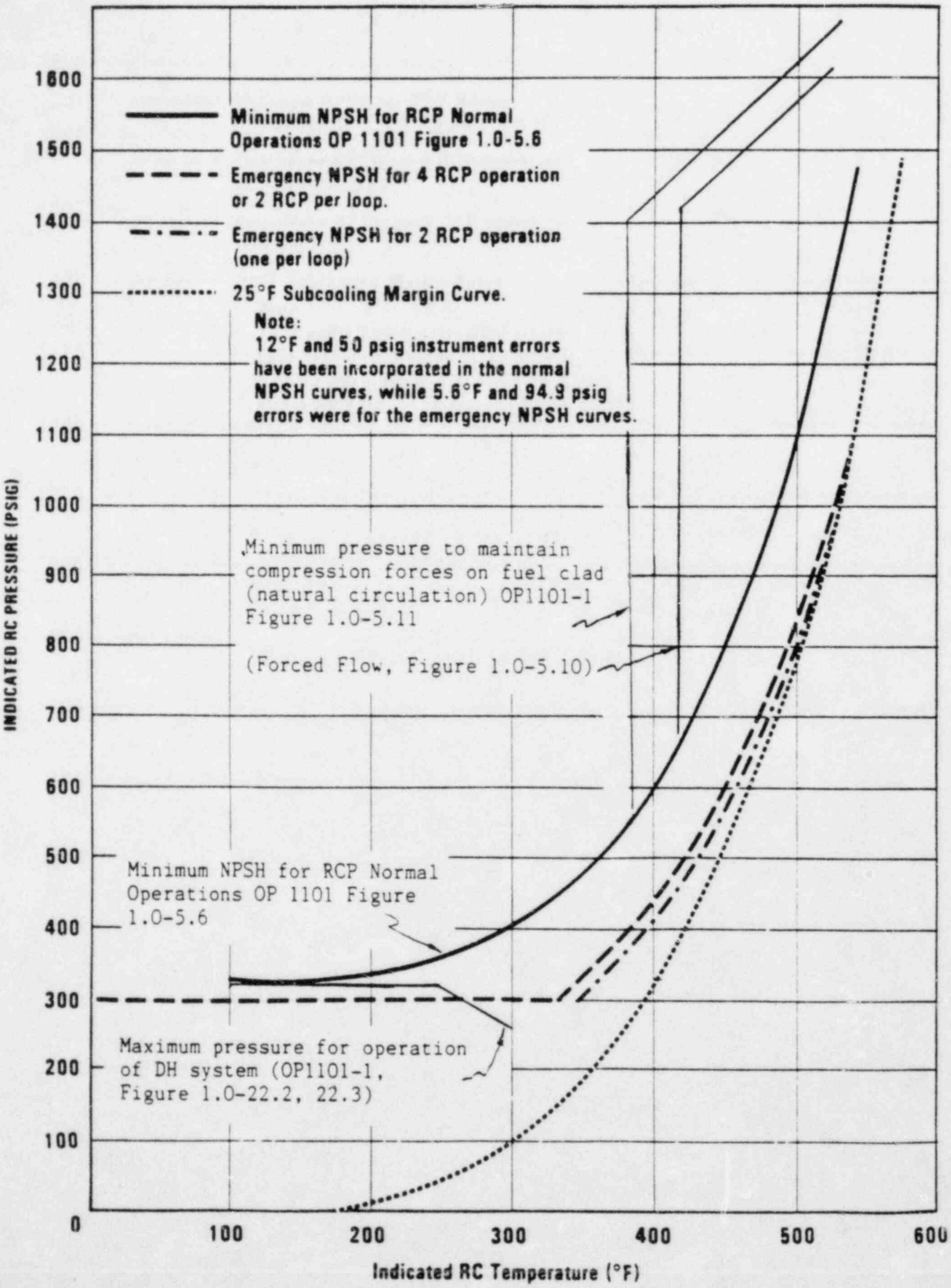


FIGURE 6

RCP NPSH Curves



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.

+ Rev. 1
* Rev. 2
¢ Rev. 3

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

Requirement	Source: ATOG	0737	0916	SOER	Addressed By: 1202-5	406	Comments
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.

