TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

January 6, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

In the Matter of the	)	Docket Nos. 50-259
Tennessee Valley Authority	)	50-260
		50-296

By my letter to you dated March 4, 1982 we submitted a request for technical specification change, TVA BFNP TS 172, to allow single recirculation loop operation of the Browns Ferry Nuclear Plant. Additionally, by my letter to you dated September 3, 1982 we submitted a response to an NRC request for additional information regarding single-loop operation. Enclosed is our response to the NRC request for additional information forwarded to TVA by letter from D. B. Vassallo to H. G. Parris dated November 8, 1982.

We regret that due to the theoretical and developmental nature of the subcritical bifurcation topic discussed in NUREG/CR-1718, TVA does not have the resources available at this time to satisfactorily apply this to the Browns Ferry reactor systems. The concerns of the staff regarding stability are understandable, and we hope that the response enclosed will show that the flux variations experienced at Browns Ferry were not related to reactor stability and thus preclude the necessity of addressing subcritical bifurcation in order to obtain NRC approval of single-loop operation.

During the Browns Ferry single-loop operating experience of fall 1978, jet pump and neutron flux noise greater than normal was observed above 65-percent pump speed and 60-percent power. This lead TVA to administratively limit operating conditions to 60-percent pump speed pending further investigation. Based on the results of testing performed by TVA and General Electric Company, our position is that the observed flow and neutron flux variations

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Mr. Harold R. Denton January 6, 1983

were the result of irregular jet pump flow due to the effects of higher than normal individual flow in combination with off-normal hydraulic conditions in the annulus upon jet pump performance. TVA and GE do not consider this to be a safety problem.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills

L. M. Mills, Manager
Nuclear Licensing

Subscribed and sworn to before
me this 6th day of January 1983.

Paulette H. White
Notary Public

My Commission Expires 9-5-84

Enclosure
cc (Enclosure):
U.S. Nuclear Regulatory Commission
Region II
ATTN: James P. O'Reilly, Regional Administrator
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Mr. R. J. Clark
Browns Ferry Project Manager
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, Maryland 20814

ENCLOSURE

Browns Ferry Nuclear Plant Units 1, 2, and 3
Response to Request for Additional Information
Single Loop Operation

Question 1

The frequency of the oscillation at Browns Ferry Unit No. 1 (BF1) (0.3-0.5 Hz) is essentially the same as the characteristic frequency of BWR density-wave oscillations. The postulation that the increased flow noise is primarily due to the increased flow rate and inherent flow noise in the active jet pumps is plausible but we require substantive proof that this is the case. Experimental or calculational results which predict that the observed flow fluctuations can produce the observed magnitude and frequency of neutron flux would provide sufficient proof as long as the power-void feedback can be shown to be small. Calculational results which predict the observed frequency for the BF1 conditions or experimental results which are extrapolated to the BF1 conditions would provide sufficient proof. For arguments based on calculational methods, the codes used, boundary conditions, and calculational assumptions should be provided together with major input and output values. For arguments made based on experimental results, experimental values should be provided and the assumptions made to extrapolate these results to BF1 conditions should be explained.

Response 1

Figure 1 is a strip chart recording of plant parameters obtained during Browns Ferry single-loop testing in the fall of 1978. Pertinent data has been manually enhanced due to poor quality of the original. Figure 2 consists of a manual reproduction of the same data traces neglecting what are determined by visual inspection to be higher order effects for the sake of clarity. Inspection of the active jet pump 1-10 total flow and APRM D signals shows clear dependence of one signal upon the other. The magnitude of flux in response to flow exhibits a gain in amplitude which is the expected behavior when driven at or near the resonant frequency of approximately 0.5 Hz. The observed core flow variations therefore could produce the observed flux variations.

Figure 2 shows traces of two typical active jet pumps, the total of all ten active jet pumps, and APRM D flux together as a function of time. It can be seen by inspection that neither of the individual jet pump flow variations shows any apparent relationship to the APRM D or total jet pump flow variations nor do they appear to relate to each other. All jet pumps are coupled to the inlet plenum in nearly the same way due to the rotational symmetry designed into the reactor vessel and core. They therefore should be expected to exhibit similar wave forms and be in phase with each other if driven from the discharge end by a common phenomena (i.e., power-void oscillations). Because the individual jet pump flow variations show no relationship to the power-void characteristic signal (i.e., flux) and their signals show no common oscillation driving them from the discharge end, it is proven that jet pump noise observed during Browns Ferry single-loop operation is not driven by power-void feedback.

Question 2

Based on the Browns Ferry operating experience and the generic evaluations and studies the General Electric Company (GE) has performed, justify that single loop operation is safe and acceptable within the limits prescribed by GE in the specific licensing reports in which GE has analyzed the Browns Ferry units. In your answer, demonstrate that for limit cycle oscillations of flow and neutron flux that bound the magnitude of those observed and expected in single loop operation, the safety limits are not exceeded. The evaluations should include the bounding conditions of flow, temperature and pressure, and any uncertainties that are predicted for these conditions. Also show that the critical heat flux correlation used is valid.

