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EVALUATION OF THERMAL STRATIFICATION FROM POSTULATED VALVE LEAKAGE FOR THE COMANCHE PEAK UNIT 2 RHR LINES

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SECTION 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 (see Figure 1-1). 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.

An initial evaluation of the Comanche Peak 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 of Unit 1, 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 crack growth methodology, and assuming continuous 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, to evaluate the postulated valve leakage condition, and to provide recommendations for satisfying NRC Bulletin 88-08 requirements for Unit 2.

a,c,e

Figure 1-1. Sketch of the Cracking Location in the Genkai Unit 1 RHR Suction Line

SECTION 2.0
OVERALL EVALUATION APPROACH

2.1 General

TU Electric has placed temperature monitoring devices at several locations on the Unit 1 RHR suction piping to detect adverse thermal transients as described in Item 3 of Reference 1. After reviewing the monitoring results, which will be discussed in detail in Section 3.0, 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 the need for continuous monitoring on Unit 2.

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 conditions 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 factor evaluation considering postulated valve leakage, monitored transients and design transients
- Fatigue crack growth calculation considering postulated valve leakage, monitored transients and design transients
- Augmented ISI determination based on fatigue crack growth calculation.

Comanche Peak Unit 1 began commercial operation August 13, 1990. Comanche Peak Unit 2 has not yet begun commercial operation. 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 the Comanche Peak Units. Fatigue usage and fatigue crack growth have been calculated assuming that cyclic valve leakage occurs, resulting in stress cycles. Fatigue usage provides an indication of the probability of cracking to initiate. Fatigue crack growth provides a measure of the time required to propagate a crack to 60% of the wall thickness, assuming an initial crack size of 10% of the wall thickness. No credit has been taken for the time to initiate or propagate the crack to the initial crack size. Augmented inservice inspection intervals based on conservative fatigue crack growth calculations provides a strong technical justification to eliminate the need for continuous monitoring of the RHR piping, while still satisfying the requirements of NRC Bulletin 88-08.

SECTION 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)
5	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 was 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 200°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, and 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 [ ]^{a,c,e}. 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 branch pipes effect (heat transfer by a mass transport mechanism) to heat up 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 penetrations 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:

a,c,e

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

a,c,e

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

a,c,e

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

SECTION 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 [ ]^{a,c,e} in the unisolable side of the RHR line and hot water approximately [ ]^{a,c,e} 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 [ ]^{a,c,e} The total number of these events observed during the monitoring period was [ ]^{a,c,e} 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 [ ]^{a,c,e} and no significant cycling.

Transients that envelope these observed conditions are listed in Table 4-1. Transients [ ]^{a,c,e} are applicable to both loop 1 and loop 4 RHR lines. Transient [ ]^{a,c,e} to loop 4 only.

4.3 <u>Development of the Postulated Valve Leakage Transients</u>

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 connection, 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. [ ]^{a,c,e}

[ ]a,c,e as shown in Figure 4-1. This creates a rather large temperature differential between the top and bottom of the pipe, which maximizes at [

]a,c,e

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
[			
	]a,c,e		
[			
]a,c,e			

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

a,c,e

a,c,e

Figure 4-2

a,c,e

Figure 4-3

a,c,e

Figure 4-4

SECTION 5.0
STRESS ANALYSES

Flow diagram Figure 5-1 describes the procedure to determine the effects of thermal stratification on the RHR suction lines based on the transients developed in section 4.0 which included the postulation of valve leakage. [

]a,c,e

Section 5.1 Addresses the global structural effect of stratification

Section 5.2 Addresses the local stress effects due to the nonlinear portion of the temperature profile from stratification

Section 5.3 Addresses the stress from axisymmetric thermal shock transients

5.1 Piping System Global Structural Analysis

5.1.1 Introduction

The thermal stratification computer analysis of the piping system to determine the pipe displacement and support reaction loads as well as moment and force loads in the piping, is referred to as the piping system structural analysis. These loads are used as a part of the input to the fatigue and fatigue crack growth evaluations. The thermal stratification condition consists of both axial and top-to-bottom variations in the pipe metal temperature, as described in section 4.0. The model consists of straight pipe and elbow elements for the WESTDYN computer code. [