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* Rev. 2
¢ Rev. 3

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.

+ Rev. 1
* Rev. 2

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	

+ Rev. 1
* Rev. 2
ℓ Rev. 3

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 a new guideline to be incorporated into the revised EP 1202-5.

A.1.4 Comments

* This column provides additional information if necessary. 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 a new guideline to be incorporated into the revised EP 1202-5.

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*
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+ Rev. 1
* Rev. 2
/ Rev. 3

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. * Shell/tube delta T of 70F° 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.
10. * Isolation of core flood tanks

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 associated with the subcooling margin monitor and alarm. Under normal containment
* conditions, the loop error is $\pm$ 10.1F° (Ref. 20a & b). Under the temperature and radiation environment of a small break LOCA, this error is no worse than - 21.7F°. The basis for the original SCM was

\+ Rev. 1
* Rev. 2
¿ Rev. 3

5F° geometry correction plus 45F° string inaccuracy. Recent calculations (Ref. 27) have shown that only 1.3°F geometry correction to the top of the hot leg is required. 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 Shell to Tube Delta T

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

The more restrictive 70F° 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 shell to tube 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 25. 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.

+ Rev. 1
* Rev. 2
ℓ Rev. 3

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).

+ Rev. 1
* Rev. 2
¢ Rev. 3

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 25F° 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. Pressurizer 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.

+ Rev. 1
* Rev. 2
¢ Rev. 3

B.9 + ADV's Open when RCS is above 1000 psig with no OTSG Heat Sinks

+ 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.

B.10 * Criteria for Core Flood Tank Isolation

* The purpose of the core flood tanks is to assure core cooling for LOCA's in which: 1) RCS pressure is below 600 psig, 2) HPI cannot provide core cooling, and 3) RCS pressure is too high for the LPI system to operate. The only situations when these conditions occur are: 1) design basis LOCA's, which HPI does not initiate before the core begins to uncover and 2) core flood line break accidents with an HPI failure, and 3) small break LOCA's in which the break is just large enough to remove decay heat, but not to depressurize the RCS.

* For the large break LOCA situation, CFT isolation is not a principal concern. The operator should isolate to prevent nitrogen introduction into the RCS once the tank is empty.

* For small break LOCA conditions, a subcooled RCS means that there is sufficient heat removal. In the pressure ranges in which core flood tank isolation is of interest, one HPI pump supplies sufficient flow to keep the core covered (500 gpm). If the RCS is 25F subcooled with the RCS below 700 psig, then the CFT can be isolated.

* The core flood tanks also function in one non-LOCA situation - steam line break accident. For large steam line break accidents, the CFT's provide shutdown margin assuming the most reactive rod is stuck out. Therefore, CFT isolation cannot occur until either HPI is operating and providing a source of borated water to the core or until all rods have inserted. If both these conditions are met, then a plant procedural change can be made without introducing an unreviewed safety question.

+ Rev. 1
* Rev. 2
ℓ Rev. 3

APPENDIX C

GUIDELINES FLOW CHART

+ Rev. 1
* Rev. 2
é Rev. 3

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, then 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.

+ Rev. 1
* Rev. 2
ℓ Rev. 3

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 with it), 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.

\+ Rev. 1
* Rev. 2
¢ Rev. 3

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 the following
* 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 Shell to Tube Differential Temperature

* It is necessary to minimize shell to tube differential temperature to
+ minimize tensile stresses on the OTSG tubes. As noted above, steaming
* is one 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.

