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SOUTHERN COMPANY
Energy to Serve Your World℠

February 20, 2004

Docket Nos.: 50-348
50-364

NL-04-0069

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant Units 1 and 2
Application for License Renewal –
January 22, 2004 Requests for Additional Information

Ladies and Gentlemen:

This letter is in response to your letter dated January 22, 2004 requesting additional information for the review of the Joseph M. Farley Nuclear Plant, Units 1 and 2, License Renewal Application. Responses to these Requests for Additional Information (RAI's) are provided in Enclosures 1 and 2.

Mr. L. M. Stinson states he is a vice president of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

If you have any questions, please contact Charles Pierce at 205-992-7872.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson
Vice President, Farley

Sworn to and subscribed before me this 20th *day of* February, *2004.*

Notary Public

My commission expires: 6-7-05

LMS/JAM/slb

A099

Enclosures: 1. Response to January 22, 2004 Requests for Additional Information, Joseph M. Farley Nuclear Plant, Units 1 and 2

2. FNP 01Q-301, Revision 2, "Surge Line and Pressurizer Lower Head Fatigue Analysis"

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley Jr., Executive Vice President
Mr. D. E. Grissette, General Manager – Plant Farley
Document Services RTYPE: CFA04.054; LC# 13919

U. S. Nuclear Regulatory Commission
Ms. T. Y. Liu, License Renewal Project Manager
Mr. L. A. Reyes, Regional Administrator
Mr. S. E. Peters, NRR Project Manager – Farley
Mr. C. A. Patterson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Enclosure 1
Response to January 22, 2004 Requests for Additional Information
Joseph M. Farley Nuclear Plant Units 1 and 2
License Renewal Application

RAI 4.3.1-1

Table 4.3.1 of the LRA provides the current cycle counts and estimated cycle counts at 60 years of plant operation for transients used in the design of Class 1 components. Note 5 of the table indicates that step load change transients were not counted prior to the installation of fatigue monitoring software, and that the number of these prior transients would be estimated using the current fatigue monitoring software. Describe the method that will be used to estimate the number of transient cycles that occurred prior to the installation of the fatigue monitoring software. Provide a list of the transients that will be monitored by the fatigue monitoring program.

Response

The method Southern Nuclear Operating Company (SNC) will use to determine the number of step load change transients that occurred prior to establishing the fatigue monitoring software is a backwards projection using five or more years worth of data from the fatigue monitoring software as a baseline. To account for improved operating performance in recent years as compared to the first few years of plant operation, a weighting factor will be applied to assume additional cycles during the early years of operation. The weighting factor will be based on a ratio of reactor trips during the early years divided by the number of reactor trips during the five or more years baseline data is collected. The weighting factor may be revised if SNC determines there is a more suitable technique than the ratioing of reactor trip data. Since load follow operation is not used at FNP, the number of step load change transients is expected to be a small fraction of the number allowed.

(Response continued on the following page.)

In addition to the list of transients below, the fatigue monitoring program will be used to monitor thermal stratification cycles at susceptible locations on each unit.

Design Transient Description	Number Allowed
RCS Heatup	200
RCS Cooldown	200
Pressurizer Heatup	200
Pressurizer Cooldown (to 400 psia)	200
Pressurizer Cooldown (from 400 psia)	200
Small-Step Load Increase	2,000
Small-Step Load Decrease	2,000
Large-Step Load Decrease	200
Loss of Load without Reactor Trip	80
Loss of RCS Flow in a single Loop	80
Reactor Trip, No Cooldown	230
Reactor Trip, with Cooldown	160
Reactor Trip, with Cooldown and Safety Injection	10
Inadvertent Auxiliary Spray	10
Primary Side Leak Test	50
Primary Side Hydro Test	5
Secondary Side Hydro Test	5
Turbine Roll Test	10
Loss of Offsite Power	40
Operating Basis Earthquake	5
Main Steam Line Break	1

RAI 4.3.1-2

The Westinghouse Owners Group issued Topical Report WCAP-14577, Revision 1-A, "Aging Management for Reactor Internals," to address the aging management of the reactor vessel internals. Section 2.3.1 of the LRA indicates that WCAP-14577, Revision 1-A was reviewed as a source of input information for FNP. The staff's review of WCAP-14577, Revision 1-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 11 specified in WCAP -14577, Revision 1-A indicates that the fatigue TLAA of the reactor vessel internals should be addressed on a plant specific basis. Discuss the design basis for the components listed in Table 3-3 of WCAP-14577, Revision 1-A. Indicate how fatigue of these components is managed.

Response

Since the Farley reactor internals were designed prior to the introduction of Subsection NG of the ASME Boiler and Pressure Vessel Code Section III, a plant specific stress report on the reactor internals was not required. However, the design of the Farley reactor internals was evaluated according to Westinghouse internal criteria which were similar to the criteria described in Subsection NG of the ASME Code. The structural integrity of the Farley reactor internals design has been shown by analyses performed on both generic and plant specific bases. Included in these evaluations was consideration of the assumed thermal transient cycles.

Section 4.3.1 of the LRA states that SNC "has determined that the assumed transient cycles are conservative for 40 years and bounding for the extended term of operation, except in certain specific cases described below." The exceptions are not related to the internals.

Section B.5.7 of the LRA states, "The other Class 1 components that have received a fatigue analysis will also be included, since the cycles they were designed for are bounded by the cycle limits used by the program." This statement does apply to the internals.

Therefore the cycles assumed by the reactor vessel internals evaluations are bounded by those monitored by the fatigue monitoring program so that the original number of cycles assumed will not be exceeded during the extended period of operation.

RAI 4.3.1-3

The Westinghouse Owners Group has issued the generic Topical Report WCAP-14574-A to address aging management of pressurizers. Section 2.3.1 of the LRA indicates that WCAP-14577, Revision 1-A was reviewed as a source of input information for FNP. The staff's review of WCAP-14574-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 1 requests the applicant to demonstrate that the pressurizer sub-component cumulative usage factors (CUF's) remain below 1.0 for the period of extended operation. Table 2-10 of WCAP-14574-A indicates that the ASME Section III Class 1 fatigue CUF criterion could be exceeded at several pressurizer sub-component locations during the period of extended operation. WCAP-14574-A also identified recent unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses, including inflow/outflow thermal transients. Provide the following information:

A. Confirm that the additional transients discussed in WCAP-14574-A, not considered in the original design, have been addressed at FNP.

