ENCLOSURE
TXX-92009

3/23/92

RHR LINE SUMMARY REPORT

1.0 INTRODUCTION AND BACKGROUND

The United States Nuclear Regulatory Commission issued Bulletin 88-08 Supplement 3 (Reference 1) following the discovery of a valve leakage induced fatigue crack in the residual heat removal (RHR) suction piping at Genkai Unit 1 nuclear power plant. This bulletin requested utilities to identify susceptible piping systems, inspect potential crack locations and provide continuing assurance of piping integrity for the life of the unit.

TU Electric has been very responsive to NRC Bulletin 88-08 requests. An initial evaluation of the Unit 1 RHR piping was completed in April 1989 (Reference 2 - original issue). A second evaluation was completed in August 1989 (Reference 2 - Supplement 2). This evaluation considered a variation of the stratification loading, i.e. stratification initiating in the horizontal piping upstream of the first isolation valve.

As a result of the evaluation performed in Reference 2, temporary temperature monitoring locations and criteria were established, and TU Electric has been continuously monitoring the Unit 1 RHR piping to provide continuing assurance that the RHR suction piping is not subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the units.

As a result of successful data collection for the first fuel cycle, a review has been conducted to determine if valve leakage is occurring. In addition, an evaluation has been performed to determine augmented inservice inspection intervals (based on fatigue usage and fatigue crack growth methodology, and assuming continuous cyclic valve leakage), thus satisfying NRC Bulletin 88-08 requirements without continuous monitoring. The purpose of this report is to document the results of the monitoring data review and to evaluate the postulated valve leakage condition.

2.0 OVERALL EVALUATION APPROACH

2.1 General

TU Electric has placed temperature monitoring devices at several locations on the RHR suction piping to detect adverse thermal transients as described in item 3 of Reference 1. After reviewing the monitoring results, it was found that there was no evidence of any cyclic leakage in the valves. This report addresses NRC Bulletin 88-08 requirements, evaluates monitoring data and presents technical justification to eliminate temporary monitoring devices on Unit 1.

2.2 Technical Approaches

While no temperature distributions which would indicate cyclic valve leakage were observed from the monitoring data, it was conservatively postulated that the out leakage from the hot leg would occur through the isolation valves (8702A and 8702B). During plant operation, such leakage is postulated to cause stress cycles between leakage and no-leakage. (The phenomena of leak/no-leak is considered herein as a postulated condition and should not be treated as a design condition.)

The steps in the structural evaluation of such postulated condition are listed below:

- Definition of stratified transients from postulated valve leakage
- Definition of stratified transients from monitored data (other than valve leakage)
- Stress calculations from all cases of thermal stratification
- Fatigue usage calculation including postulated valve leakage, monitored transients and design transients
- Fatigue crack growth calculation including postulated valve leakage, monitored transients and design transients

- Augmented ISI determination based on fatigue usage and fatigue crack growth calculations.

Comanche Peak Unit 1 began commercial operation August 13, 1990. Monitoring data from the Unit 1 piping has shown no evidence of cyclic valve leakage. Given that the units are new and have experienced little or no fatigue cycles, it is highly unlikely that cracks are present in the RHR piping of Comanche Peak Unit 1. Fatigue usage and fatigue crack growth have been calculated assuming that cyclic valve leakage occurs, resulting in stress cycles. Fatigue usage provides a minimum time required to initiate the crack. Fatigue crack growth provides a minimum time required to propagate a crack to 60% of the wall thickness, assuming an initial crack size of 10% of the wall thickness. Augmented inservice inspection intervals based on conservative fatigue usage and fatigue crack growth calculations provide a strong technical justification to eliminate temperature monitoring of the RHR piping, while still satisfying the requirements of NRC Bulletin 88-08.

3.0 MONITORING DATA REVIEW

Monitoring Data for the Comanche Peak Unit 1 RHR loops 1 and 4 suction piping (Reference 3) were reviewed to determine if significant thermal stratification and/or cycling had occurred. These data were reviewed for the period from 3/16/90 to 7/14/91.

The Loop 1 and Loop 4 RHR suction lines were instrumented on the pipe outer wall with resistance temperature detectors (RTD's) as shown in Figures 3-1 and 3-2. The purpose of each monitoring location is as shown below:

RTD ID (13A, B)	Purpose
6, 7	Monitor temperature of vertical leg to establish boundary condition temperature, and provide a qualitative measure of turbulent penetration.
1, 2, 5	Monitor stratification magnitude, profile and frequency of cycling.
4	Monitor valve leakoff temperature (provide root cause information - packing leak)
3	Monitor bypass line temperature (provide root cause information - bypass valve leakage)

In addition to the temporary sensors shown above, the following plant information was also reviewed. (This information was obtained from the plant computer and operator logs.)