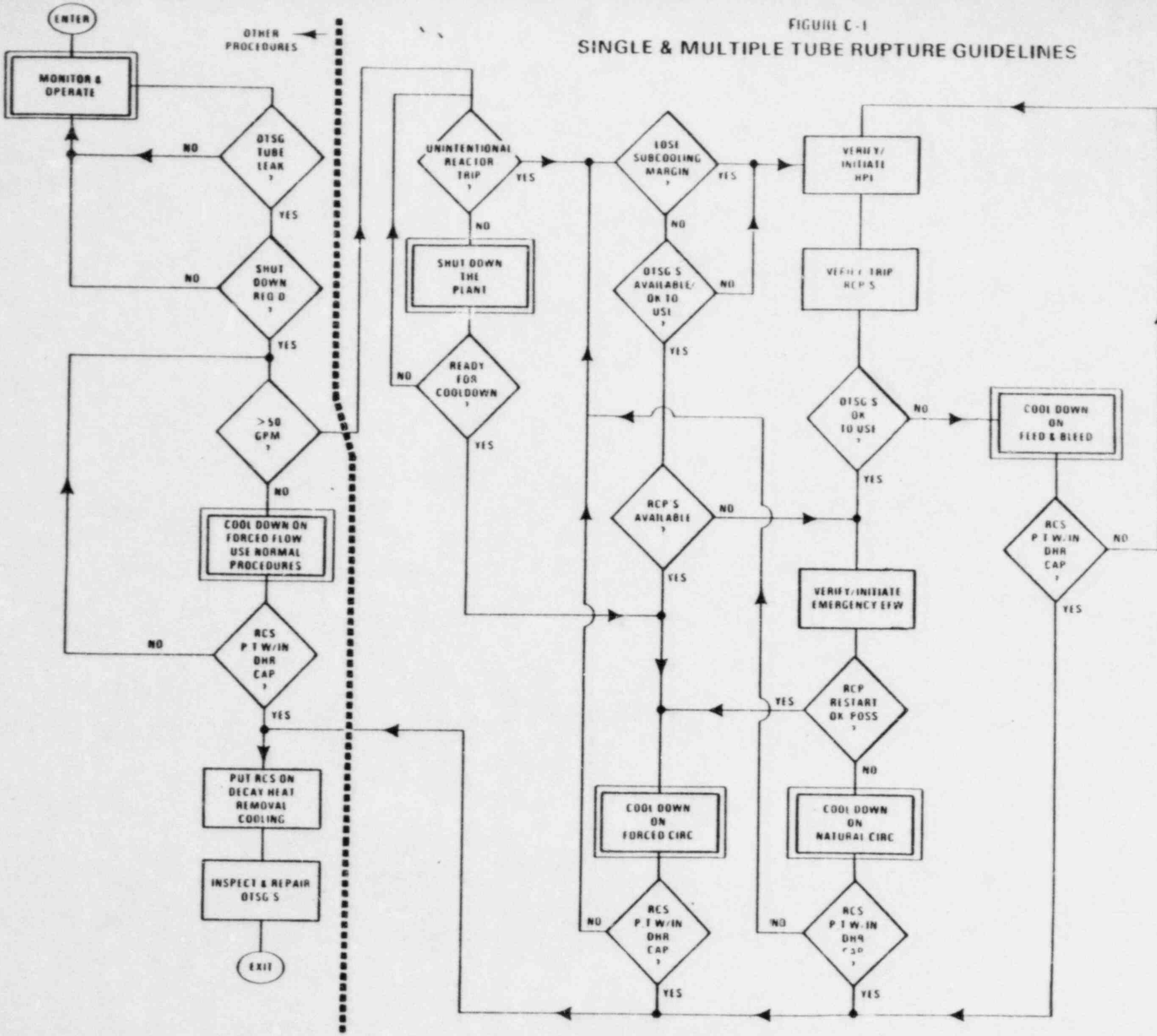
C.6 + EXIT POINT

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

+ Rev. 1
* Rev. 2
£ Rev. 3

FIGURE C-1

SINGLE & MULTIPLE TUBE RUPTURE GUIDELINES



APPENDIX D

SIMPLIFIED EVENT TREE

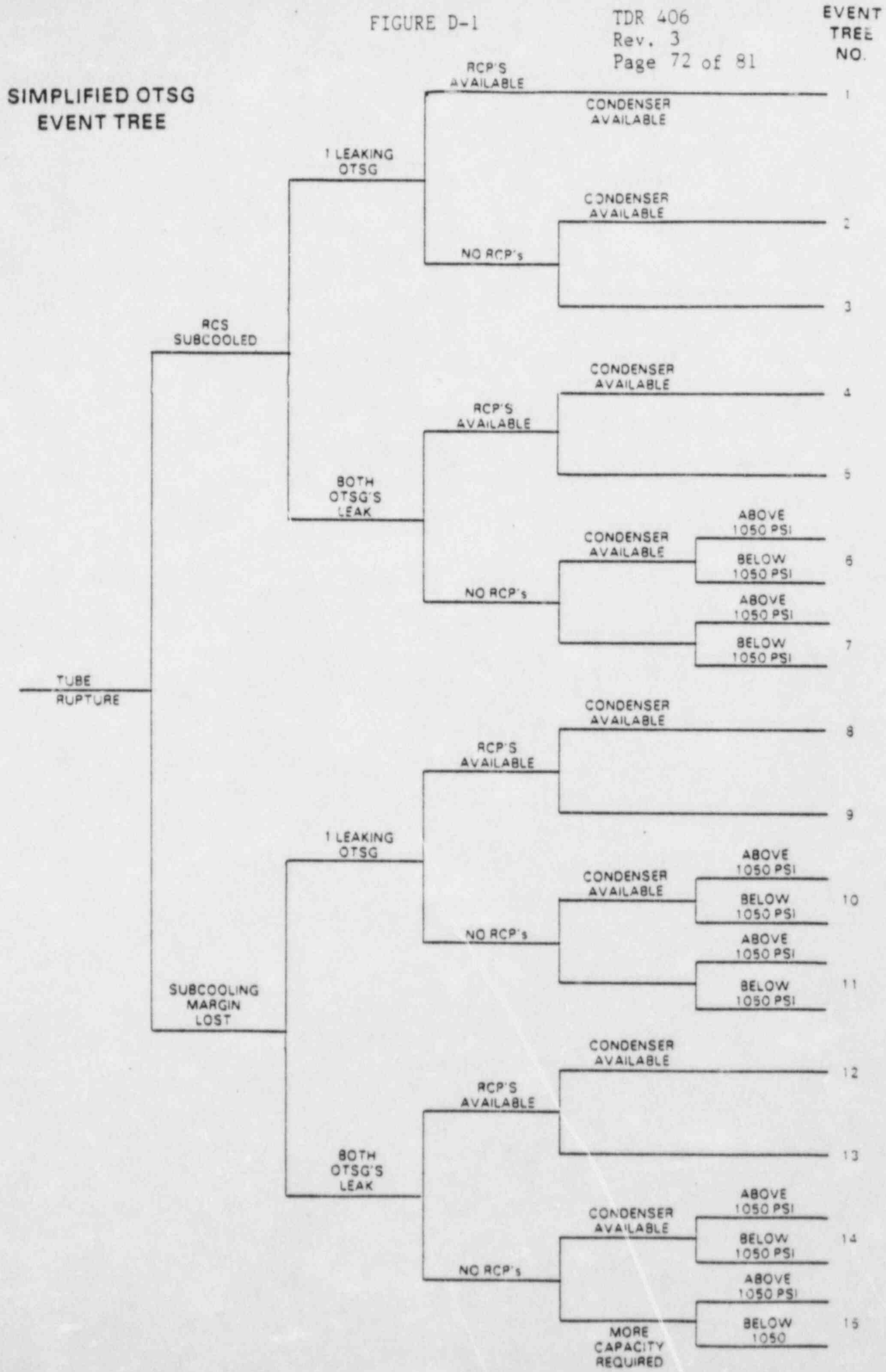
D.0 SIMPLIFIED EVENT TREE

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.

+ Rev. 1
* Rev. 2
ź Rev. 3

FIGURE D-1


SIMPLIFIED OTSG
EVENT TREE
EVENT TREE NO.
TUBE RUPTURE
RCS SUBCOOLED
SUBCOOLING MARGIN LOST
1 LEAKING OTSG
BOTH OTSG'S LEAK
RCP'S AVAILABLE
NO RCP's
CONDENSER AVAILABLE
ABOVE 1050 PSI
BELOW 1050 PSI
BELOW 1050
MORE CAPACITY REQUIRED
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15

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 25F°

E.1.2 + OTSG Tube to Shell Differential Temperature

¢ The process computer calculates shell temperature as follows for each
\+ OTSG if all shell thermocouples are operable:

\+ $T_{shell} = 0.242\ T_1 + 0.176\ T_2 + 0.201\ T_3 + 0.143\ T_4 + 0.238\ T_5$
(E-1)