Response

As stated in Section B.5.7 of the LRA, SNC confirms that the stress-based monitoring of the surge line and lower pressurizer that is part of the fatigue monitoring program accounts for the additional transients discussed in WCAP-14574-A. While this WCAP is not credited for FNP license renewal, the unanticipated transients of thermal stratification and insurge/outsurge are addressed in the Fatigue Monitoring Program. Specifically, the following loadings are considered:

1. Internal Pressure
2. Surge line piping thermal expansion
3. Surge line piping thermal stratification
4. Thermal shock, or "insurge/outsurge" temperature transients from flow reversals

B. Show the ASME Section III Class 1 current licensing basis (CLB) CUFs for the applicable sub-components of the FNP pressurizers specified in Table 2-10 of WCAP-14574-A and the corresponding CUFs for the extended period of operation.

Response

At FNP, all of the pressurizer sub-components have a design CUF equal to or less than that shown in Table 2-10 of WCAP-14574-A based on an assumed set of transients.

As stated in Section 4.3.1 of the LRA, SNC "has determined that the assumed transient cycles are conservative for 40 years and bounding for the extended term of operation, except in certain specific cases described below." The exception related to the pressurizer is that stress based fatigue monitoring is used for the surge line and lower pressurizer to ensure the surge line CUF does not exceed 1.0 for 60 years when environmental factors and unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses, including inflow/outflow thermal transients, are included.

Section B.5.7 of the LRA states, "The other Class 1 components that have received a fatigue analysis will also be included, since the cycles they were designed for are bounded by the cycle limits used by the program." This statement applies to the pressurizer sub-components not included in the stress based monitoring of the lower pressurizer.

Therefore the cycles assumed by the pressurizer fatigue calculations are bounded by those monitored by the fatigue monitoring program so that the original number of cycles assumed will not be exceeded during the extended period of operation and, in addition, stress based fatigue monitoring is used to ensure that the surge line and lower pressurizer CUF remains less than 1.0.

The 60-year projected fatigue usage (including insurge/outsurge but without considering environmental effects) for the stress based locations monitored in the Farley fatigue monitoring program are shown below:

Unit	RCS Hot Leg Surge Nozzle	Pressurizer Heater Penetration	Pressurizer Surge Nozzle
1	0.0014	0.0954	0.0082
2	0.0018	0.1115	0.0033

C. Discuss the impact of the environmental fatigue correlations provided in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," on the above results.

Response

SNC established stress-based fatigue monitoring of the surge line and lower pressurizer head in order to address the impact of the environmental fatigue correlations provided in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," for these components. Stress-based fatigue monitoring of the surge line and lower pressurizer head provide "per event" CUFs for 3 locations, the RCS surge line nozzle, the pressurizer surge line nozzle and the pressurizer heater penetration. The "per event" CUFs for each location is multiplied by 200 allowed events. The pressurizer heater penetration is bounding for both units, with projected 60-year environmentally-assisted fatigue usage values of 0.6427 and 0.975 for Units 1 and 2, respectively. The F_{en} values of 7.90 and 8.28 were calculated for the bounding location on Units 1 and 2, respectively using the correlations in NUREG/CR-5704. The projected 60-year CUF, with environmental effects, is satisfactory for all three monitored locations.

D-RAI 4.3.1-4

Section 4.3.1 of the LRA indicates that the fatigue usage for the surge line is not expected to exceed 1.0 during the period of extended operation. The LRA also indicates that stress based fatigue monitoring software was used to analyze the fatigue usage of the surge line hot leg nozzle. Describe the stress based fatigue monitoring of the surge line hot leg nozzle. Indicate whether there were any changes in the plant operations since the startup of Units 1 and 2 that could affect the fatigue usage of the surge line hot leg nozzle. Provide a copy of reference 11.

Response

The FatiguePro software installed at FNP-1/2 monitors several components in the surge line and pressurizer using the stress-based fatigue method. The components monitored for fatigue usage are the pressurizer surge nozzle and hot leg surge nozzle in the surge line and the pressurizer heater penetration in the pressurizer lower head. The FatiguePro software computes transient stress histories at the monitored components using pertinent thermal and mechanical loadings and actual plant instrument data. These stress histories are then evaluated using ASME Section III methodology to produce cumulative fatigue usage for each monitored component.

Interviews with shift supervisors from the first several operating cycles indicate that the current method of operation was established very early in plant life, probably before the second cycle on Unit 1. FNP currently uses the Modified Steam Bubble Method described in WCAP-13588, "Operating Strategies for Mitigating Pressurizer Insurge and Outsurge Transients" and was the demonstration plant for that method since it was already in place at FNP. Procedures from that period allowed, but did not require the current method of operation, so it cannot be proven that the current method has been in use that long.

To compensate for any differences in early operation and current operation, for the RCS Hot Leg Surge Nozzle, a factor of 1.5 was applied to the calculated stress-based CUF per event for events early in plant life.

Enclosure 2 is the current revision of Reference 11 in Section 4 of the Farley License Renewal Application.

D-RAI 4.3.3-1

Section 4.3.3 of the LRA indicates that the number of thermal cycles for the EDG air start subsystem may exceed 7,000 during the period of extended operation. The LRA also indicates that the equivalent number of full-temperature cycles will be less than 7,000 cycles. Describe the method used to calculate the equivalent number of full-temperature cycles.

Response

The EDG air start subsystem for Plant Farley actuates several times per day to recharge the air pressure in the air receivers lost due to normal leakage. To determine the impact of these recharging cycles, SNC installed temperature monitoring equipment on two of the skids to measure the temperatures just downstream of the compressors over several days of normal operation. SNC analyzed the resulting data to determine the number of temperature cycles per day and the magnitude of the temperature cycles. From the data SNC conservatively assumed 7 cycles a day for 22,000 days (154,000 partial temperature cycles) over 60 years of operation.

None of the measured temperature cycles approached the design maximum temperature cycle used in the original stress analysis for the air start piping. The maximum measured temperature from the data was 226 °F for a recharge cycle. Whenever the air receivers are fully recharged following major maintenance to the respective EDGs, a maximum temperature cycle (from ambient to 615 °F) is assumed to occur. This kind of major maintenance occurs on the EDGs once every outage (18 months – a total of 40 for each air start line was assumed). SNC calculated the equivalent full temperature cycles (N) per the ASME Code Section III 1971 Edition, Sub-section NC-3611.1.

$$N = N_E + r_1^5 N_1 + r_2^5 N_2 + \ldots + r_n^5 N_n$$

Where:

N_E = number of cycles at run temperature change ΔT_E for which expansion stress S_E has been calculated (maximum design temperature change).