- Hot leg temperature for Loops 1 and 4
- RCS flowrate for Loops 1 and 4
- RCP Operation
- RHR Operation
- Safety Injection Operation

Thermal stratification was observed in both RHR lines (connecting to loops 1 & 4) during heatup and cooldown operations that involved lineup and operation of the RHR systems. The stratification events were directly caused by opening of the RHR isolation valves and relatively low flow in the lines. This stratification was characterized by low delta T's (less than 20°F) and no significant cycling was observed.

During normal operations (reactor thermal power >95%) a more significant observation was made. Temperature measurements on the unisolable side of the loop 1 RHR isolation valve were hot, close to that of the loop 1 hot leg temperatures. This result compares favorably to results from flow model testing which suggest that turbulent penetration from primary loop flow should penetrate approximately 22 pipe diameters. The RHR isolation valve is approximately 14 pipe diameters from the loop pipe. Except for certain test conditions, RHR operations, and one reactor trip (in which all RCP's tripped) there were no unexpected thermal events in loop 1 RHR line. However, loop 4 RHR monitoring data displayed a significantly different response to normal operating conditions (reactor thermal power >95%). During normal operations temperature measurements on the unisolable side of the loop 4 RHR isolation valve were cold, between 95 and 120°F. It should be noted that during this time no significant stratification was observed during power operations. Since the two RHR lines were for all practical intents identical in layout, further investigation into the cause of the cold temperature readings during normal operations was merited. It was eventually concluded that there was an insufficient turbulent penetration effect (heat transfer by a mass transport mechanism) to heatup all of the inventory in the unisolable section of loop 4 RHR. It is further postulated that the reduced turbulent penetration effect is the result of having two branch pipes in very close proximity to each other on the primary loop. In this case, the loop 4 RHR line (a 12 inch line) is 15 inches away from the pressurizer surge line connection (a 14 inch line). It is postulated that the two branch pipes in close proximity to each other result in reduced turbulent penetration energies available to either line. Therefore, the total mass exchange that occurs in the loop 4 RHR line is less than that of the loop 1 RHR line and hence the line cools to ambient after some period of time.

Details of the observed stratification are shown below:

RHR Line	Extent Stratified	Stratification Duration	Max ΔT	No. of Occurrences	Probable Cause
Loop 4	Upstream of 8702B	3 days	391°F	6	Plant Cooldown/RCP Operation
Loop 1 & 4	Upstream and downstream of 8702A and 8702B	6 hours	170°F	1 (Loop 1) 1 (Loop 4)	RHR Operation RHR Operation
Loop 1 & 4	Upstream of 8702A and 8702B	5 hours	112°F	1 (Loop 1) 1 (Loop 4)	Reactor Trip, RHR Operation
Loop 1	Upstream of 8702A	2 hours	90°F	2	Reactor Trip
Loop 1	Upstream and downstream of 8702A	7 hours	94°F	1	Test Condition

It should be noted that the above transients from the monitored data do not reflect any cause from valve leakage, rather from plant operation. These transients have been conservatively included in the fatigue and fatigue crack growth analyses.