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

¢ If the process computer is not available, the arithmetic average of
¢ the five thermocouples can be used. Experience from the Fall 1983
¢ cooldown tests demonstrated that an arithmetic average approximates
¢ the weighted average of equations E-1 within several degrees (see
¢ Table E-1 for a comparison of the arithmetic and weighted average).
¢ If a thermocouple is unavailable, then the arithmetic average should
¢ be calculated using the substitutions of Table E-2 for shell
¢ thermocouples. Table E-3 provides the substitution for cold leg
¢ temperatures (indicating tube temperature).

¢ It is also possible that thermocouple temperatures are not
¢ available in any form except from via translation from voltage
¢ readings. In this situation, one thermocouple from the upper and
¢ lower downcomers should be used and the shell temperature
¢ calculated as:

¢ $T_{shell} = .6\ T_e + .4\ T_u$

¢ where $T_e = T_1$, T_2 or T_3, and

¢ $T_u = T_4$ or T_5

\+ Rev. 1
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¢ Rev. 3

b

TABLE E-1

Comparison of Weighted vs. Arithmetic Average of OTSG Shell Thermocouples

1. 90F/hr Cooldown with MFW (9/19/83)

TIME (min)	WEIGHTED AVERAGE	ARITHMETIC AVERAGE
0	516.9	516.8
45	515.1	515.1
85	481.7	482.5
125	438.3	439.8
165	391.3	393.6
195	382.0	383.4
252	376.2	378.8

2. 90F/hr Cooldown using EFW (10/2/83)

TIME (min)	WEIGHTED AVERAGE	ARITHMETIC AVERAGE
0	524.1	524.3
49	509.9	510.7
89	483.8	483.4
127	452.3	454.4
167	401.2	401.2
197	386.1	386.1
257	397.4	400.6

ℓ

Table E-3 Wide Range T_{cold} Input Substitutions

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

+ Rev. 1
* Rev. 2
ℓ Rev. 3

ℓ Table E-2 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 Shell to Tube delta T as follows:

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

\+ T_{T-S} should trigger an alarm state if

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

\+ Rev. 1
* Rev. 2
ℓ Rev. 3

APPENDIX [illegible]

ADDITIONAL STEAMING AND ISOLATION CRITERIA FOR

REDUCTION OF RADIOLOGICAL RELEASES

¢ F.0 ADDITIONAL STEAMING AND ISOLATION CRITERIA FOR REDUCTION OF RADIOLOGICAL RELEASES

¢ The steaming and isolation guidelines provided in Section 5.2.7 are intended for the control room operator. Their development required the balancing of diverse factors including ease of use and equipment protection. The latter consideration in itself had safety implications, since protection of one piece of equipment, namely the steam generators, impacts offsite releases.

¢ Once the various emergency support groups assemble, opportunities develop for minimizing releases for the specific plant condition. While this specific event treatment is not appropriate for the operator to deal with in the short term, it is appropriate for the emergency support groups. This appendix provides guidelines for varying from the isolation and steaming criteria. The recommendations in the following sections spring from the following considerations:

¢ 1. Isolation of one or both OTSG's reduces the RCS cooldown rate which increases the time to reach cold shutdown and to terminate the primary to secondary leak.

¢ 2. Isolation of one OTSG when both are leaking may increase the integrated dose since the release will continue from the unisolated OTSG for a longer period.

¢ 3. Isolation of both OTSG's requires feed and bleed cooling which could result in releases of steam or steam and water directly to the atmosphere.

¢ 4. An isolated OTSG may flood, after which it may not be possible to unisolate and return the OTSG to service.

¢ 5. Isolation of direct steam releases to the atmosphere is expected to reduce the offsite thyroid dose by a factor of at least .

¢ 6. Isolation for dose reduction should be based on measured dose rate to preclude premature isolation.

¢ Table F-1 summarizes the guidance provided in Section F-1 through F-3.

¢ F.1 OTSG ISOLATION SHOULD BE AVOIDED IF

¢ 1. RCP's are not available - natural circulation cooldown may not be possible with one OTSG since flow in one loop might stagnate and a bubble could form in the hot leg as primary pressure is reduced.

