Controlled Copy No. 7

XTV-02-008
Revision 0
August 1983
File Number:
165.1202.0100

STRUCTURAL INTEGRITY ASSESSMENT OF BROWNS FERRY FEEDWATER NOZZLES

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Date: August 9, 1983

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REVISION CONTROL SHEET

TITLE: Structural Integrity Assessment of Browns Ferry Feedwater Nozzles

DOCUMENT FILE NUMBER: 165.1202.0100 Revision 0

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AFFECTED PAGE(S)	DOC REV	PREPARED BY / DATE	ACCURACY CHECK BY / DATE	CRITERIA CHECK BY / DATE	REMARKS
i - v	0	HLG/8-4-83	LCH 8/9/83	LCH 8/9/83	
1.1 through 1.3					
2.1 through 2.4					
3.1 through 3.5					
4.1 through 4.6					
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ABSTRACT

The Browns Ferry Nuclear Power Station reactor vessel feedwater nozzles were evaluated analytically to determine the rate of fatigue damage accumulation for a range of thermal sleeve seal leakage rates. Results are presented which correlate leakage rate with fatigue usage factor.

The operation of the feedwater system at low flow rates was studied to determine the extent to which low flow/unsteady flow-induced thermal cycling affects the feedwater nozzles. The conclusion is that present Browns Ferry equipment and operating procedures are adequate to minimize the crack growth concerns of NUREG-0619.

A proposed Reactor Water Clean-Up system cross-tie modification was evaluated as a possible method of mitigating thermal cycling at the feedwater nozzles. The modification was shown to have no significant effect on fatigue due to thermal cycling.

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1.0 INTRODUCTION

1.1 Background

In the late 1970's, Tennessee Valley Authority (TVA) replaced the original-equipment loose fit thermal sleeve feedwater spargers at Browns Ferry Nuclear Power Plant with interference fit thermal sleeve spargers, and later with triple sleeve spargers. The purpose of these modifications is to minimize feedwater leakage past the thermal sleeve, and thereby minimize thermal cycling in the vicinity of the feedwater nozzle. Rapid thermal cycling-induced fatigue is a major cause of feedwat nozzle cracking. Rapid cycling fatigue damage is a strong function of feedwater leakage rate past the thermal sleeve seals.

Thermocouples are being installed on the outside of the feedwater lines at the junctions with the feedwater nozzles. These thermocouples will be used to monitor feedwater leakage past the thermal sleeve interference seals. By monitoring leakage versus time, TVA is taking action to anticipate any cracking occurrence.

A second fatigue process which may affect the service life of feedwater nozzles is that of low frequency

temperature cycling at the feedwater nozzles due to unsteady operation of the feedwater system at very low power levels. Although low frequency cycling does not contribute significantly to crack initiation in BWR feedwater nozzles, this phenomenon can have a strong effect on crack growth. TVA has collected temperature data in the feedwater lines immediately upstream of the thermal sleeve seal to determine the extent of low frequency thermal cycling experienced by the feedwater nozzles.

1.2 Objectives

The NUTECH scope of work on the Browns Ferry feedwater nozzle evaluation program consists of four tasks.

1. Evaluation of feedwater nozzle modifications performed on Browns Ferry Units 1, 2, and 3. NUTECH has evaluated the effects of rapid thermal cycling-induced fatigue on the feedwater nozzles for plant-specific operational characteristics and configurations. In addition, fatigue effects due to system cycling have been studied. Predictions of the effects of thermal sleeve seal bypass leakage on feedwater nozzle fatigue life have also been developed.

2. <u>Study of low flow controller requirements</u>.

NUTECH has studied the details of Browns Ferry Feedwater System operation at low flows, to determine the effects of low frequency thermal cycling due to system operating characteristics.

3. <u>Reactor water clean-up crosstie study</u>.

NUTECH has evaluated the effectiveness of a proposed Reactor Water Clean-Up Crosstie modification in reducing nozzle fatigue.

4. <u>Feedwater nozzle inservice inspection requirements</u>.

NUTECH has prepared predictions of fatigue usage vs time for several postulated bypass leakage rates. Since nozzle cracking is traceable to thermal fatigue which in turn is related to leakage rate, continuous monitoring of leakage coupled with a fatigue/leakage correlation such as presented herein should provide an acceptable alternative to in vessel nozzle inspection. However, since the leakage monitoring system is not yet operational at Browns Ferry as of the date of this report, the results presented are parametric in nature, rather than quantitative.

2.0 DISCUSSION OF THERMAL CYCLING IN FEEDWATER NOZZLES

2.1 Types of Thermal Cycling

In the early 1970's, Boiling Water Reactors world-wide began to experience problems with cracking of feedwater nozzles and related components such as safe-ends. Cracks in the nozzles, and to a lesser extent the safe ends, are traceable to fatigue derived from thermal and pressure cycling. Three different types of cycling have been shown to be significant contributors to the cracking problem. These are:

1. System Cycling: This category includes major operational transients such as start-ups, shut downs, etc. The fatigue effects of these transients were considered in the original feedwater nozzle stress report (Reference 5).

2. Rapid Thermal Cycling: This type of thermal cycling is caused by the turbulent mixing of hot reactor water with relatively cold feedwater during steady state operation. The feedwater nozzle blend radius has historically been susceptible to fatigue caused by cycling of this sort. Feedwater leakage which bypasses the thermal sleeve produces signifi-

cant thermal cycling in the vicinity of the nozzle, which has been shown (Reference 2) to be the dominant cause of crack initiation in the feedwater nozzle.

3. Unstable/Low Frequency Thermal Cycling: This phenomenon is experienced primarily at low power and hot standby conditions, when the feedwater system is operated in an unsteady manner. When a plant is in hot standby, some steaming still occurs due to decay heat. As a result, although the full flow capabilities of the feedwater system are not required, it is occasionally necessary to add water to maintain reactor water level. In addition, since the feedwater heaters are usually not available during such operation, the feedwater temperature is low. The cycling occurs as the feedwater nozzles, which contain essentially stagnant reactor water (high temperature), are flushed with cold feedwater, at a frequency of a few cycles per hour.

   Low frequency cycling of this type has been shown (Reference 2) to have minimal effect on crack

initiation (that is, it contributes little to fatigue usage factors), but to have appreciable effect on crack growth.

2.2 Generic Mitigation of Fatigue Effects

The effects of system cycling are unchangeable without making major operational changes. Consequently, the designers and owners of affected Boiling Water Reactors have emphasized mitigation of the effects of rapid and unsteady (low frequency) thermal cycling in their responses to feedwater nozzle cracking.

Leakage of feedwater past the thermal sleeve is the principal cause of rapid cycling. New thermal sleeve designs have been developed to minimize such leakage. TVA has installed double piston ring seal, triple thermal sleeve/sparger equipment to replace their original loose-fit equipment. In addition, thermocouples are being installed on the outside surface of the feedwater line at the pipe-to-nozzle junction. Experience has shown that feedwater leakage past the thermal sleeve seals can be detected and quantified by monitoring feedwater nozzle temperatures in such a manner. Significant leakage past the thermal sleeve seals can be anticipated in time to prevent accumulation

of excessive rapid thermal cycling-induced fatigue damage by use of a leakage monitoring system of this type.