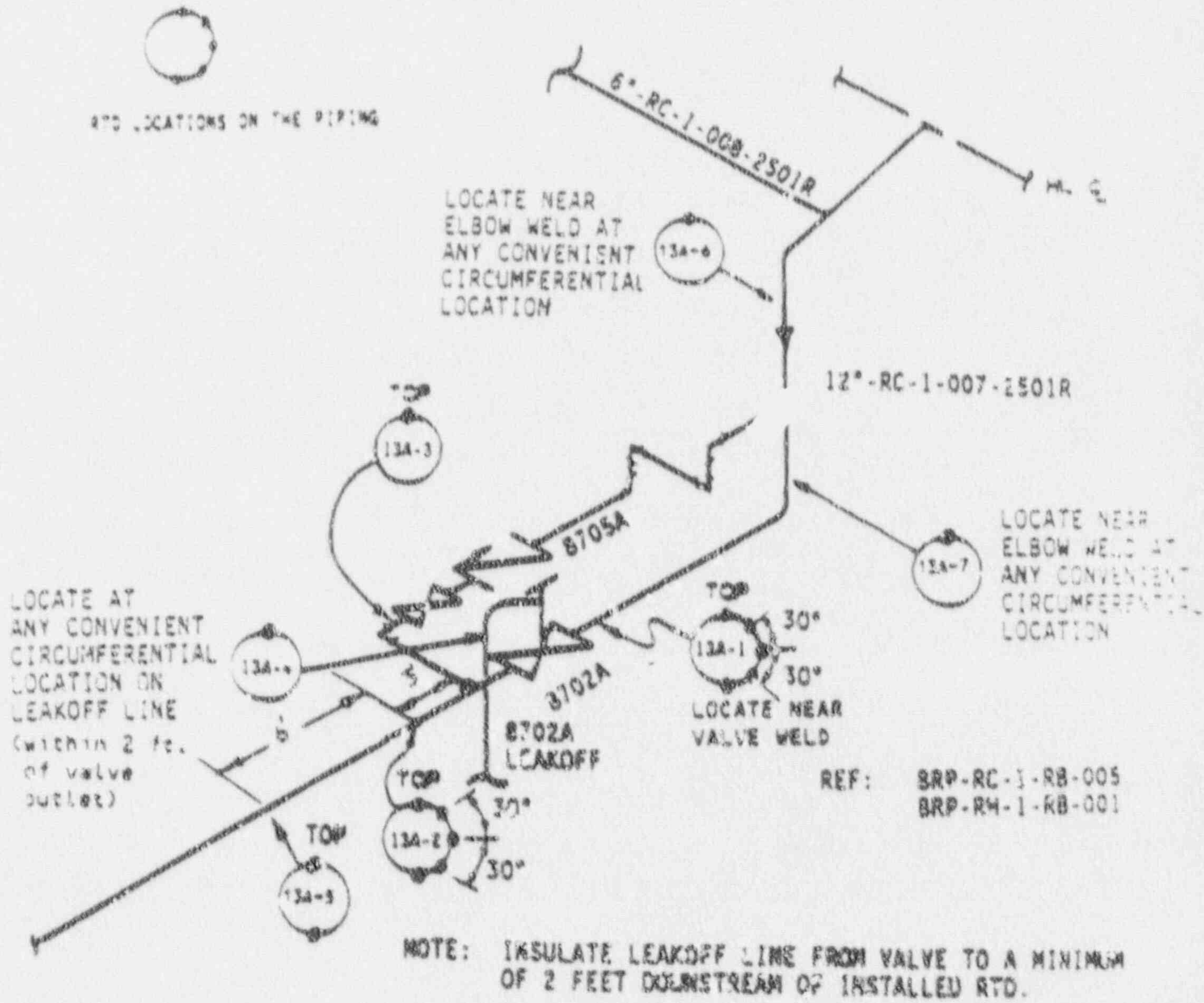


Figure 3-1. Unit 1, Loop 1 RHR Monitoring Locations

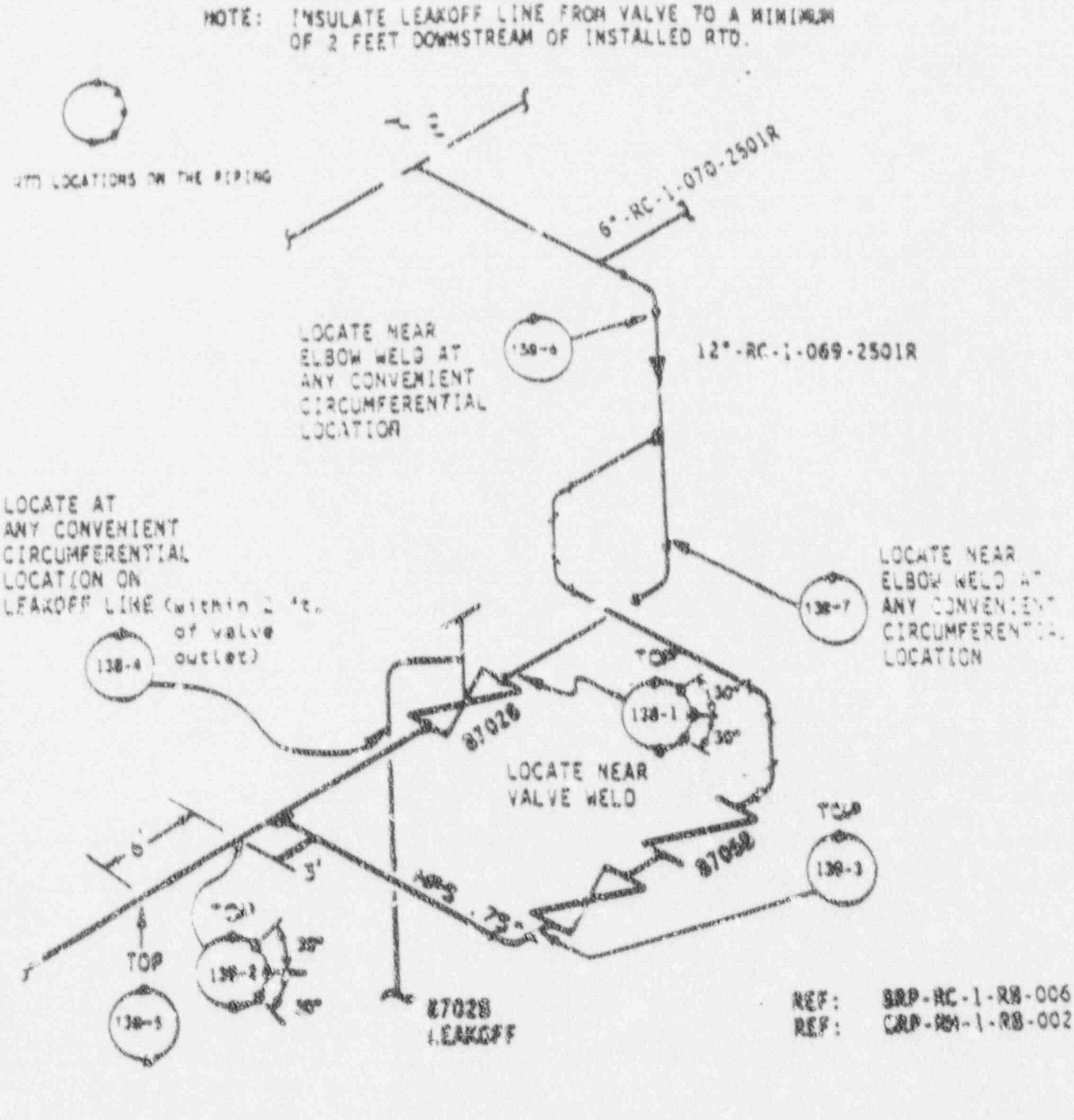


Figure 3-2. Unit 1. Loop 4 RHR Monitoring Locations

4.0 TRANSIENT DEVELOPMENT

4.1 General Discussion of Postulated Valve Leakage Transients

The NRC Bulletin 88-08 requires licensees to postulate that valve leakage may occur in the RHR isolation valves. In general, the only leak scenarios that can be applied to the RHR lines are out leakage (due to the pressure differences between the primary side and the downstream portion of the lines). The out leakage could either be through the leak off line of the isolation valve or to the downstream side. In either case, if a periodic (cyclic) leak occurred in the loop 1 RHR isolation valve there would be no impact on the unisolable portion of the piping. This conclusion is true since the loop 1 unisolable portion of the RHR line is already hot due to the turbulent penetration. The introduction of primary coolant into that region of the piping would not change the thermal state. In addition to postulated valve leakage, the possibility of stratification in the unisolable piping was also addressed.

4.2 Development of the Operational Related Stratification Transients

4.2.1 General

Stratification was observed in both loop 1 and loop 4 RHR lines during lineup and operation of the RHR system as discussed in Section 3.0. Stratification was not observed in loop 1 during operating modes 1 through 3.

During hot standby operations stratification was observed in the loop 4 RHR line. The stratification that was observed was the direct result of the existing condition (cold water approximately 100°F in the unisolable side of the RHR line and hot water approximately 557°F in the loop) and RCP operations that resulted in loop 4 primary coolant flow increasing to approximately 110% of normal flow. With the loop 4 line cooled to ambient near the isolation valve during hot standby operations and the increase in primary loop flow, the already depleted turbulent penetration depth was increased. This

resulted in a higher mass transfer rate with the primary loop coolant that resulted in the introduction of hot water near the isolation valve. This condition stratified the horizontal section of the pipe for some period of time. The maximum pipe delta temperature observed during these events was 391°F. The total number of these events observed during the monitoring period was 6, and all were associated with increased primary loop flow.