¢ 2. Both OTSGs leak but the difference in leak rate is less than a factor of eight - otherwise, the delay in cooldown may negate the dose reduction from isolating one OTSG.

+ Rev. 1
* Rev. 2
¢ Rev. 3

F.2 OTSG ISOLATION MAY BE DESIRABLE IF

1. RCP's are operating, the condenser is unavailable - only one OTSG leaking and iodine dose rates are high - in this situation, high iodine release rates could be terminated by isolation of the leaking OTSG.

   Although cooldown time is increased, radioactivity releases will be terminated. RCP operation enables control of the RCS, which in turn allows cooldown of the leaking OTSG.

F.3 OTSG ISOLATION CRITERIA SHOULD BE RE-EVALUATED IN THE FOLLOWING SITUATIONS

1. RCP's operating, condenser unavailable, both OTSGs leaking, iodine dose rates are high - isolation of one OTSG may be desirable if the leak rate in one OTSG is significantly (about 8) greater than in the other. The reduced dose rate from isolation of one OTSG must be weighed against the shorter cooldown time with steaming of both OTSGs.

2. Condenser available - isolation of one or both OTSGs greatly increases cooldown times and increases risk of an inadvertent or uncontrolled release. A decision to isolate earlier than required by procedural guidelines should be based on measured dose rates if possible. In the absence of fuel failures, actual releases under such conditions are expected to be quite low.

3. Only one OTSG is leaking and BWST level is 21 feet - if the good OTSG is not expected to leak because shell/tube delta T is being controlled, then isolation is not required. Recall that the BWST level isolation criterion was based on both steam lines being flooded. If only one OTSG may be flooded, then BWST depletion could not occur until level reaches 15 ft.

F.4 TEMPORARY SHORT TERM MEASURES TO REDUCE OR TERMINATE RELEASES
Releases can be temporarily terminated without initiating feed and bleed cooling by -

a. Terminating steaming to the condenser and not steaming to atmosphere.

b. Not more than 1 RCP should be run.

c. If natural circulation is lost, steam again.
   If steam generator pressure reached 1000 psi, steam again.
   Heatup rate is 100-170F°/hr.

These steps may provide enough time to return an RCP to operation, restore a condenser to service, or initiate protective actions, while delaying the initiation of feed and bleed cooling.

TABLE F-1

SUMMARY OF DOSE REDUCTION CONSIDERATIONS

RCP	CONDENSER	ONE OTSG LEAKING	BOTH OTSGs LEAKING MORE THAN 8:1 DIFFERENCE	BOTH OTSGs LEAKING EQUALLY
On	Available	Avoid Isolation (F.3.2)	Avoid Isolation (F.3.2)	Avoid Isolation (F.3.2, F.1.2)
On	Not Available	Consider Isolation (F.1)	Consider Isolation (F.1, F.3.1)	Avoid Isolation Of One (F.1.2)
OFF	Available	Avoid Isolation Of One OTSG (F.1)	Avoid Isolation Of One OTSG (F.1)	Avoid Isolation Of One (F.1.2, F.3.2)
OFF	Not Available	Avoid Isolation Of One OTSG	Avoid Isolation Of One OTSG (F.1)	Avoid Isolation Of One (F.1, F.2)

NOTE: Condenser available means main condenser or steaming to auxiliary condenser via MFP.

+ Rev. 1
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May 17, 1984
5211-84-2093

Office of Nuclear Reactor Regulations
Attn: John F. Stolz, Chief
Operating Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Stolz:

Three Mile Island Nuclear Station Unit I, (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
TMI-1 Steam Generator Tube Rupture Guidelines
TDR-406 Rev.3

Attached for your information is the most recent revision to the TMI-1 Steam Generator Tube Rupture Guidelines. This revision incorporates changes indicated in previous correspondence. It should be understood that this TDR is a living document and as such will be revised from time to time to include more recent information. The NRC staff will be kept informed as significant revisions occur.

Sincerely,

RJ Toole for

H. D. Hukill,
Director, TMI-1

HDH/CWS/mle

Enclosures

CC: R. Conte
H. Silver

A001
1/1