The extent of low frequency thermal cycling is determined by the degree to which the feedwater flow can be accurately controlled at extremely low flow rates (e.g., 1-10%). Reviews of the flow controller equipment at several operating U.S. plants have shown a wide disparity in the acceptability of feedwater low flow controller equipment. Some plants require no modifications whatsoever, while others require extensive equipment and operational changes to minimize the concerns above.

3.0 COMPONENT DESCRIPTION

3.1 Geometry

Figure 3-1 presents the detailed nozzle/safe-end/thermal sleeve configuration which was analyzed. The nozzle is SA-508 Class 2, low alloy steel, and the safe end is SA-105, Grade II carbon steel. A triple thermal sleeve is inserted in the nozzle.

3.2 Material Properties

Minimum required ASME Section III material properties for the above materials are assumed to apply. In addition, due to the high frequency nature of the rapid cycling loading condition, the ASME Section III design fatigue curve for carbon and low alloy steels was extended (Figure 3-2).

3.3 Functional Description

The function of the feedwater nozzle/sparger system in BWRs is to introduce and distribute relatively cold feedwater into the reactor. The nozzle penetrates the reactor pressure vessel just above the active core level. The sparger serves the dual purpose of distrib-

uting the feedwater uniformly around the periphery of the reactor and mitigating the effects of thermal stresses in the nozzle caused by the introduction of the cold water. The Browns Ferry nozzle/sparger design addressed in this report incorporates a seal arrangement which is used to minimize the potential for bypass leakage of the cold feedwater around the thermal sleeve.

3.4 Loading Conditions

3.4.1 Design Loads

The design pressure and temperature for the nozzle are 1250 psig and 575°F. Normal operational pipe reaction loads on the nozzle, including seismic loads are given in Reference 1.

3.4.2 Operating Conditions

The normal operating pressure and temperature for the nozzle are approximately 1000 psig and 546°F, respectively.

3.5 Analysis Criteria

The criteria used in this report for stress analysis of the Browns Ferry feedwater nozzles are the stress limitations of the ASME Boiler and Pressure Vessel Code, Section III, Article NB-3000 (Reference 3).

An estimated design life was determined using information describing system operational conditions and thermal cycling provided by TVA (Reference 4). In addition, fatigue usage was also determined on a per operational condition basis for various assumed seal leakage rates. This permits TVA to monitor the actual system operational conditions and thermal sleeve seal leakage rates as described in Section 4.0 of this report. The governing criterion for establishing fatigue design life is that the ASME Code allowable fatigue usage factor of 1.0 may not be exceeded during the design lifetime.

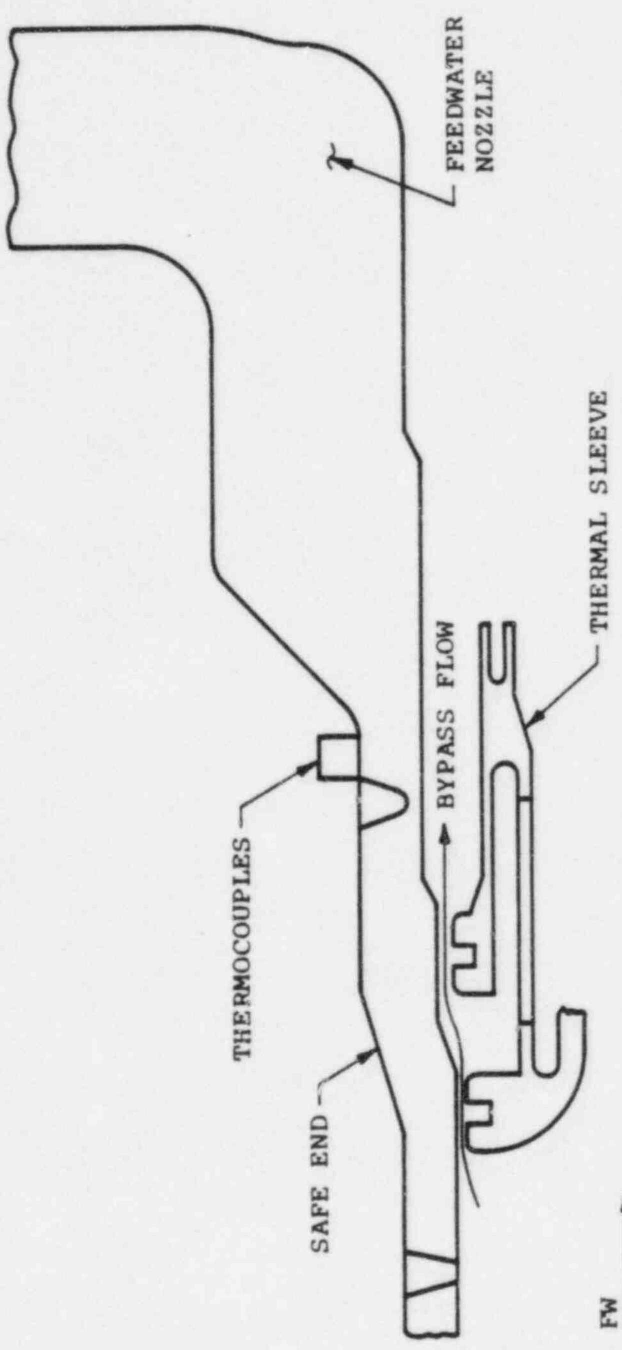

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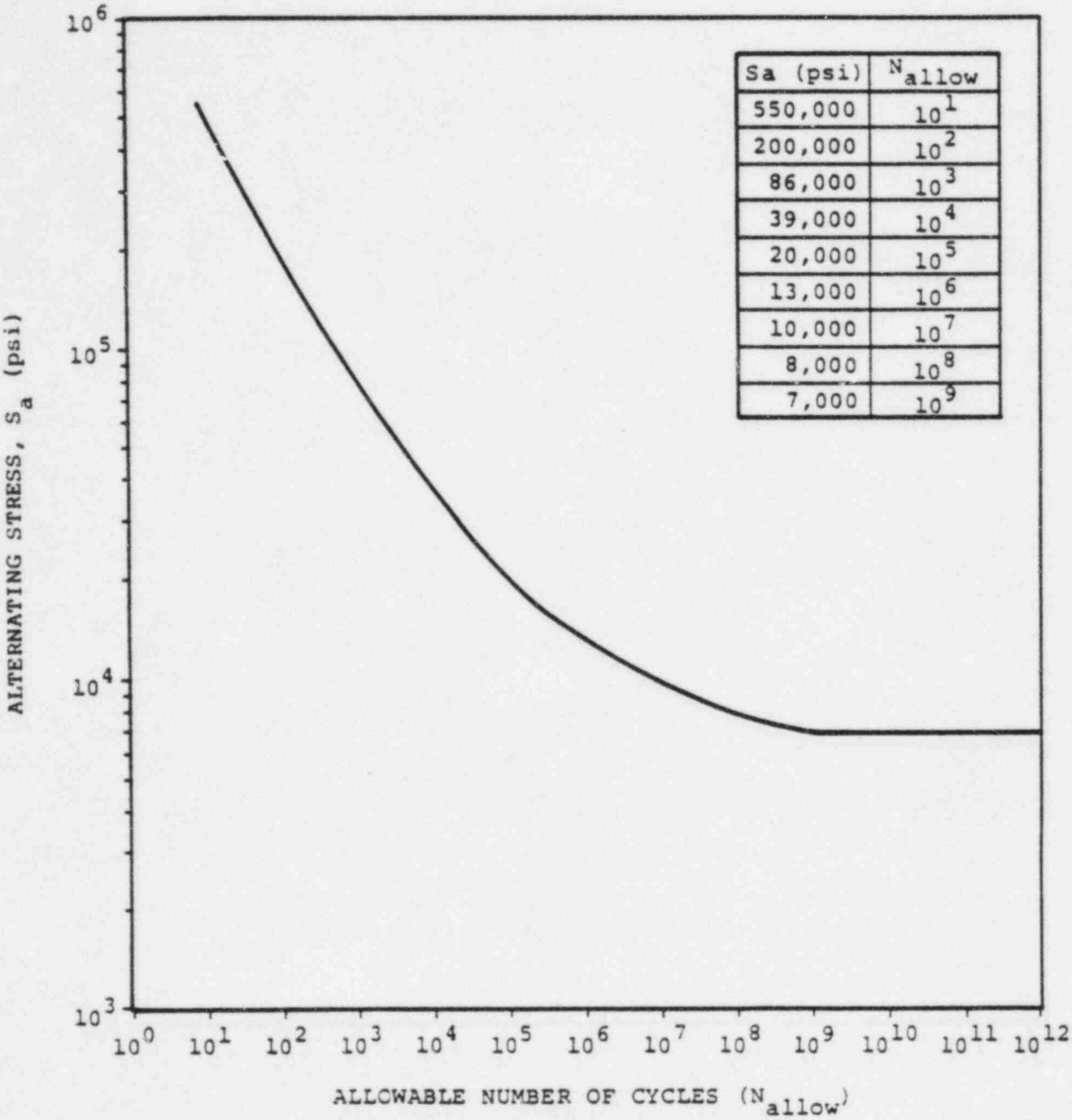
Figure 3-1