4.2.2 Behavior of Loop 4

The presence of cold water in the unisolable section of loop 4 RHR during normal operations raises the question of what the interface is like between the primary coolant and the isolated RHR inventory. It should be noted that there is insufficient data at this time to conclusively support any single hypothesis; however, there are at least two possible scenarios. One, the interface between the hot primary coolant and the cold water is a gradual temperature gradient that is stable and non-cyclic and restricted to the vertical section of the loop 4 RHR line. Therefore, the only adverse loading would be those events already accounted for by considering 240 cycles of increased primary loop flow in loop 4. The load condition for this scenario (vertical temperature gradient) is described in Table 4-1 as transient number 1. A vertical temperature gradient restricted to the vertical section would result in an axial temperature distribution that dropped off quickly from the primary loop temperature at the top of the vertical segment to ambient temperature at the bottom.

The second scenario is suggested from review of the Comanche Peak monitoring data. From Figure 4-2 it can be seen that the relative circumferential locations of RDT 6 and RTD 7 are 180 degrees apart. The monitoring data from these locations suggest that a current exists in the vertical section of the pipe. This condition is highly speculative since location 6 did display erratic data: at some times the readings were negative. However, at other times the readings were normal and within acceptable engineering ranges. During periods of normal power operations (reactor thermal power > 95%) the temperature readings at location 6 are colder than location 7. If the readings from

location 6 are to be believed, this condition could only be explained by a vertical current that is driven by the turbulent penetration. Turbulent penetration at the 45 degree bend would provide the pumping action by establishing an entrainment region that pulled cooler water from the lower region of the pipe. Since the overall turbulent penetration is low, the rate of mass transfer of the RHR line inventory with the primary loop inventory is low. Hence, the introduction of heat through the mass transport mechanism is not sufficient to heat the entire line; this is consistent with the observed data at location 1. This condition is illustrated in Figure 4-3. Transients that consider these effects are described in Figure 4-2 as Case 1, Case 2, and Case 3.