]a,c,e These studies verified the suitability of the WESTDYN computer code for the thermal stratification analysis. [

]a,c,e

[

]^{a,c,e}

5.1.2 Discussion

The piping layout for the RHR suction line analysis (loop 4) is shown in Figure 5-2. Since the analysis envelops all four RHR lines from both Comanche Peak Units, loop 4 of Unit 2 was selected conservatively for the single analysis. Such selection was based on a combination of the highest design thermal stresses and the largest effect from the thermal stratification among the four lines. The piping analysis model consists of straight pipe and elbows as shown in Figure 5-3. The elements provide the capability to load the piping with a top-to-bottom temperature gradient. Six thermal stratification loadings were applied to the WESTDYN structural model to determine the effects of cyclic valve leakage and operational transients defined in Section 4.0.

The first case represented stratification from valve leakage and assumed that hot leakage flowed, beginning at the second elbow from the loop, through the valve (8702) along the top of the piping for about [ ]^{a,c,e}, cooling as it flowed (see Figure 4-1). The other five cases representing stratification from operational transients, are summarized below:

a,c,e

[ ]

The analysis was based on these five stratification cases in addition to the stratification case from postulated valve leakage, along with the original design thermal cases.

For the WESTDYN code an [

]^{a,c,e}

5.2 Local Stress Due to Non-Linear Cross-Sectional Profile

5.2.1 Explanation of Local Stress

Figure 5-5 shows the local axial stress components in a beam with a sharply nonlinear metal temperature gradient. Local axial stresses develop due to the restraint of axial expansion or contraction. This restraint is provided by the material in the adjacent beam cross section. For a linear top-to-bottom temperature gradient, the local axial stress would not exist. [

]^{a,c,e}

5.2.2 Finite Element Model of Pipe for Local Stress

A representative pipe finite element model is shown in Figure 5-7, along with thermal boundary conditions. The entire cross section was used for modeling and analysis. [

]^{a,c,e}

[

]a,c,e

5.2.3 Local Stress Results

The temperature and stress results for the local finite element model are presented in the plots in Figures 5-8, 5-9 and 5-10. [

]a,c,e The high temperature region is very localized at the top of the pipe, as expected, and the pipe wall temperature quickly drops to the stagnant water temperature for the majority of the circumference. The axial stresses from this stratified flow are shown in Figure 5-9, and it can easily be seen that the highest stress is near the hot-cold water interface, and is positive. Compressive stresses are found at both the top and bottom of the pipe. The stress intensity (Figure 5-10) was highest at the top of the pipe.

5.3 Thermal Shock Transient Stresses from Plant Design Conditions

The thermal shock transient stresses are also a necessary part of the stress data that are required for the fatigue evaluation. Such stresses, caused by temperature change in the pipe due to hot or cold transients, were calculated based on the original plant design condition [Reference 4]. The plant design specification gives fluid system pressure, temperature, and shock transients. The shock transient causes time-varying temperature distributions across the pipe wall. These temperature distributions result in pipe wall stresses (across the thickness) in three parts: a uniform, a linear and a non-linear portion. The uniform portion provides general expansion loads, the linear portion causes bending moment loads and the non-linear portion yields skin stress. In the ASME Code stress calculation, these are commonly referred as ΔT_1 and ΔT_2 through wall stresses.

5.4 Total Stresses

For the thermal stratification loading, the previously mentioned global stress, and local stress, can be combined by the method of superposition to account for the total effect of the stratification. Figure 5-6 presents the results of a test case that was performed to demonstrate the validity of the superposition. As shown in the figure, the superposition of local and global stresses is valid. The stress states caused by thermal stratification and by axisymmetric shock transients can then be combined to find the stress ranges that are required for fatigue usage factor evaluation.