FEEDWATER NOZZLE/THERMAL SLEEVE CONFIGURATION



Sa (psi)	N_{allow}
550,000	10^1
200,000	10^2
86,000	10^3
39,000	10^4
20,000	10^5
13,000	10^6
10,000	10^7
8,000	10^8
7,000	10^9

Figure 3-2

ASME SECTION III DESIGN FATIGUE CURVE EXTENDED BEYOND 10^6 CYCLES (CARBON/LOW ALLOY STEEL)

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4.0 LEAKAGE RATE CORRELATION

4.1 Introduction

The rate of accumulation of fatigue usage at the feedwater nozzles is related to the rate of feedwater leakage past the thermal sleeve seals. Leakage will also affect local temperatures, since the feedwater temperature is much lower than reactor water temperature. If there is no seal leakage, there will be some difference between top and bottom temperatures within the thermal sleeve annulus, due to natural thermal stratification. When the seals are leaking, the difference will be appreciably greater because of the difference in density between hot reactor water and much cooler feedwater (leakage) water. That is, because of the density gradient, the cooler leakage water will tend to collect at the bottom of the annulus.

To determine the actual seal leakage rate through the feedwater thermal sleeve seals, TVA is installing a leakage monitoring system (LMS) on the three Browns Ferry units. By comparing the LMS data with analytical predictions, an assessment of actual seal leakage and its effects on the feedwater nozzles can be made.

4.2 Analytic Procedures

Generic finite element models of feedwater nozzle configurations similar to Browns Ferry's have been developed using the ANSYS computer code. Analyses based upon these models were used to generate information relating temperature predictions (for leaking and non-leaking cases) to axial distance from the thermal sleeve seals (Figure 4-1).

The outside surface temperatures calculated by the analysis should correspond to thermocouple-measured temperatures at the thermocouple locations. The analytically predicted temperatures were normalized as follows:

$$T_N = \frac{T_{Nodal} - T_{FW}}{T_{Reactor} - T_{FW}} \qquad (4\text{-}1)$$

where:

T_N = Normalized temperature

T_{FW} = Feedwater temperature

$T_{Reactor}$ = Reactor temperature

T_{Nodal} = Normalized Analytical Prediction

For these calculations, the values

$$T_{FW} = 300°F$$

$$T_{Reactor} = 550°F$$

were used for comparison and numerical convenience. Although these values do not exactly correspond to actual operating conditions, the specific values used will have no actual effect on T_N as long as the same values for T_{FW} and $T_{Reactor}$ are used in each calculation.

The calculated values of T_N are plotted against axial distance from the seal in Figure 4-1. These curves represent the analytical predictions for normalized external temperatures for each combination of azimuthal location (top, bottom) and leakage (0.0, 1.5 GPM).

4.3 Use of Field Leakage Monitoring Data

TVA will be taking data with the Leakage Monitoring System following installation. This data will be used by NUTECH to calibrate the system for leakage measurement.

From review of the analytical curves of Figure 4-1 it is apparent that the temperature measured on the bottom of the nozzles is far more sensitive to leakage than is the top temperature. That is, the distance between the 0.0 GPM leakage and 1.5 GPM leakage curves for the nozzle bottom is much greater than for the nozzle top. As a result, the bottom temperature is a more responsive indication of leakage.

To determine the actual value of leakage rate, the band of field data for each nozzle is compared with the analytical predictions. This is done by indicating the normalized top and bottom temperature bands on a curve such as Figure 4-1 at the axial location of the thermocouples. The mean bottom normalized temperature is interpreted approximately as seal bypass leakage rate by interpolation between the zero and 1.5 GPM curves as follows:

$$\text{Leakage (GPM)} = \frac{T_o - T_{LMS}}{T_o - T_{1.5}} \times 1.5 \text{ GPM} \qquad (4\text{-}2)$$

where:

T_o = normalized bottom zero leakage temperature.

T_{LMS} = mean normalized bottom temperature as measured by Leakage Monitoring System.

$T_{1.5}$ = normalized bottom 1.5 GPM leakage temperature.

By the method described above, estimates of present seal leakage for each Browns Ferry feedwater nozzle may be calculated.