4.2.3 Operational Transients that Apply to Both Loop 1 and Loop 4

Thermal stratification was observed in both RHR lines (connecting to loops 1 & 4) during heatup and cooldown operations that involved line up and operation of the RHR systems. The stratification events were directly caused by opening of the RHR isolation valves and relatively low flow in the lines. These events were characterized by delta T's less than 200 deg F and no significant cycling.

Transients that envelope these observed conditions are listed in Table 4-1. Transients 2 through 5 are applicable to both loop 1 and loop 4 RHR lines. Transient 1 applies to loop 4 only.

4.3 Development of the Postulated Valve Leakage Transients

The transient at Genkai was due to intermittent valve leakage, which provided a path for hot water to be drawn into the RHR line from the main loop. In the horizontal piping downstream of the second elbow from the RCS connectic , a stratified flow was established, with hot water filling the top of the pipe to a depth of 10 percent of the inner diameter.

To establish a postulated valve leakage transient for this scenario, stratification was assumed to exist in the horizontal piping upstream and downstream of the isolation valve (8702B on loop 4). The same portion of the pipe as at Genkai was assumed to be filled with leakage flow (i.e. 10 percent of the inner diameter with a leakage rate of 1.0 gpm). The bulk fluid was assumed stagnant, and therefore its temperature declines quickly with axial distance, since heat transfer is primarily conduction. The leakage at the top of the pipe does not cool as quickly, due to its flow as shown in Figure 4-1. This creates a rather large temperature differential between the top and bottom of the pipe, which maximizes at an axial distance of approximately 4 feet from the second elbow, and diminishes at an axial distance of approximately 20 feet from the second elbow. At the first isolation valve inlet weld, Figure 4-1 provides 565°F at top of the pipe and 360°F at the bottom of the pipe with $\Delta T = 205°F$.

As mentioned in Section 4.2, the actual monitoring data from the unisolable section of loop 4 revealed an alternate interpretation of the axial temperature distribution that could be present during valve leakage conditions. In order to account for the alternate interpretation, several additional independent load cases were postulated for valve leakage. These cases assumed that the vertical current existed (Scenario 2 in Section 4.2.2). These load cases are considered as alternate states that are independent of and replace loading conditions during hot standby and normal operating conditions that did not assume a vertical current (Scenario 1 in Section 4.2.2). They were analyzed for their impact on fatigue life and fatigue crack growth. The postulated valve leak transients for these cases are shown in Figure 4-4.

TABLE 4-1

LOAD CONDITION FOR THE VERTICAL TEMPERATURE GRADIENT

Transient	Max Pipe ΔT	Top-to-Bottom Temperature Profile @ Max ΔT	Total Number of Cycles Extrapolated
1	391°F	Step Change at 3.3" from top I.D.	240
2	170°F	Step Change at 1.6" from top I.D.	200
3	112°F	Linear	80
4	90°F	Linear	400
5	94°F	Step Change at 7.3" from top I.D.	N/A*

*Test condition

Note: Transient numbers 2 through 5 represent all other observed events with cycles extrapolated for the life of the plant. Transients 2 through 5 are applicable to both loop 1 and loop 4 lines. The observed outer wall temperature distributions from the monitoring data were utilized to develop these transients.