$N_1, N_2, \ldots, N_n$ = number of cycles at ***lesser*** temperature changes, $\Delta T_1, \Delta T_2, \ldots, \Delta T_n$

$r_1, r_2, \ldots, r_n = (\Delta T_1)/(\Delta T_E), (\Delta T_2)/(\Delta T_E), \ldots, (\Delta T_n)/(\Delta T_E)$,
= the ratio of any lesser temperature cycles for which the expansion stress S_E has been calculated

For the FNP EDG air start subsystem cycling,

N_E = 40 cycles $\quad$ N_1 = 154,000 cycles $\quad$ $r_1 = (226-70) \div (615-70) = 0.2862$

$N = N_E + r_1^5 N_1$
$= 40 + (0.2862)^5 (154,000)$
= 336 equivalent full cycles

Since 336 < 7000, the 7000 thermal cycles assumption is appropriate for 60 years of operation, and the stress range reduction factor (f) of 1.0 is acceptable for evaluating expansion stress.

RAI 4.3.4-1

10CFR54.21(c)(1)(ii) requires that the applicant demonstrate the adequacy of the analysis projected for the extended period of operation. In order for the staff to make a reasonable assurance conclusion, the applicant is requested to provide the following information:

(a) Minimum required prestressing forces for each group of tendons,

(b) Trend lines of the projected prestressing forces for each group of tendons based on the regression analysis of the measured prestressing forces (see NRC Information Notice 99-10 for more information).

(c) Plots showing comparisons of prestressing forces projected to the end of the extended period of operation with the minimum required prestress for each group of tendons.

Response:

This information will be provided to the NRC by March 5, 2004.

D-RAI 4.3.4-2

In Section A.4.3 in the UFSAR Supplement of the LRA, the applicant states, "The calculation indicates that acceptable containment prestress will continue to exist throughout the extended period of operation." In order for the summary to be meaningful, as a minimum, the applicant should provide a Table showing the minimum required prestressing forces and the projected (to 60 years) prestressing forces for each group of tendons which would demonstrate the validity of the analysis results. The applicant is requested to supplement this information in Section A.4.3 of the UFSAR Supplement.

Response

This information will be provided to the NRC by March 5, 2004.

Enclosure 2

Surge Line and Pressurizer Lower Head Fatigue Analysis

STRUCTURAL INTEGRITY Associates	# CALCULATION PACKAGE	FILE No.: FNP-01Q-301 PROJECT No.: FNP-01Q

PROJECT NAME: Farley License Renewal

CLIENT: Southern Nuclear Operating Company

TITLE: Surge Line and Pressurizer Lower Head Fatigue Analysis

Document Revision	Affected Pages	Revision Description	Project Mgr. Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1-8 A1-8	Original Issue	D.A. Gerber DAG 7/2/03	T.D. Gilman TDG 7/2/03 D.A. Gerber DAG 7/2/03
1	2, 5, 7	Made typographical revisions	D.A. Gerber DAG 11/5/03	T.D. Gilman TDG 11/5/03 D.A. Gerber DAG 11/5/03
2	1, 3, 5-8	Reduced conservatism in software and F_{en} calculations.	D.A. Gerber DAG 12/18/03	T.D. Gilman TDG 12/18/03 D.A. Gerber DAG 12/18/03

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1.0 OBJECTIVE

The purpose of this calculation package is to compute stresses and associated fatigue usage for components in the Farley Nuclear Plant Units 1 and 2 surge line and pressurizer lower head for a projected 60 year operation, including environmental effects.

2.0 ANALYSIS

The EPRI FatiguePro software plant-specific version (Farley) 3.00.01-03342 was used to compute fatigue at these locations for the Farley plant. The version identifiers are shown in Figure 1.

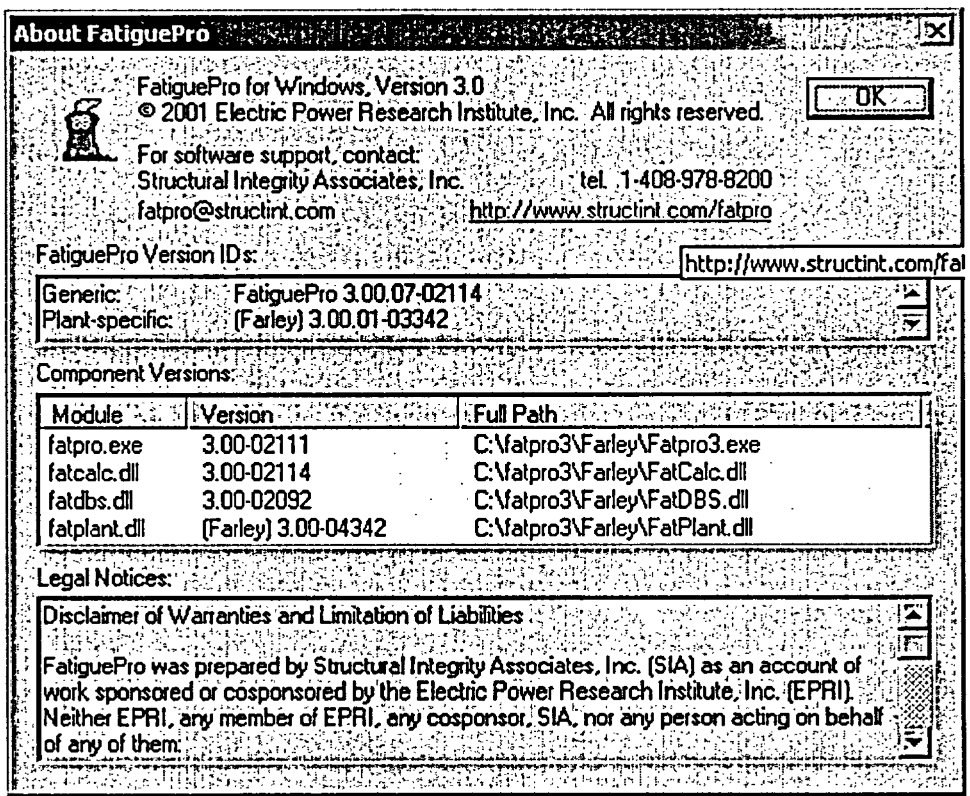


Figure 1. FatiguePro Version Identifiers

Extensive experience with fatigue monitoring has demonstrated that a significant system temperature differential (the difference between pressurizer water temperature and RCS hot leg temperature) is required to produce thermal fatigue in the surge line and lower head. This occurs during plant Heatups and Cooldowns. Other transients such as Reactor Trip due not produce stresses above the minimum fatigue threshold.

Several types of loadings contribute to Heatup and Cooldown transients, including:

5. Internal Pressure
6. Surge line piping thermal expansion
7. Surge line piping thermal stratification
8. Thermal shock, or "insurge/outsurge" temperature transients from flow reversals

FatiguePro computes stresses in various fatigue-sensitive components based on real plant data. The logic for this, called the "transfer functions," is described in detail in the Farley Transfer Function Report [1]. A computational scheme was devised to compute the water temperature at various zones in the surge line and pressurizer lower head based on available temperatures, flows and other applicable instruments to capture any insurge/outsurge effect that the plant may experience during operation.