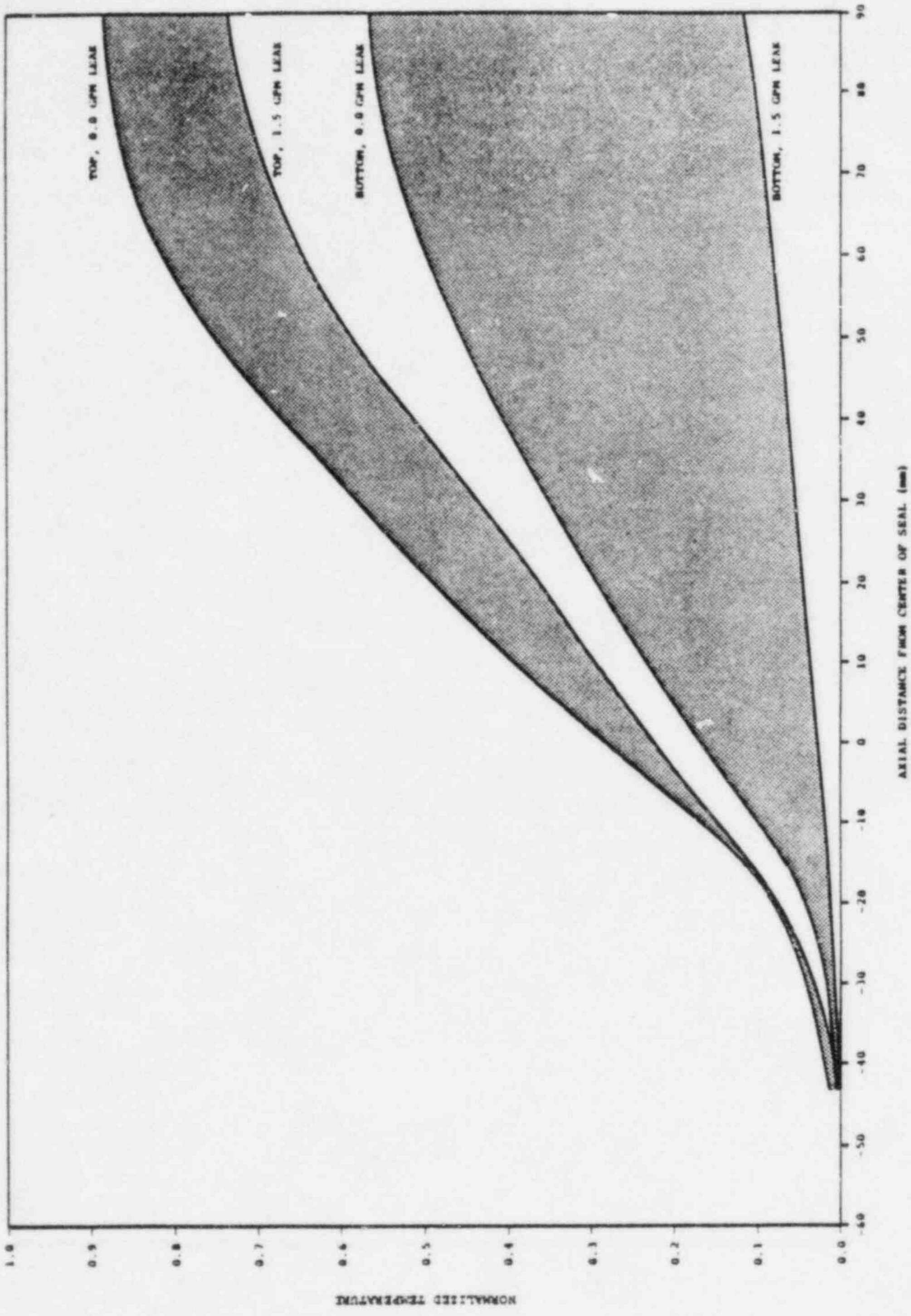

Figure 4-1

ANALYTICAL PREDICTIONS FOR NORMALIZED EXTERNAL TEMPERATURES

5.0 FEEDWATER NOZZLE STRUCTURAL INTEGRITY ASSESSMENT

5.1 Introduction

Crack initiation in BWR feedwater nozzles has been demonstrated (Reference 2) to be primarily a fatigue process. This section describes the evaluation of the fatigue effects on the feedwater nozzle due to rapid thermal cycling, system cycling, and unsteady flow cycling. This evaluation is based upon a review of Browns Ferry operational data (Reference 4) and the revised stress reports which describe the new thermal sleeve configuration (Reference 5).

5.2 Plant Specific Thermal Duty Map

A plant specific Thermal Duty Map (TDM) for Browns Ferry was developed based upon a review of plant operating records contained in Reference 4. Browns Ferry Unit 1 Cycle 4 was selected for study as a particularly severe cycle, based upon recommendations of plant personnel. This map describes the average hours per year spent in various operating regions, defined by power level and corresponding reactor and feedwater temperatures. The data were divided into 27 regions. Table 5-1 presents the resulting Plant Specific Feedwater Duty Map.

Regions 1 through 19 were developed based solely upon Browns Ferry data. Regions 20 through 26 have been added to Regions 1 through 19 in accordance with the General Electric Generic Feedwater Duty Map (Reference 5) to account for periods of reactor power maneuvering. Region 27 represents periods of cold shutdown.

5.3 Rapid Temperature Cycling

5.3.1 Description of Rapid Temperature Cycling Phenomena

Rapid temperature cycling (on the order of 0.1 Hz to 1.0 Hz) occurs at the nozzle inside surface due to mixing of hot reactor water and cold feedwater. The most dominant cause of this cycling is bypass leakage of feedwater past the thermal sleeve seals. However, rapid cycling can also occur in the absence of such leakage due to the mixing of hot reactor water and a cold water boundary layer which builds up on the outside surface of the thermal sleeve.

The magnitude of the nozzle surface thermal cycling which occurs due to this rapid thermal mixing phenomenon is given by the following expression, which is derived from experimental data presented in Reference 2.

$$\Delta T_{p-p} = \bar{A} \times C_3 \times C_4 \times (T_R - T_{FW}) \quad (5\text{-}1)$$

where:

ΔT_{p-p}	=	Metal surface peak to peak temperature range.
$\bar{A}$	=	Amplitude coefficient for a given frequency of cycling, from Table 5-2.
C_3	=	Coefficient from Table 5-3.
C_4	=	Leakage coefficient from Table 5-4.
T_R	=	Reactor water temperature.
T_{FW}	=	Feedwater temperature.

In order to determine the number of cycles at various peak-to-peak temperature ranges from equation 5-1, it is also necessary to know the amount of time which is spent at various feedwater temperatures, reactor temperatures and feedwater flow rates. This information is contained in the Browns Ferry thermal duty map described above.

5.3.2 Fatigue Usage Calculation

A computer program (DAMSUM) has been developed to evaluate the rapid cycle fatigue usage factors for BWR feedwater nozzles.

A series of constant leakage rates is input to the program. Using this data the program determines ΔT_{p-p} (Equation 5-1) for each flow map region using the appropriate feedwater flow rate for the region.

Alternating stress is then determined for each value of ΔT_{p-p} using the following equation:

$$S_a = \frac{E\alpha\Delta T_{p-p}}{2\ (1-\nu)} \qquad (5\text{-}2)$$

where:

ΔT_{p-p} = Peak-to-Peak Temperature Range from Equation 5-1

E = Young's Modulus

α = Coefficient of Thermal Expansion

ν = Poisson's Ratio

The properties E and α are determined in the program from tabulated ASME Code values at the average temperature about which the cycling occurs.

For incorporation into the DAMSUM program, the fatigue curve was linearized into straight line log-log segments as follows:

$$\log_{10} N = C - B \log_{10} S_a \qquad (5\text{-}3)$$

where

$$B = \frac{\log_{10} N_n - \log_{10} N_{n+1}}{\log_{10} S_{n+1} - \log_{10} S_n}$$

$$C = \log_{10} N_n + B \log_{10} S_n$$

where N_n, N_{n+1}, S_n and S_{n+1} are values of S_a and N at the beginning and end of the log decade surrounding the calculated value of S_a (See Table in Figure 3-2).

The allowable number of cycles (N_{allow}) can thus be determined for each calculated value of σ_{alt} from the fatigue curve, and the cumulative fatigue usage factor due to rapid cycling is computed versus time using the following multiple summation:

$$U = \sum_{i=1}^{I} \left(\sum_{j=1}^{J} \sum_{k=1}^{K} N_{jk}/N_{Allow\ jk} \right)_i \qquad (5\text{-}4)$$

where:

N_{jk} = Applied number of cycles for the ith amplitude and frequency in Table -2

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and for the kth flow map region in Table 5-1.