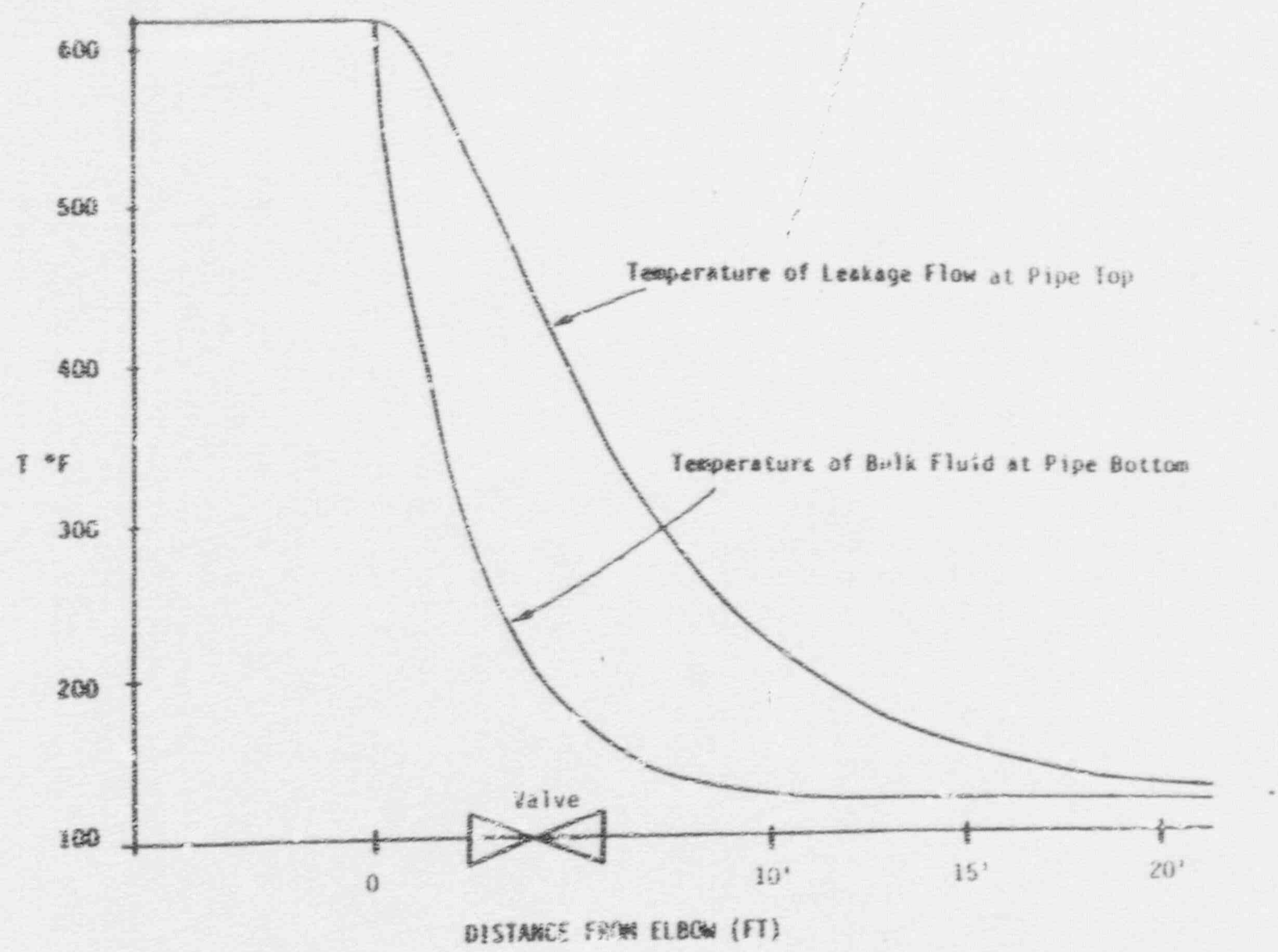
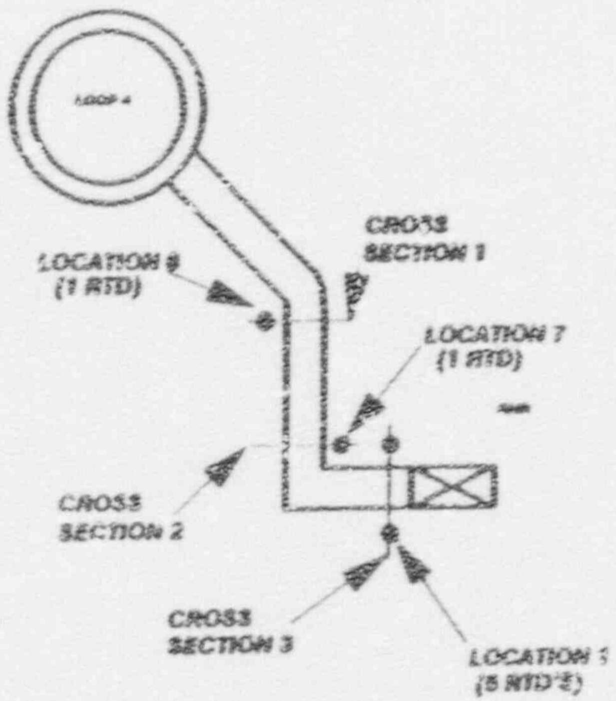


Figure 4-1. Temperature of Stratified Flow - Comanche Peak Units 1 and 2 12-inch RHR Line



POSTULATED LOAD CASES

CASE 1 (LINEAR TRANSITION AT SECTION 3)

CROSS SECTION 1		CROSS SECTION 2		CROSS SECTION 3	
LOOP SIDE	FAR SIDE	LOOP SIDE	FAR SIDE	TOP	BOTTOM
250	550	115	491	491	100

CASE 2

CROSS SECTION 1		CROSS SECTION 2		CROSS SECTION 3	
LOOP SIDE	FAR SIDE	LOOP SIDE	FAR SIDE	TOP	BOTTOM
115	618	115	420	100	100

CASE 3 (NON LINEAR TRANSITION AT SECTION 3)