The following locations in the surge line and pressurizer lower head are monitored:

1. Hot Leg Surge Nozzle (HL_NOZ)
2. Pressurizer Surge Nozzle (SRG_NOZ)
3. Pressurizer Heater Penetration Weld (PZR_Heater_Pen)

One-minute plant data was available for Farley Units 1 and 2 from approximately mid-1996 to mid-2003. The data was converted to FatiguePro CDT format and screened for Heatup and Cooldown transients to be analyzed in FatiguePro.

A Cooldown followed by a Heatup was assumed to represent a cycle or overall transient. The transients considered in this analysis are shown graphically in Appendix A.

Tables 1 and 2 list the cycles analyzed by FatiguePro for each unit. Each CDT data file represents one day of data, and the date is reflected in the name of the file. Each analysis produced an incremental fatigue usage for the locations of interest.

The FatiguePro projects, results databases, and data files are stored on the project CDROM.

Fatigue Usage Projections

Fatigue usage projections were based on the design number of cycles (200 Heatups and Cooldowns each). For the purpose of computing fatigue usage, the incremental usage for a Cooldown/Heatup cycle was assumed to be the average incremental fatigue from the template periods.

The projected fatigue usage (without environmental effects) for the life of the plant is shown on Tables 1 and 2. The pressurizer surge nozzle had the highest fatigue usage computed.

Environmental Effects

The fatigue usage projections discussed above do not include consideration of environmental effects on the fatigue curve.

The effects of environmentally-assisted fatigue are a function of several parameters, including material type, temperature, dissolved oxygen content, strain rate and strain range. These factors, called F_{en} factors) may be computed using equations from References 2 and 3, for stainless steel and carbon/low alloy steels, respectively. These effects on individual CUF load pairs can be as high as a factor of 15.35 for stainless steel when all relevant conditions are present. However, it is typical for the overall effects for all load pairs for a given component location to be a factor of four or less, since environmental effects do not affect all individual load pairs (due to thresholds beyond which environmental effects are negligible).

Based on the fatigue usage projections, and allowable F_{en} multiplier was computed. This is the number that can be multiplied by the fatigue usage projection with the result remaining below 1.0. These numbers are shown on Tables 1 and 2. F_{en} multipliers were calculated for each transients according to the methodology of NUREG/CR-5704 [4] in the accompanying spreadsheet (FENCALCS.XLS). The values are shown on Tables 1 and 2. For Unit 1 the average F_{en} was 7.90. For Unit 2 it was 8.28.

3.0 CONCLUSIONS

Fatigue usage with considerable margin was demonstrated for the design life of the plant (200 Heatups and Cooldowns each). Allowable environmentally assisted fatigue (F_{en}) multipliers were also computed to be 10.48 for Unit 1 and 8.97 for Unit 2. For Unit 1 the average F_{en} was 7.90. For Unit 2 it was 8.28, so both are below the allowable values.

The maximum fatigue usage for the life of the plant (200 Heatups and Cooldowns), including environmental effects, was 0.643 for Unit 1 and 0.975 for Unit 2. The critical location in the surge line and lower head was the Pressurizer Heater Penetration Weld.

Transient	FatiguePro Project	File Begin	File End	PzHlSuNo	PzrHtrPen	PzrSuNo	FEN	PzrHtrPen_FEN
Cooldown and Heatup	U1_1997_03.fpp	19970315.cd1	19970602.cd1	4.68E-06	2.78E-06	5.79E-06	9.06	2.52E-05
Cooldown and Heatup	U1_1998_08.fpp	19980817.cd1	19980903.cd1	1.56E-06	3.82E-06	2.43E-05	8.95	3.42E-05
Cooldown and Heatup	U1_1998_10.fpp	19981017.cd1	19981226.cd1	1.56E-05	1.03E-03	1.25E-04	6.12	6.33E-03
Cooldown and Heatup	U1_2000_03.fpp	20000304.cd1	20000520.cd1	6.04E-06	8.68E-04	9.92E-06	7.45	6.47E-03
~~CD/HU/CD/HU~~	~~U1_2001_10.fpp~~	~~20011006.cd1~~	~~20011114.cd1~~	~~2.01E-05~~	~~4.30E-03~~	~~1.90E-03~~	~~7.85~~	~~3.38E-02~~
		average CD/HU cycle:		6.96E-06	4.77E-04	4.12E-05	7.90	3.21E-03
		Cycles =	200					
		Allowable FEN =	10.47839643					
			Projections:	0.0014	0.0954	0.0082		
			w/ FEN:	1.46E-02	1.00E+00	8.64E-02		0.6427