$N_{allow,\ jk}$ = Allowable number of cycles for S_{alt} for the jth amplitude and frequency in Table 5-2 and for the kth flow map region in Table 5-1.

$(\quad)_i$ = Usage factor for the ith year of reactor operation.

Results of representative rapid thermal cycle-induced fatigue calculations as described above are contained in Tables 5-5, 5-6, and 5-7. Figure 5-1 shows the relationship between rapid cycle fatigue usage and leakage rate predicted for Browns Ferry.

The above fatigue usage calculations assumed a series of constant leakage rates. In reality, leakage can be expected to increase gradually with time due to corrosion of seal and pipe materials.

A series of calculations was performed in which an initial interference fit and corrosion rates were assumed, and leakage was allowed to increase based upon mean gap calculations performed by program DAMSUM.

Fatigue usage per year was calculated based upon predicted leakage for that year. Cumulative usage was also calculated.

5.3.3 Rapid Cycle Evaluation for Calculated Leakage

The rate of accumulation of rapid cycle fatigue usage for increasing leakage rates is shown in Table 5-8. This information, together with the system cycling values determined in Section 5.5 predicts the current fatigue accumulation situation. If the leakage rate changes with time, the revised fatigue usage can be determined by entering the curve of Figure 5-1 at the calculated leakage rate to determine the associated rapid thermal cycling-induced fatigue usage. This value can be used with the system cycling value as described above.

Contribution of each thermal duty region to total fatigue usage factor is shown in Table 5-9 for a representative case. For different leakage rates, the magnitude of the individual contributions will vary, but the relative importance of each region will remain substantially the same, as long as the calculated stresses are above the endurance limit. When stresses

fall below the endurance limit, no fatigue usage contribution is expected. By inspection of tables of this type, those operating regimes which contribute most strongly to rapid cycle fatigue damage can be identified.

It should be noted that in Figure 5-1 the slope of the fatigue usage accumulation curve is shallow at low leakage rates. A small increase in leakage will produce only a small increase in fatigue usage for low leakage rates. In the vicinity of 1 GPM leakage however, the local fatigue curve is much steeper. A small leakage increase in this region can produce an appreciable increase in fatigue usage. This comment is presented for use in planning repairs, etc.

Leakage past triple thermal sleeve-type seals has historically been very low. Leakage is expected to increase gradually with time (over a period of years) due to seal degradation and corrosion. In Figure 5-1, it may be seen that below 0.5 GPM leakage, the 40 year fatigue usage due to rapid thermal cycling is minimal. Up to about 1 GPM, the fatigue usage is still very low.

By continually monitoring leakage using Browns Ferry's monitoring equipment, the necessity of repairs to thermal sleeve seals can be anticipated well in advance. Properly timed repairs should prevent accumulated usage from exceeding allowable values.

Since thermal sleeve seals degrade only gradually, sudden changes in data from the leakage monitoring system (e.g., large step increases indicated leakage) are indicative of monitoring system component problems such as thermocouple detachment. A thorough LMS component check should be made at the next convenient time.

5.4 Unstable Flow/Low Flow Evaluation

TVA provided data (Reference 6) which describes variation of feedwater temperature within the horizontal portion of the pipe upstream of the feedwater nozzle. Resistance Temperature Detectors (RTDs) located at 12, 3, 6, and 9 o'clock azimuthally were installed to monitor the thermal effects of feedwater low flow operation.

The data, which were taken at 1 minute intervals, suggest that the feedwater flow at Browns Ferry is very

accurately controllable. During typical start-ups and scrams, few thermal transients were observed (4-10 generally). Transients had a period of 1-2 hours.

Thermal cycling of this character is very mild compared to most of the cases previously analyzed. A comparison of the crack growth cases considered in Reference 2 with the cycling experienced at Browns Ferry shows that a 1/4 inch crack would require roughly 20 years to propagate to a 1 inch depth under the impetus of such cycling. This is a worst case estimate.

A review of plant piping and operational procedures revealed that the feedwater system itself is rarely used when extremely low water flow rates are required to maintain vessel water level. Instead, the Control Rod Drive Return line, which is tied into one feedwater loop and which has a capacity of 100 GPM, is used to supply vessel make up flow. Since the flow can be maintained steadily, thermal transients due to intermittent operation of the feedwater pumps are minimized. Feedwater flow is stabilized by pumping to the condenser until the greater capacity of the feedwater system is required.

Consequently, the crack growth concerns related to cyclic low flow feedwater system operation are

minimized. The present method of system operation is considered adequate. Modifications to feedwater low flow control equipment solely for the purpose of mitigating low flow/unsteady flow induced thermal cycling is not deemed to be necessary.

5.5 System Cycling

System cycling includes the effects of major operational transients on the reactor pressure boundary. Such transients include the pressure and thermal transient portions of start-ups, shutdowns, scrams, turbine rolls, etc. Fatigue due to system cycling contributes to both crack initiation and crack growth. Consequently, a fatigue usage factor due to system cycling must be added to that due to rapid thermal cycling to determine the total fatigue usage factor.

Values for the fatigue usage factor due to system cycling were taken from Reference 5 for the Browns Ferry feedwater nozzles. The system cycle fatigue usage factors for 40 years are approximately 0.40 between the thermal sleeve seals, and 0.23 at the nozzle blend radius, based upon the referenced Stress Reports. TVA maintains an on going record of System Cycling base upon transients actually experienced at each unit.

5.6 Total Fatigue Evaluation

The system cycling fatigue values were added to the values determined for rapid thermal cycling to generate a total usage factor. These factors are compiled in Table 5-10 for various leakage rates.

5.7 Reactor Water Clean Up Cross-tie Evaluation

NUREG 0619 (Reference 7) proposes several methods for reducing the fatigue usage of the feedwater nozzles. One of these recommendations is the installation of a cross-tie line to allow Reactor Water Clean Up (RWCU) return flow to be routed to both feedwater loops, (RWCU flow presently returns via loop B only except on Unit 3). The mixing of high temperature RWCU water with lower temperature feedwater would tend to mitigate the thermal fatigue effects of feedwater flow on the feedwater nozzles by decreasing the temperature difference between reactor and feedwater. To evaluate the effectiveness of such a modification, the thermal duty map developed above was revised to include the effects of RWCU flow. The following characteristics of the RWCU system were assumed:

1. RWCU return temperature at normal operating reactor temperature (539°F) = 472°F.

2. RWCU flow (100%) = 300 GPM

These numbers were taken from discussions with TVA personnel, and from Browns Ferry Drawing 47W810 (Reference 8).

It should be noted that since RWCU return heating is accomplished by a regenerative/non-regenerative self heat exchanging system (i.e., there is no source of heating other than RWCU inlet flow), the achievable RWCU return temperature is governed by actual reactor temperature. In other words, a drop in reactor temperature must necessarily produce a comparable drop in achievable RWCU temperature. Furthermore, the clean up system at Browns Ferry is designed as a high pressure system. This implies that as reactor pressure drops, maximum achievable RWCU flow will also drop. The assumption that the above temperature and flow are available throughout the thermal duty map produces results which are an upper bound on the improvement in fatigue usage which may be expected through the use of RWCU. A lower limit on expected improvement can be obtained by assuming that RWCU is only available when the reactor is

at operating temperature (539°F). Analysis of these two cases provides an indication of the range of fatigue usage improvement which may be expected.