a,c,e

Figure 5-1. Determination of the Effects of Thermal Stratification

a,c,e

Figure 5-2. Piping System Isometric Drawing - RHR Loop 4

a,c,e

Figure 5-3. Finite Element Model of the RHR Line Piping, Loop 4

a,c,e

Figure 5-4. [ ]^{a,c,e} Profile

a,c,e

Figure 5-5. Local Stress in Piping Due to Thermal Stratification

a,c,e

Figure 5-6. Test Case for Superposition of Local and Structural Stresses

a,c,e

Figure 5-7. Piping Local Stress Model and Thermal Boundary Conditions

a,c,e

Figure 5-8. RHR Line Temperature Distribution at Maximum Temperature Difference Location

a,c,e

Figure 5-9. RHR Line Local Axial Stress Distribution at Maximum Temperature Difference Location

a,c,e

Figure 5-10. RHR Line Local Stress Intensity Distribution at Maximum Temperature Difference Location

SECTION 6.0
FATIGUE USAGE FACTOR EVALUATION

6.1 Code and Criteria

Fatigue usage factors for the Comanche Peak Unit 2 RHR suction lines were evaluated using the concept of the ASME B & PV Code, Section III, Subsection NB-3600 (1979 Edition), for piping components. The fatigue evaluation required for level A and B service limits in NB-3653 is summarized in Table 6-1. ASME III fatigue usage factors were calculated for each type of component between the hot leg branch nozzle and the second isolation valve where the piping was most affected by the stratification.

6.2 Previous Design Methods

Previous evaluations of RHR suction line piping fatigue used the NB-3653 techniques but with thermal shock transients defined by Westinghouse design specifications, assuming the fluid flows to sweep the RHR line piping with an axisymmetric temperature loading on the pipe inside wall, and that no stratified flow due to postulated valve leakage exists. Those evaluations produced typical usage factors of approximately [ ]^{a,c,e} at elbows and bends,
[ ]^{a,c,e} at valve ends, and [ ]^{a,c,e} at the RCL hot leg nozzle safe end.

It must be noted that these usage factors are conservative since, in the design process, calculations are carried to the point where results meet code requirements, and are not further refined to reduce the usage factor.

6.3 Analysis with Thermal Stratification

With the thermal transients redefined to account for thermal stratification due to valve leakage and operational related transients, as described in section 4.0, the stresses in the piping components were established (section 5.0) and new fatigue usage factors were calculated. Due to the non-axisymmetric nature of the stratification loading, stresses due to all other loadings such as axisymmetric thermal from design transients were then combined.

Stresses in the pipe wall due to thermal stratification loadings were obtained from the WECAN 2-D analysis of a 12 inch, schedule 140 pipe. The total stress for the transient load case was then obtained by superposition of these stresses with the global stresses which account for the hot-cold interface level.

Two types of stress were calculated - S_n (Eq 10), to determine elastic-plastic penalty factors, K_e, and S_p (Eq 11) - peak stress. For most components in the RHR line (girth butt welds, elbows, bends) no gross structural discontinuities are present. As a result, the code-defined "Q" stress (NB-3200), or $C_3E|\alpha_a t_a - \alpha_b t_b|$ in Eq (10) of NB-3600 is zero. Therefore, for these components, the Eq. (10) stresses are due to pressure and moment.

Peak stresses, including the total surface stress from all loadings - pressure, moment, stratification - were then calculated for the transients. [

]^{a,c,e}

This evaluation used the S_n and S_p stresses calculated for each transient to determine usage factors at selected locations in the pipe cross section. Using a standard ASME method, the cumulative damage calculation is performed according to NB-3653.

This includes:

1) Calculating the S_n and S_p ranges, K_e, and S_{alt} for every combination of the transient loads.

2) For each value of S_{alt}, use the design fatigue curve to determine the maximum number of cycles which would be allowed if this type of cycle were the only one acting. These values, N_1, N_2...N_n, were determined from Code Figures I-9.2.1 and I-9.2.2, for austenitic stainless steels.

3) Using the actual cycles of the transient loadset, calculate the contribution to the usage factor U_i, from the postulated stratification. If N_i is greater than 10^{11} cycles, the value of U_i is taken as zero.

4) The cumulative usage factor, U_{cum}, is calculated as $U_{cum} = \sum_{i=1}^{n} U_i$

The code allowable value is 1.0.