Table 1. FatiguePro Incremental Fatigue Calculations for Unit 1

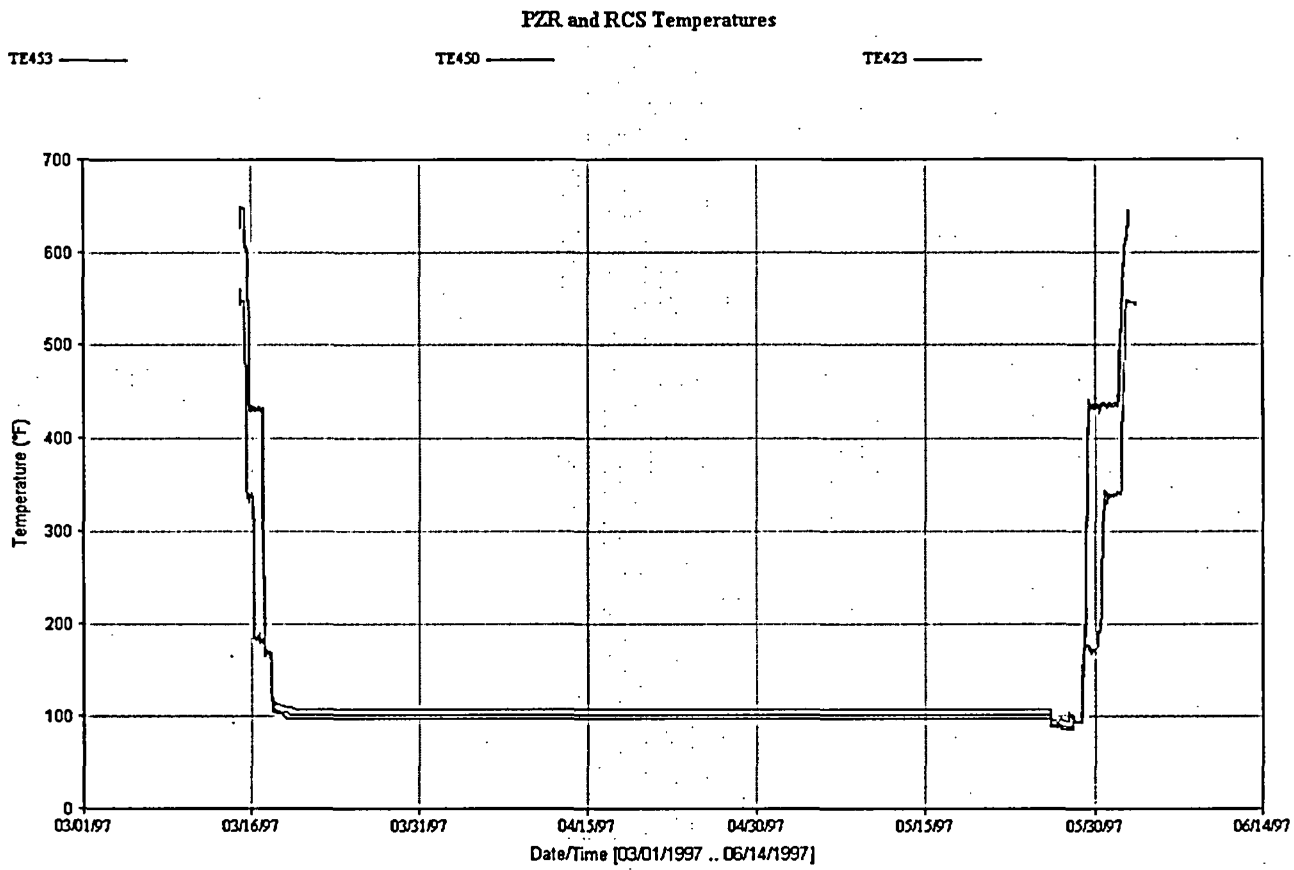
Transient	FatiguePro Project	File Begin	File End	PzHlSuNo	PzrHtrPen	PzrSuNo	FEN	PzrHtrPen
Cooldown and Heatup	U2_1999_10.fpp	19991016.cd2	19991212.cd2	9.56E-06	8.59E-06	1.00E-05	7.66	6.58E-05
Cooldown and Heatup	U2_2001_02.fpp	20010224.cd2	20010505.cd2	1.57E-05	2.21E-03	1.57E-05	8.75	1.93E-02
CD/HU/CD/HU	U2_2002_09.fpp	20020914.cd2	20021027.cd2	1.06E-05	1.64E-05	4.06E-05	8.43	1.38E-04
		average CD/HU cycle:		8.96E-06	5.58E-04	1.66E-05	8.28	4.87E-03
		Cycles =	200					
		Allowable FEN =	8.97					
			Projections:	0.0018	0.1115	0.0033		
			w/ FEN:	1.61E-02	1.00E+00	2.97E-02		0.9750

Table 2. FatiguePro Incremental Fatigue Calculations for Unit 2

4.0 REFERENCES

1. Structural Integrity Report SIR-02-162, Rev. 0, "Transfer Function and System Logic for Fatigue Monitoring System for Farley Nuclear Plant Units 1 and 2."
2. US NRC Report: NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," prepared by Argonne National Laboratory (ANL-98/31).
3. US NRC Report: NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," prepared by Argonne National Laboratory (ANL-97/18).
4. US NRC Report: NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels.

APPENDIX A:
FARLEY HEATUP AND COOLDOWN TRANSIENTS ANALYZED



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Preparer/Date	TDG 7/1/03	TDG 11/5/03	TDG 12/18/03	
Checker/Date	DAG 7/1/03	DAG 11/5/03	TDG 12/18/03	
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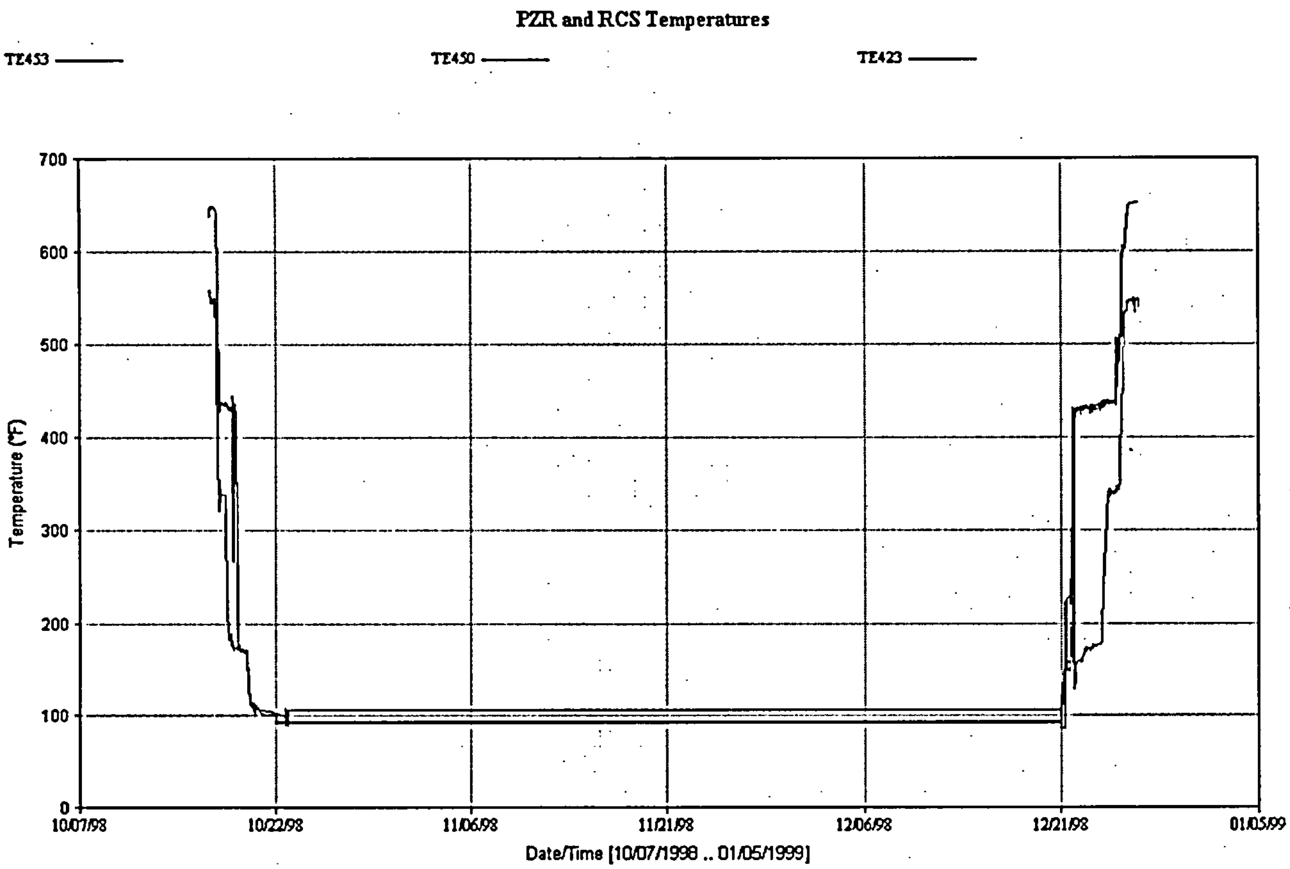