Comparison of the results of rapid cycle evaluation with and without RWCU shows that:

1. Beyond the second thermal sleeve seal an improvement in fatigue usage of less than 1% would result from incorporation of the mitigating effect of RWCU.

2. Between the seals, and improvement of about 6% could be expected.

To corroborate these analytical conclusions, Browns Ferry directly monitored the temperature effects of operation with and without RWCU cross tie on Unit 3, using the system described in Section 5.4. This data showed no detectable difference in temperature measured upstream of the seals (Reference 6).

These improvements are insignificant. Consequently, the installation of a RWCU cross-tie modification at Browns

Ferry Units 1, 2, and 3 is not considered to be justified unless additional operational flexibility is gained thereby.

5.8 In-Service Inspection

The analysis presented herein shows that the total fatigue usage factor remains substantially below the ASME allowable value of 1.0 during the lifetime of the Browns Ferry Units, unless secondary thermal sleeve seal leakage exceeds 1.0 GPM. With continual leakage monitoring, the potential for crack initiation can be accurately assessed without inspection.

If the thermal sleeve seals are shown to allow only minimal leakage, NUREG-0619 suggests that the NRC may be receptive to delaying required inspections on a case-by-case basis. The justification for such delays would be that by continually monitoring leakage and evaluating fatigue usage resulting from the measured leakages, repairs to thermal sleeve seals could be planned to prevent the fatigue usage from becoming excessive. If leakage remains low, minimal fatigue usage may be expected. If leakage increased with time, repair can be planned sufficiently in advance to prevent signific nt

rapid thermal cycle fatigue usage accumulation. Note that leakage is not expected to increase suddenly, but rather gradually with time.

Table 5-1

BROWNS FERRY PLANT SPECIFIC THERMAL DUTY MAP

Region	% Power	Hours/year	Rx Temp	FW Temp
1	100.00	5528.00	539.00	372.00
2	95.00	1195.00	539.00	360.00
3	80.00	373.00	539.00	350.00
4	70.00	302.00	539.00	337.00
5	50.00	94.00	539.00	315.00
6	50.00	3.00	539.00	285.00
7	40.00	42.00	539.00	305.00
8	40.00	14.00	539.00	240.00
9	32.00	14.00	539.00	185.00
10	23.00	12.00	539.00	175.00
11	23.00	8.00	539.00	120.00
12	15.00	16.00	539.00	120.00
13	5.00	64.00	530.00	120.00
14	5.00	14.00	470.00	120.00
15	5.00	10.00	360.00	120.00
16	5.00	40.00	240.00	120.00
17	5.00	48.00	512.00	210.00
18	5.00	29.00	298.00	210.00
19	5.00	13.00	536.00	308.00
20	0.00	43.00	340.00	300.00
21	1.00	.40	360.00	350.00
22	2.00	1.78	350.00	190.00
23	2.00	1.38	340.00	125.00
24	2.00	.25	330.00	70.00
25	2.00	1.60	400.00	190.00
26	3.00	.38	340.00	160.00
27	0.00	900.00	70.00	70.00

Table 5-2

AMPLITUDE/FREQUENCY DATA FOR RAPID CYCLING

	BETWEEN SEALS		DOWNSTREAM OF SEALS	
Index I	Amplitude $\bar{A}$	Frequency Cycles/Hr	Amplitude $\bar{A}$	Frequency Cycles/Hr
1	1.00	45	1.00	15
2	0.95	35	0.98	15
3	0.90	20	0.955	15
4	0.85	120	0.91	30
5	0.77	100	0.84	75
6	0.66	105	0.75	120
7	0.56	100	0.65	150
8	0.46	100	0.55	180
9	0.36	100	0.45	450
10	0.26	1200	0.35	1200
11	0.15	7500	0.20	7500

Table 5-3

COEFFICIENT C_3

Pt.	100% Rated Feedwater Flow	20% Rated Feedwater Flow	0% Rated Feedwater Flow
Nozzle Bore	1.0	1.0	1.0

Table 5-4

COEFFICIENT C_4

Thermal Sleeve Bypass Leakage	0 GPM	.35 GPM	0.5 GPM	1.5 GPM
Coefficient C_4	0.1	0.1	0.24	0.3

Table 5-5

RAPID CYCLE FATIGUE USAGE (Nozzle Blend Radius):

1.0 GPM LEAKAGE

Year	Cumulative Fatigue Usage Factor	Leakage
1	.003	1.000
2	.005	1.000
3	.008	1.000
4	.011	1.000
5	.014	1.000
6	.016	1.000
7	.019	1.000
8	.022	1.000
9	.024	1.000
10	.027	1.000
11	.030	1.000
12	.033	1.000
13	.035	1.000
14	.038	1.000
15	.041	1.000
16	.044	1.000
17	.046	1.000
18	.049	1.000
19	.052	1.000
20	.054	1.000
21	.057	1.000
22	.060	1.000
23	.063	1.000
24	.065	1.000
25	.068	1.000
26	.071	1.000
27	.073	1.000
28	.076	1.000
29	.079	1.000
30	.082	1.000
31	.084	1.000
32	.087	1.000
33	.090	1.000
34	.092	1.000
35	.095	1.000
36	.098	1.000
37	.101	1.000
38	.103	1.000
39	.106	1.000
40	.109	1.000

Table 5-6

RAPID CYCLE FATIGUE USAGE (Nozzle Blend Radius): 1.25 GPM LEAKAGE

Year	Cumulative Fatigue Usage Factor	Leakage
1	.007	1.250
2	.013	1.250
3	.020	1.250
4	.026	1.250
5	.033	1.250
6	.039	1.250
7	.046	1.250
8	.052	1.250
9	.059	1.250
10	.066	1.250
11	.072	1.250
12	.079	1.250
13	.085	1.250
14	.092	1.250
15	.098	1.250
16	.105	1.250
17	.111	1.250
18	.118	1.250
19	.125	1.250
20	.131	1.250
21	.138	1.250
22	.144	1.250
23	.151	1.250
24	.157	1.250
25	.164	1.250
26	.170	1.250
27	.177	1.250
28	.184	1.250
29	.190	1.250
30	.197	1.250
31	.203	1.250
32	.210	1.250
33	.216	1.250
34	.223	1.250
35	.229	1.250
36	.236	1.250
37	.243	1.250
38	.249	1.250
39	.256	1.250
40	.262	1.250

Table 5-7

RAPID CYCLE FATIGUE USAGE (Nozzle Blend Radius):

1.5 GPM LEAKAGE

Year	Cumulative Fatigue Usage Factor	Leakage
1	.014	1.500
2	.027	1.500
3	.041	1.500
4	.054	1.500
5	.068	1.500
6	.092	1.500
7	.095	1.500
8	.109	1.500
9	.122	1.500
10	.136	1.500
11	.149	1.500
12	.163	1.500
13	.177	1.500
14	.190	1.500
15	.204	1.500
16	.217	1.500
17	.231	1.500
18	.245	1.500
19	.258	1.500
20	.272	1.500
21	.285	1.500
22	.299	1.500
23	.312	1.500
24	.326	1.500
25	.340	1.500
26	.353	1.500
27	.367	1.500
28	.380	1.500
29	.394	1.500
30	.408	1.500
31	.421	1.500
32	.435	1.500
33	.448	1.500
34	.462	1.500
35	.476	1.500
36	.489	1.500
37	.503	1.500
38	.516	1.500
39	.530	1.500
40	.543	1.500