CROSS SECTION 1		CROSS SECTION 2		CROSS SECTION 3	
LOOP SIDE	FAR SIDE	LOOP SIDE	FAR SIDE	TOP	BOTTOM
300	550	115	491	491	100

NOTE: 10% OF PIPE DIAMETER HOT WITH NON-LINEAR
TEMPERATURE DISTRIBUTION UNO

FIGURE 4-2

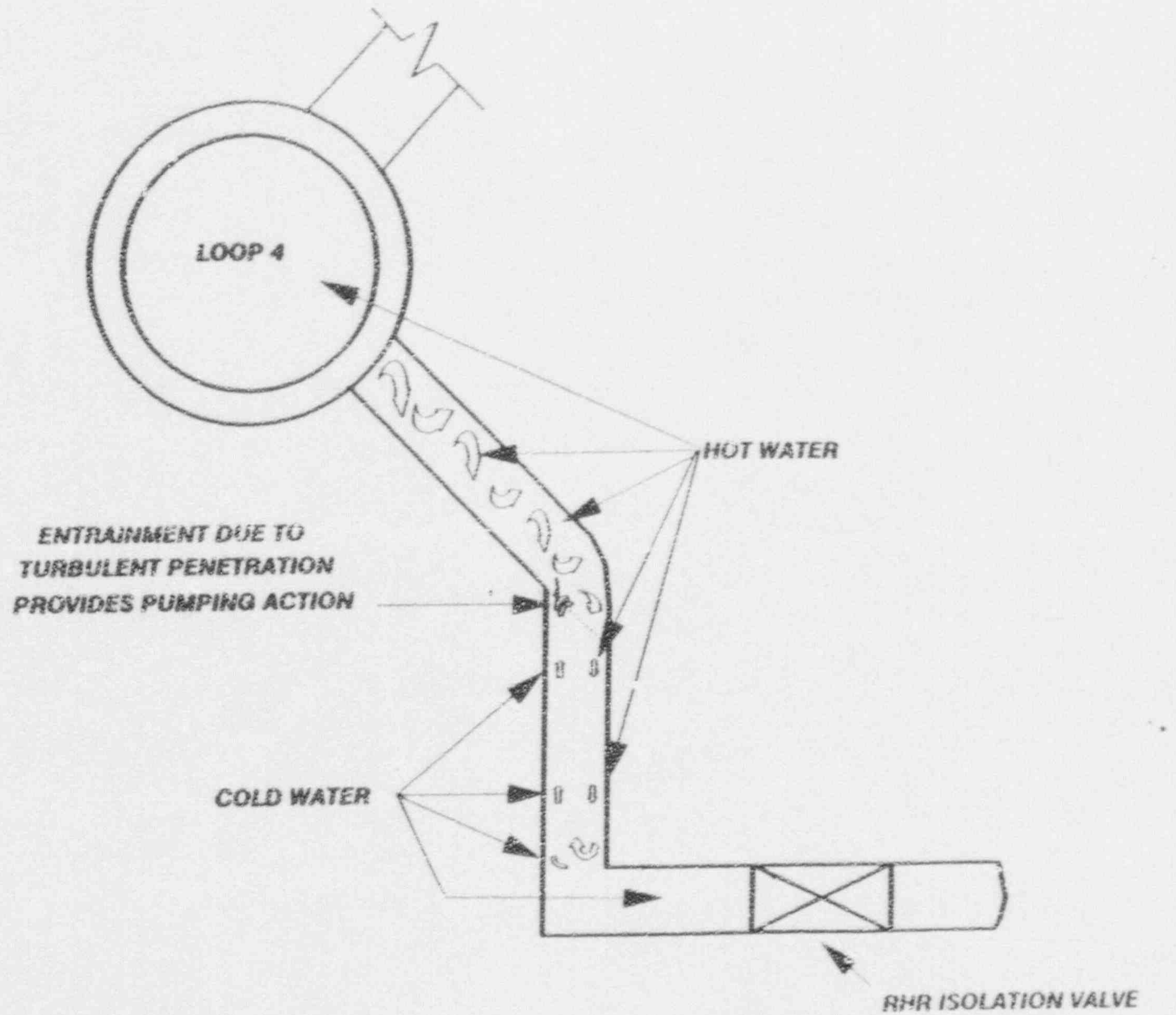
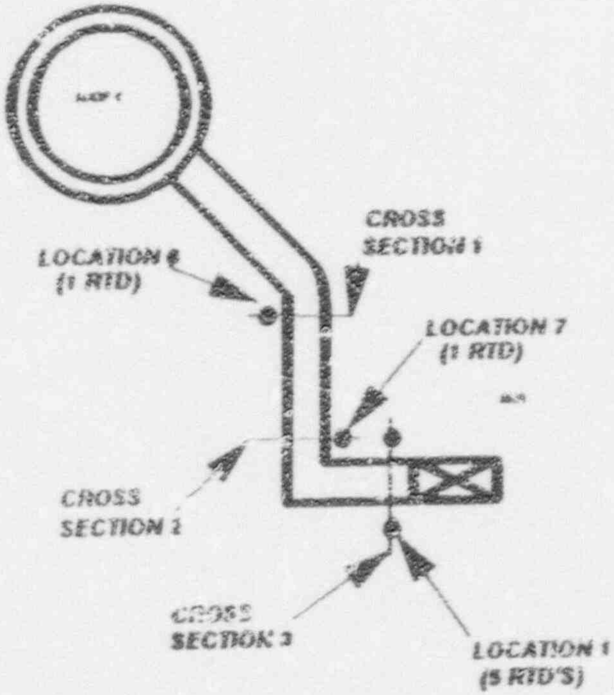