6.4 Fatigue Usage Results

A stress analysis was completed for the stratified flow condition, including local stresses and global stresses resulting from stratification. Deadweight stresses were constant, so they were not included since they would not contribute to the alternating stress. The crite ia used are shown in Table 6-1. The analysis was performed on loop 4 of Unit 2 which was determined to have the highest stresses.

[

]a,c,e

TABLE 6-1
CODE/CRITERIA

- ASME B&PV Code, Sec. III, 1986 Edition
  - NB3600
  - NB3200
- Level A/B Service Limits
  - Primary Plus Secondary Stress Intensity $\leq 3S_m$ (Eq. 10)
  - Simplified Elastic-Plastic Analysis (when Eq. 10 > $3\ S_m$)
  - Expansion Stress, $S_e \leq 3S_m$ (Eq. 12) - Global Analysis
  - Primary Plus Secondary Excluding Thermal Bending < $3S_m$ (Eq. 13)
  - Elastic-Plastic Penalty Factor $1.0 \leq K_e \leq 3.333$
  - Peak Stress (Eq. 11)/Cumulative Usage Factor (U_{cum})
  - $S_{alt} = K_e S_p/2$ (Eq. 14)
  - Design Fatigue Curve
  - $U_{cum} \leq 1.0$

SECTION 7.0
FATIGUE CRACK GROWTH EVALUATION

7.1 General

Per the previous section, it was shown that, should a postulated mechanism exist to induce cycling, crack initiation could occur. This section deals with the time required for crack propagation, and consequently, the determination of augmented inservice inspection intervals.

7.2 Method Description

The ASME Section XI method is based on stress analysis results and material crack growth laws. The stress intensity factor (K_I) required for the fatigue crack growth calculations is obtained from the (K_I) expression given in reference 7 for an aspect ratio (2a/l) of 0.167. The fatigue crack growth law for stainless steel in a pressurized water environment was obtained from reference 8. The crack growth in inches per cycle is

$$da/dn = (C)(F)(S)(E)\ \Delta K^{3.30}$$

where: C = 2.42×10^{-20}

F = frequency factor ($F = 1.0$ for temperatures below 800°F)

S = minimum K to maximum K ratio correction ($S = 1.0$ for $R = 0$; $S = 1 + 1.8R$ for $0 < R < 0.8$; and $S = -43.35 + 57.97R$ for $R > 0.8$)

E = environmental factor. $E = 1.0$*

ΔK = range of stress intensity factor, psi $\sqrt{\text{in}}$

R = the ratio of the minimum K_I to the maximum K_I

* A compilation of data for austenitic stainless steels in a PWR water environment was made by Bamford (Reference 5), and it was found that the effect of the environment on the crack growth rate was very small. For this reason it was estimated that the environmental factor should be set at 1.0 in the crack growth rate equation.

The stress intensity range input to the fatigue crack growth analysis [

]^{a,c,e}

[

]^{a,c,e} Stresses were obtained from transient thermal and stress analyses of a 2-D WECAN finite element model.

7.3 Fatigue Crack Growth Results

For Loop 4, 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 Table 4-1 and Figure 4-4. The critical locations are [

]^{a,c,e} Results of this analysis indicate that a minimum of [

]^{a,c,e} All other welds in the Loop 4 RHR line should be inspected in accordance with standard ASME Section XI criteria.

For the loop 1 RHR line, since the 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, fatigue crack growth calculation was not performed.

SECTION 8.0
SUMMARY AND CONCLUSIONS

A detailed evaluation of the residual heat removal suction lines for Comanche Peak Unit 2 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 the 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. The critical locations are [ ]^{a,c,e} Due to the extremely conservative assumption in the fatigue usage calculation, a fatigue usage factor of less than 1 could 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 on the loop 4 RHR line. All other welds in the loop 4 RHR line should be inspected in accordance with standard ASME Section XI criteria.

For the loop 1 RHR line, since the 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 2 long-term transient and fatigue cycle monitoring program

SECTION 9.0
REFERENCES

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