Table 5-8

RAPID CYCLE FATIGUE USAGE (Nozzle Blend Radius):
LEAKAGE INCREASING WITH TIME

Year	FIT	Cumulative Fatigue Usage Factor	Leakage
1	.0046	.000	0.000
2	.0037	.000	0.000
3	.0029	.000	0.000
4	.0021	.000	.035
5	.0014	.000	.103
6	.0008	.000	.171
7	.0001	.000	.239
8	-.0006	.000	.307
9	-.0013	.000	.375
10	-.0020	.000	.465
11	-.0026	.000	.567
12	-.0033	.001	.669
13	-.0040	.002	.771
14	-.0047	.003	.873
15	-.0054	.005	.975
16	-.0060	.009	1.051
17	-.0067	.013	1.119
18	-.0074	.018	1.187
19	-.0081	.025	1.283
20	-.0088	.034	1.385
21	-.0094	.047	1.487
22	-.0101	.061	1.500
23	-.0108	.074	1.500
24	-.0115	.088	1.500
25	-.0122	.102	1.500
26	-.0128	.115	1.500
27	-.0135	.129	1.500
28	-.0142	.142	1.500
29	-.0149	.156	1.500
30	-.0156	.170	1.500
31	-.0162	.183	1.500
32	-.0169	.197	1.500
33	-.0176	.210	1.500
34	-.0183	.224	1.500
35	-.0190	.237	1.500
36	-.0196	.251	1.500
37	-.0203	.265	1.500
38	-.0210	.278	1.500
39	-.0217	.292	1.500
40	-.0224	.305	1.500

Table 5-9

RAPID CYCLE FATIGUE USAGE CONTRIBUTION

BY MAP REGION LEAKAGE = 1.5 GPM

MAP	ΔT_{p-p}	Eα	Sa	Fatigue Usage Per Region	Cumulative Fatigue Usage Factor
1	50.100	239.76	8.58	.0059	.0059
2	52.895	239.58	9.05	.0024	.0083
3	53.298	239.56	9.12	.0008	.0091
4	55.146	239.44	9.43	.0010	.0101
5	57.120	239.32	9.76	.0005	.0106
6	64.770	238.83	11.05	.0001	.0106
7	57.564	239.29	9.84	.0002	.0108
8	73.554	238.28	12.52	.0010	.0118
9	78.352	237.98	13.32	.0018	.0136
10	36.400	240.62	6.26	0.0000	.0136
11	41.900	240.27	7.19	.0000	.0136
12	41.900	240.27	7.19	.0000	.0136
13	41.000	239.20	7.01	.0000	.0136
14	35.000	232.02	5.80	0.0000	.0136
15	24.000	218.85	3.75	0.0000	.0136
16	12.000	204.48	1.75	0.0000	.0136
17	30.200	237.61	5.13	0.0000	.0136
18	8.800	211.99	1.33	0.0000	.0136
19	22.800	241.10	3.93	0.0000	.0136
20	4.000	217.59	.62	0.0000	.0136
21	1.000	220.30	.16	0.0000	.0136
22	16.000	218.09	2.49	0.0000	.0136
23	21.500	216.49	3.32	0.0000	.0136
24	26.000	214.94	3.99	0.0000	.0136
25	21.000	224.80	3.36	0.0000	.013[illegible]
26	18.000	216.71	2.79	0.0000	.013[illegible]
27	0.000	183.82	0.00	0.0000	.0136

Table 5-10

TOTAL FATIGUE USAGE - 40 YEARS (ASSUMING CONTINUOUSLY INCREASING LEAKAGE)

Unit	Location	40 Year* System Cycling Fatigue Usage Factor	40 Year Rapid Cycling Fatigue Usage Factor	Total
1	Nozzle radius	.2575	.3056	.5631
2-3	Nozzle radius	.2205	.3056	.5261

* From Reference 5.

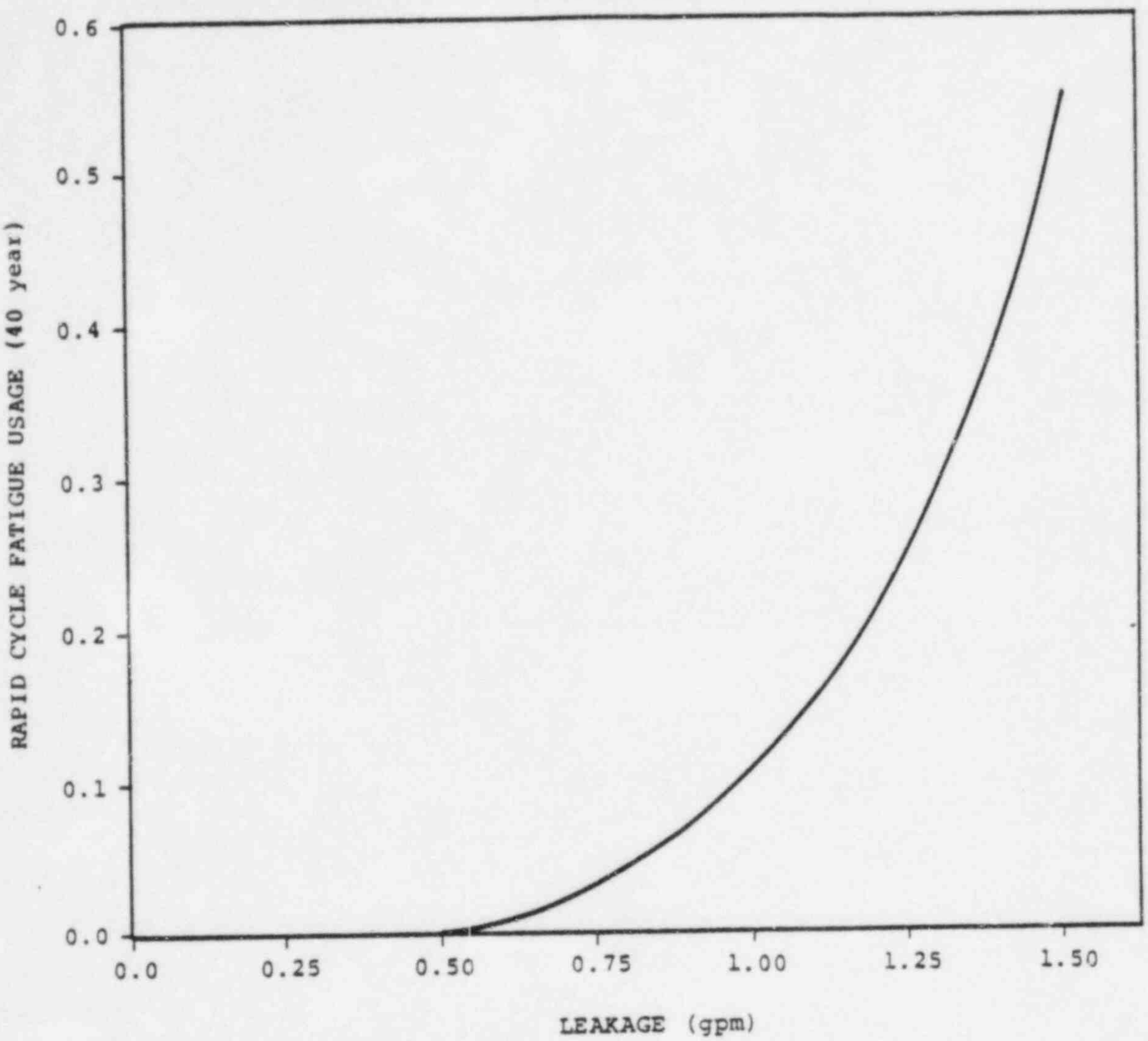

Figure 5-1

FATIGUE USAGE FACTOR VS LEAKAGE RATE

6.0 CONCLUSIONS

6.1 Results

The NUTECH evaluation of the Browns Ferry feedwater nozzles has produced the following conclusions:

1. If leakage past the secondary seals is low, fatigue-induced cracks of the feedwater nozzle bore and blend radius are not expected.