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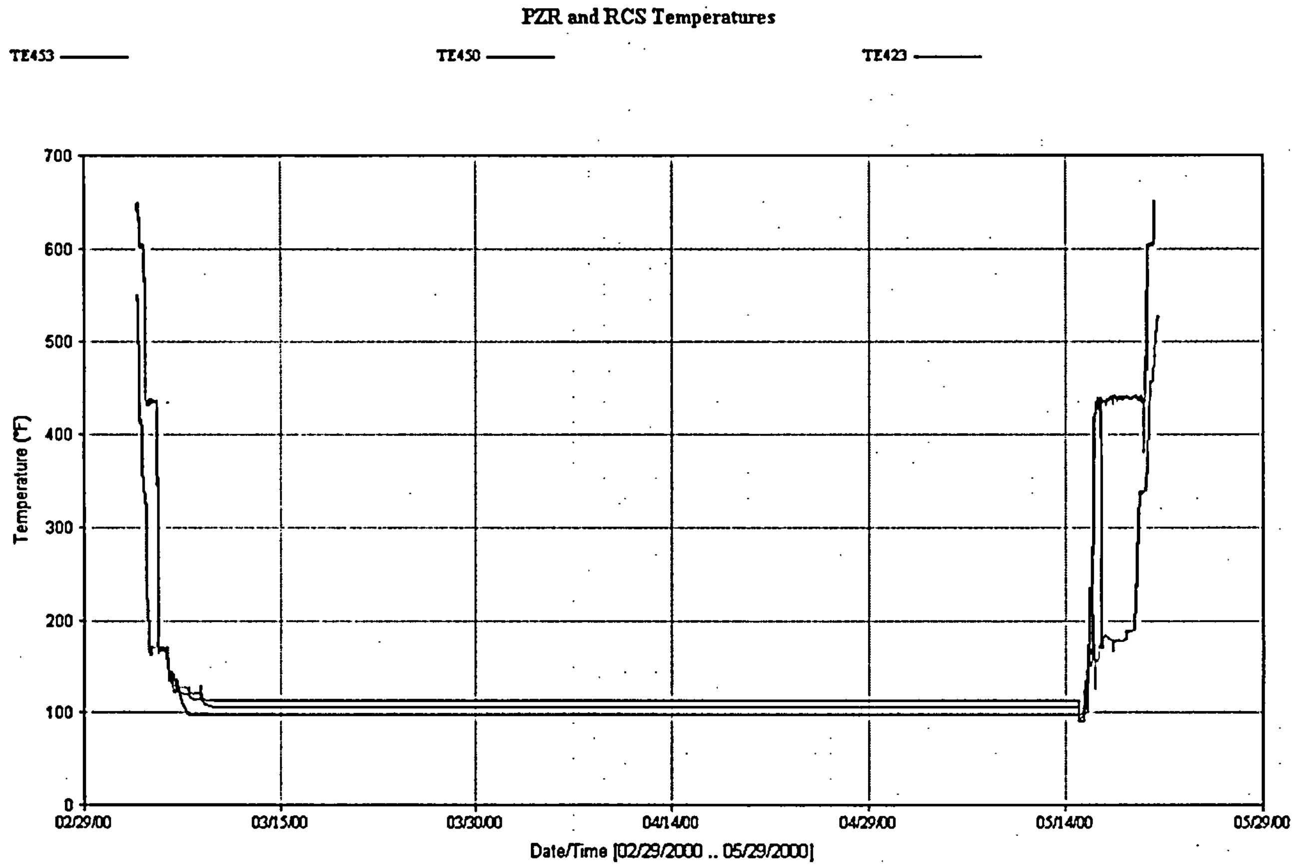

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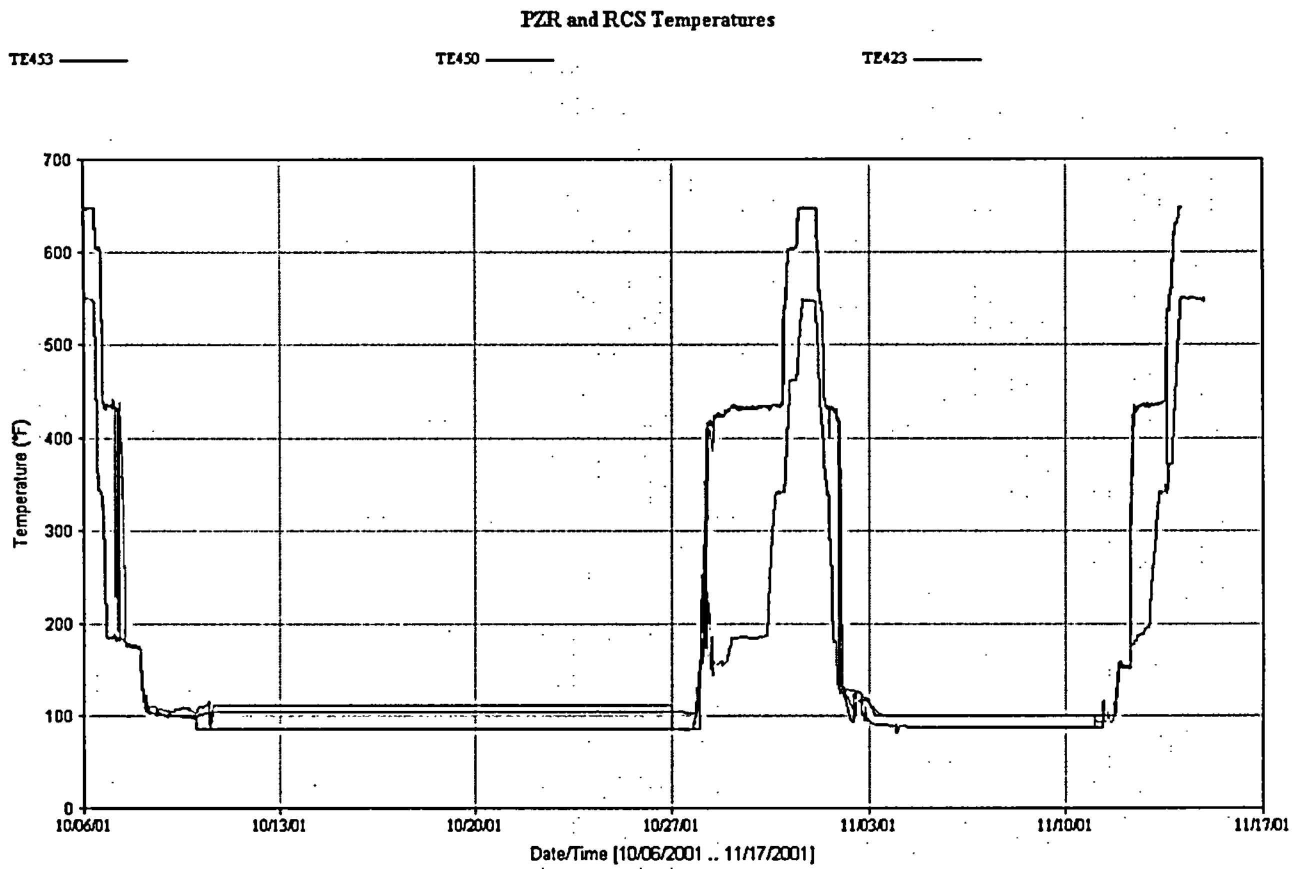
Preparer/Date	TDG 7/1/03	TDG 11/5/03	TDG 12/18/03	
Checker/Date	DAG 7/1/03	DAG 11/5/03	TDG 12/18/03	