LOOP 4
HOT WATER
ENTRAINMENT DUE TO
TURBULENT PENETRATION
PROVIDES PUMPING ACTION
COLD WATER
RHR ISOLATION VALVE

FIGURE 4-3



POSTULATED VALVE LEAK CASE

CROSS SECTION 1 (MAX)		CROSS SECTION 2 (MAX)		CROSS SECTION 3 (MAX)	
LOOP SIDE	FAR SIDE	LOOP SIDE	FAR SIDE	TOP	BOTTOM
225	618	110	318	618	110

CROSS SECTION 1 (MIN)		CROSS SECTION 2 (MIN)		CROSS SECTION 3 (MIN)	
LOOP SIDE	FAR SIDE	LOOP SIDE	FAR SIDE	TOP	BOTTOM
225	618	110	481	310	110

NOTE: THE MAXIMUM CASE IS ASSUMED TO RANGE WITH THE MINIMUM CASE

PERIOD FOR EACH CYCLE = 60 MIN

FIGURE 4-4

5.0 SUMMARY AND CONCLUSIONS

A detailed evaluation of the residual heat removal suction lines for Comanche Peak Unit 1 has been completed in response to concerns raised by a pipe crack incident which occurred at Genkai Unit 1 in Japan and subsequent NRC Bulletin 88-08.

The monitored data from the Unit 1 RHR suction lines have been reviewed and evaluated. No NRC Bulletin 88-08 type of valve leakage was observed in the data. However, conservative assumptions were made to postulate a Genkai type of leakage in the Loop 4 RHR line. Based on such assumptions, the resulting stratification loading and associated stresses were calculated. Using these calculated stresses and postulated high number of stress cycles, conservative fatigue usage and fatigue crack growth calculations were then performed for Loop 4 RHR line. Loop 1 RHR suction line leakage has no impact on the unisolable portion of the piping as previously discussed in Section 4.1.

For loop 4 RHR line, fatigue usage calculation provides an indication of the probability of cracking and of the time required to initiate. Fatigue crack growth analysis was performed to determine the time required for a 60 percent through wall crack to occur based on the postulated transient stratification loading, as shown in Figure 4-4. The critical locations are the inlet weld of valve 8702 and the weld at the end of the 90° elbow. Due to the extremely conservative assumption in the fatigue usage calculation, a fatigue usage factor of less than 1 cound not be obtained within the plant design life at the governing location. Furthermore, results of this analysis indicate that a minimum of 1.5 years of leakage is required for an initial flaw of 10 percent wall thickness to propagate to 60 percent wall thickness. Augmented inservice inspection intervals should be developed based on this result of 1.5 years for both locations of loop 4 RHR line. All other welds in the loop 4 RHR line should be inspected in accordance with standard ASME Section XI criteria. The first augment inservice inspection should occur one and one-half effective full power years (EFPY) after August 31, 1991.

For loop 1 RHR line, since the postulated leakage has no impact on the unisolable portion of the piping, the fatigue usage calculated, based on only the operational transients described in Section 4.0, is 0.9 for 40 years of design life. Therefore, all welds in the loop 1 RHR line should be inspected in accordance with standard ASME Section XI criteria.

It is thus concluded that the requirements of NRC Bulletin 88-08, Supplement 3, are satisfied based on the following:

- Conservative technical evaluation provided in this report,

- Augmented inservice inspection intervals, and

- Implementation of the CPSES Unit 1 long-term transient and fatigue cycle monitoring program

6.0 REFERENCES

1. NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," 6/22/88; Supplement 1, 6/24/88; Supplement 2, 8/4/88; and Supplement 3, 4/11/89.

2. WCAP-12258, "Evaluation of Thermal Stratification for the Comanche Peak Unit 1 Residual Heat Removal Lines," W. H. Bamford, April 1989; Supplement 1, June 1989 and Supplement 2, August 1989, Westinghouse Proprietary.

3. Texas Utilities letter CPSES-9120542, 8/14/91, "Comanche Peak Steam Electric Station Thermal Monitoring Data Reduction," J. W. Muffet to J. L. Vota.