2. By continually monitoring seal leakage, increasing fatigue usage can be anticipated sufficiently in advance to prevent crack initiation. Seal repairs could be planned as necessary. Such a program may be acceptable to the Nuclear Regulatory Commission as an alternate to in-vessel dye penetrant inspection of the feedwater nozzles. (Reference 7).

3. Browns Ferry's present method of operation at low feedwater flows does not significantly contribute to fatigue usage or to growth of existing cracks. System flows at very low power levels appear to be accurately controllable with existing equipment. Therefore, it does not appear to be necessary to upgrade the present system if the concerns of NUREG-0619 are the only reasons for doing so.

4. The potential improvement to nozzle fatigue service life which would result from a proposed Reactor Water Clean Up System cross-tie modification is considered to be insignificant.

6.2 Recommendations

NUTECH offers the following recommendations for further actions by TVA.

1. Leakage monitoring using the new system should be implemented. Data should be taken weekly to detect leakage increases with time. This would be indicative of seal degradation due to corrosion.

2. If leakage rates appear to be increasing with time, repairs to the thermal sleeve seals should be planned sufficiently in advance to prevent total fatigue usage factor from exceeding 1.0 during the remaining plant life.

7.0 REFERENCES

1. Browns Ferry operational information. Transmitted by letter from C. R. Favreau (TVA) to H. L. Gustin (NUTECH), dated March 19, 1982.

2. Watanabe, H., Boiling Water Reactor Feedwater Nozzle/Sparger Final Report, NEDE-21821. General Electric Company, March, 1978.

3. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, 1980 Edition.

4. Browns Ferry system operational data and computer logs for Browns Ferry Unit 1, Cycle 4 provided by TVA personnel to H. L. Gustin (NUTECH) during site visit March 15-17, 1982. NUTECH File Nos. 165.1202.0012, and 165.1202.0013.

5. General Electric Company feedwater nozzle stress reports and analyses.

   a. General Electric Stress Report 22A5594, "Feedwater Nozzle," Browns Ferry 1, 2 and 3, Revision 2, June 1, 1979.

b. General Electric Design Specification (Repair) 22A5584, "Reactor Vessel," Revision 1, March 21, 1978.

c. General Electric Design Certification 22A5539, "Reactor Vessel," Revision 1, October 7, 1977.

d. General Electric Design Specification 22A5593, "Reactor Vessel" Browns Ferry 1, 2 and 3, Revision 2, May 22, 1979.

e. General Electric Stress Report 22A5562, "Reactor Vessel," Revision 0, October 7, 1977.

6. Browns Ferry feedwater low flow control data. Transmitted by letter from T. F. Ziegler (TVA) to L. C. Hsu (NUTECH), dated June 28, 1982.

7. NUREG-0619,"BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U.S. Nuclear Regulatory Commission, November 1980.

8. Browns Ferry feedwater and reactor water cleanup systems design drawings. Transmitted by letter from T. F. Ziegler (TVA) to H. L. Gustin (NUTECH), dated May 3, 1982.

ENCLOSURE 2

PLAN FOR RESOLUTION OF CRACKING PROBLEM AND
CONTINUED MONITORING OF FEEDWATER NOZZLE
PURSUANT TO NUREG-0619
BROWNS FERRY NUCLEAR PLANT

1. Sparger and Thermal-Sleeve Design Modifications (NUREG-0619, Section 4.1)

Nozzle cladding has been removed and GE triple-sleeve spargers have been installed on all Browns Ferry units.

2. Low-flow Controller (NUREG-0619, Section 4.2, and NRC Generic Letter 81-11)

Based on the NUTECH anaysis, installation of a low-flow controller having the characteristics described in section 3.4.4.3 of NEDE-21821-A has been found to be unnecessary. First, the high-cycle fatigue analysis indicated that no cracks will be initiated in the feedwater nozzle blend radius over the 40-year plant life assuming a thermal sleeve leakage of less than 1.5 gpm. Second, even if a 1/4-inch crack existed, it would take 20 years for the crack to grow to one inch (worst case estimate). TVA intends to take appropriate action when a crack is indicated by UT examination or when thermal sleeve leakage greater than 1.5 gpm is measured. The NRC will be notified in either case.

3. Reactor Water Cleanup Reroute (NUREG-0619, section 4.2, and NRC Generic Letter 81-11)

Again based on the NUTECH analysis, rerouting of the reactor water cleanup system at Browns Ferry has been found to be unnecessary. The analysis has concluded that the benefit of such rerouting is insignificant and this fact has been supported by experimental data at Browns Ferry. This modification had previously been done on unit 3.

4. Inspections and Leak Detection (NUREG-0619, section 4.3)

TVA will continue to perform the UT and visual inspections at the intervals required by Table 2 (page 18) of NUREG-0619. With regard to the routine PT requirements, TVA proposes an alternative to the NUREG-0619 inspection interval of nine refueling cycles (or 135 startup/shutdown cycles). NUTECH's worst case crack growth estimate predicts that it will take 20 years for a 1/4-inch crack (undetected) to grow to one inch. Therefore, TVA proposes to perform the PT examinations on each unit after 20 years of operation, starting from when the nozzle cladding was removed. Cladding has been removed from the Browns Ferry units, and startup of the unit from the outage when that was done is as follows:

Unit 1 - January 1978
Unit 2 - June 1978
Unit 3 - December 1979

These dates are proposed as the start of the 20-year interval for each unit as discussed above. If the end of the 20-year period occurs during a cycle of operation, the PT examinations will be performed at the following scheduled refueling outage. Performance of the PT examinations on this proposed alternative inspection interval will result in some reduction of cumulative exposure to personnel over that to be incurred from the intervals currently in NUREG-0619.

We are hopeful that advances in ultrasonic testing (UT) technology will totally eliminate the need for PT examination of the feedwater nozzles. However, until approval by NRC of improved UT techniques TVA will perform the PT examinations in accordance with the alternative outlined above.

By letter from M. R. Wisenburg to H. R. Denton dated October 13, 1981, TVA committed to install a leak detection system to monitor thermal-sleeve bypass leakage on each Browns Ferry unit. In accordance with NUREG-0619 (section 4.3.2.4) we will keep the NRC staff informed as to its performance and assessment of leakage measurements. The NUTECH analysis has shown that leakage values less than 1.5 gpm will not result in a 40-year fatigue usage factor greater than 1.0. Therefore, we commit to remedial measures or additional analysis only if the leakage exceeds 1.5 gpm.

A further benefit of the leakage detection system will be to define the time when seal refurbishment becomes necessary. GE seal refurbishment schedules based on estimated corrosion rates no longer apply.