	Revision	0	1	2	
	Preparer/Date	TDG 7/1/03	TDG 11/5/03	TDG 12/18/03	
	Checker/Date	DAG 7/1/03	DAG 11/5/03	TDG 12/18/03	
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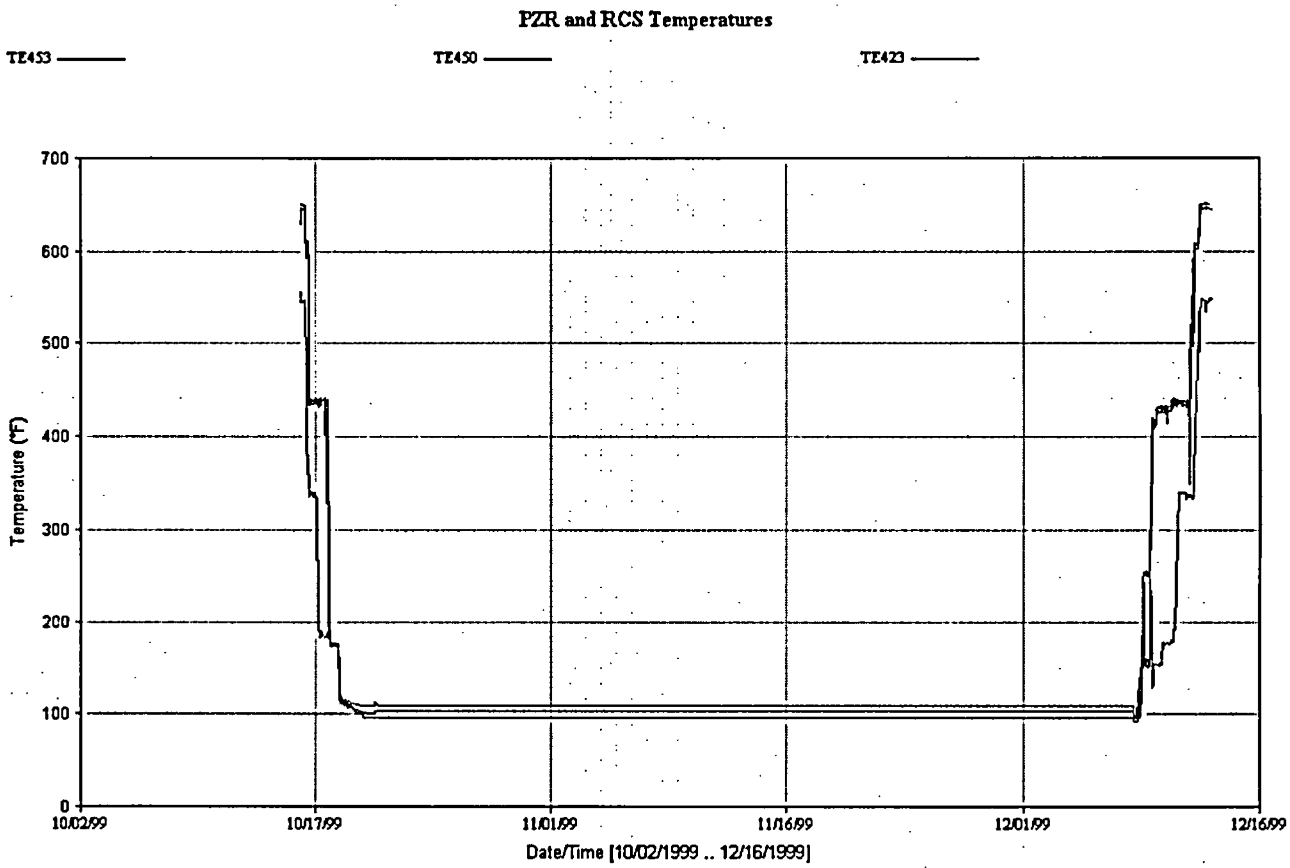
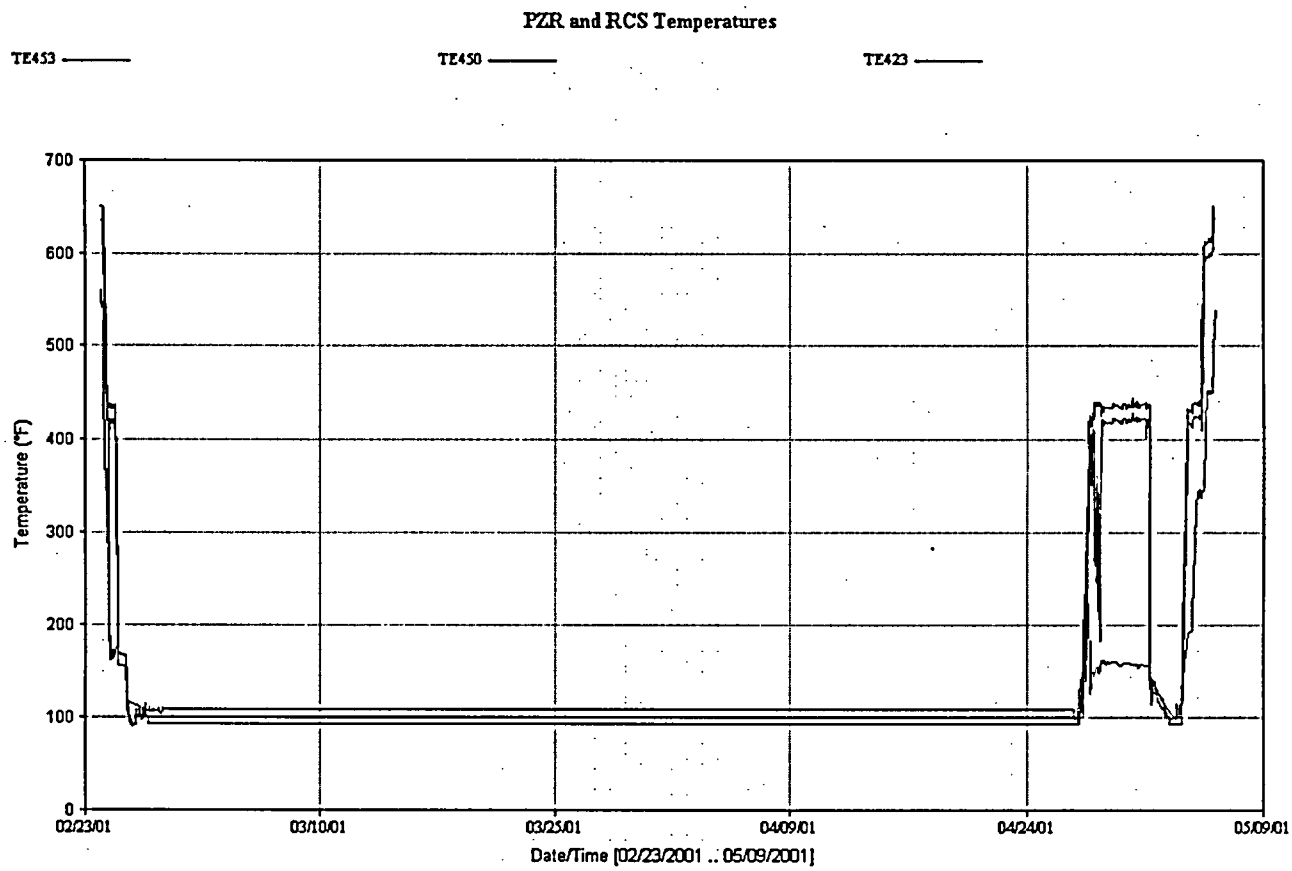
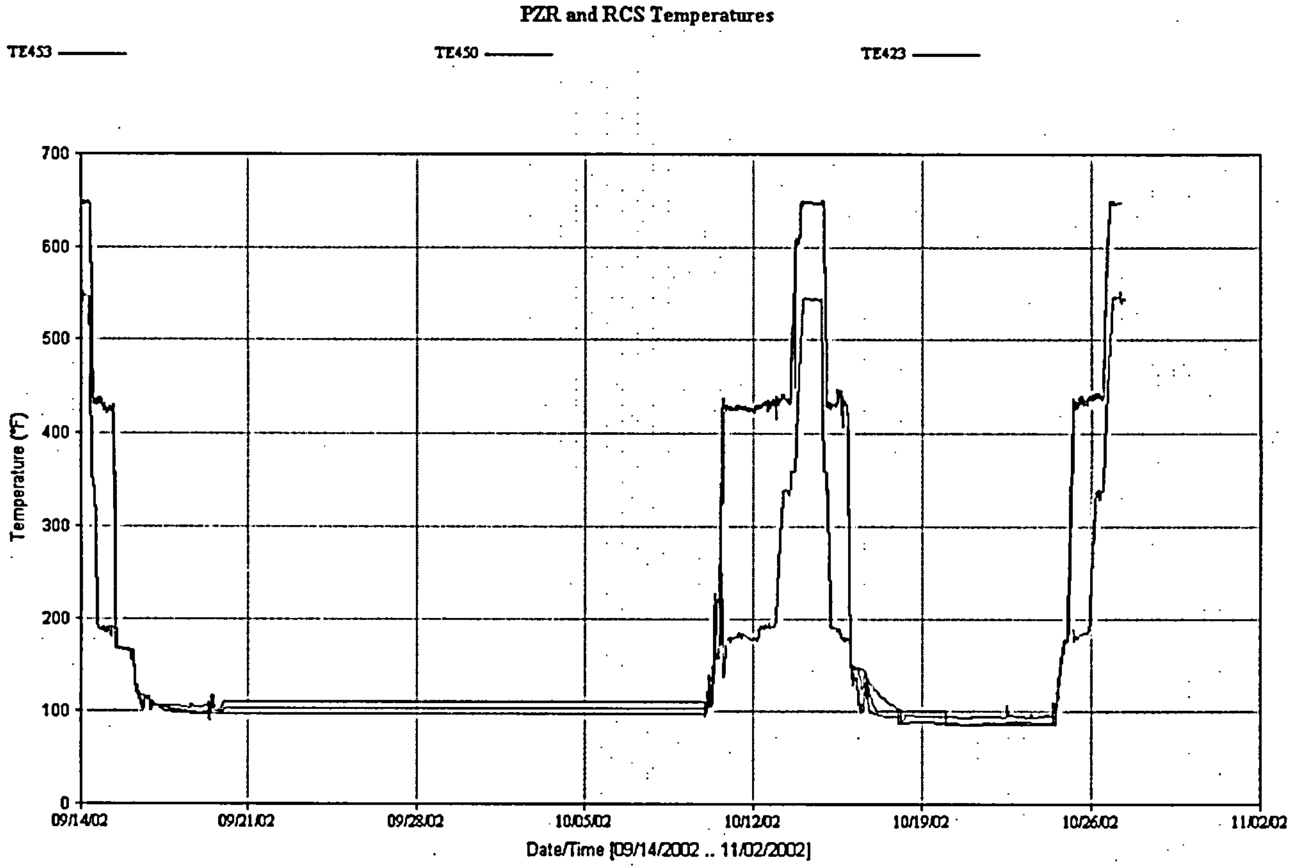


	Revision	0	1	2	
	Preparer/Date	TDG 7/1/03	TDG 11/5/03	TDG 12/18/03	
	Checker/Date	DAG 7/1/03	DAG 11/5/03	TDG 12/18/03	



Revision	0	1	2	
Preparer/Date	TDG 7/1/03	TDG 11/5/03	TDG 12/18/03	
Checker/Date	DAG 7/1/03	DAG 11/5/03	TDG 12/18/03	



Revision	0	1	2	
Preparer/Date	TDG 7/1/03	TDG 11/5/03	TDG 12/18/03	
Checker/Date	DAG 7/1/03	DAG 11/5/03	TDG 12/18/03	