Response 2

GE NEDO-24236, which is referenced in our original amendment request, evaluates the affected safety limits with oscillations of the magnitude observed during Browns Ferry single-loop operation using the bounding initial conditions attainable in single-loop mode. A CPR operating limit penalty and core flow measurement error are established therein and included as operating restrictions in the proposed amendment.

Question 3

Discuss the possible reasons for, and contributing factors to, the observed flow and neutron flux variations observed in the Browns Ferry 1 operating experience.

Response 3

TVA believes that the neutron flux variations observed were driven by variations in total core flow. The total core flow variations, we believe, were the sum of random individual jet pump variations. The individual jet pump variations are probably a result of higher individual flow encountered during single-loop mode possibly combined with off-normal hydraulic conditions at the suction end of the pumps such as crossflow and subcooling which are the result of moisture separator flow, feedwater flow, and inactive loop backflow into the annulus area opposite the active loop which must flow around the annulus to the active side in order to continue its cycle through the system.

Question 4

Crossflow components in the downcomer region may have occurred as a result of reverse flow in the inactive jet pump bank and may have contributed to the flow oscillations recorded in the individual jet pumps of the Browns Ferry plant during single-loop operation.

Provide vibration data showing that the structural integrity of the jet pumps and other vital vessel internals is not threatened by single-loop operation under these conditions of reverse flow.

Response 4

During preoperational and startup testing of Browns Ferry unit 1 between December 1972 and March 1974, extensive vibration data was recorded by General Electric Company in order to demonstrate the adequacy of the 251-inch diameter vessel design. This testing is documented in GE document number 386HA219 entitled Browns Ferry-1 Vibration Measurements. Pertinent documents are included in this attachment.

Test condition H-25 was run at 91-percent pump speed on loop B, 0 on loop A at about 70-percent reactor power and 54-percent core flow. Included in the attachment are data tables giving vibration sensor locations and operating data for this condition. The test report document concludes that all vibratory responses are within the acceptance criteria for all test conditions. Test condition H-25 bounds the conditions expected in single-loop operation and those tested during Browns Ferry single-loop experience. The results of this testing demonstrate that jet pumps and other vital vessel internals are not threatened by single-loop operation under conditions which bound the values expected during operation.

Question 5

In order to compare single-loop operation with two-loop operation, power to flow ratios should be evaluated in addition to jet pump flows. Provide available data showing a comparison of power to flow ratios and expected decay ratio ranges for single-loop operation and two-loop operation.

Discuss the expected core inlet flow distribution/symmetry during single-loop operation (for example, the effect on the hot channel vs. the average channel).

Response 5

The ratio of core thermal power to total core flow is identical for two-loop and single-loop operation. There is some uncertainty in the measurement of total core flow due to inactive loop backflow measurement capability. NEDO-24236 (Browns Ferry Nuclear Plant Units 1, 2, and 3 Single Loop Operation) section 2.1 discusses the method for conservatively adjusting total core flow readings to account for backflow error.

NEDO-24236 also discusses single-loop decay ratios in section 4. At the same total core flow and thermal power, the decay ratios of single- and two-loop responses are identical. Stability decreases as flow decreases along a constant rod density line. Since one pump is capable of achieving less flow than two pumps, it is possible to achieve less stability in that condition if the pump speed is set at minimum. However, at higher power operating conditions, normal operating constraints ensure at least the same stability as for two-loop operation. In any event, natural circulation mode remains the limiting case of merit regarding stability.

Core power symmetry was measured using a standard refueling test procedure during the single-loop operation of fall 1978. The data was compared to beginning of cycle test data in two-loop operation. The test revealed complete power symmetry with the exception of minor asymmetries which was also detected during the earlier two-loop operation test.

Question 6

Provide a power flow map which incorporates data points of the BF-1 single-loop operating history. Clearly designate the range where oscillations were experienced.

Response 6

Figure 3 contains a power flow map for the Browns Ferry single-loop experience of the fall 1978. Total core flow is a calculated value which conservatively accounts for inactive loop backflow, and it therefore should be recognized that there is some inaccuracy in this value. This is particularly true in the region of 45-50 percent of rated flow where backflow is first established. The active loop flow axis for the APRM peak-to-peak variation graph is placed to approximately correspond with the associated total core flow values for the operating map.

Question 7

It was shown in NUREG/CR-1718 that finite amplitude oscillations can trigger a subcritical bifurcation (i.e., divergent oscillation) in a region of linear stability. The larger the amplitude of the oscillations, the greater is the potential for divergent oscillations. Show that the core is stable for the largest amplitude oscillation which is predicted. Explain how nonlinear effects are included.

Response 7

As noted in the cover letter, TVA is not able to respond to this concern at this time.

Question 8

Is TVA aware of any data or experience on single-loop operation at high flow and power from any other BWRs other than Browns Ferry? If so, discuss the applicability of this data, particularly as to whether it affects conclusions drawn from the Browns Ferry data?

Response 8

TVA has not been able to obtain any single-loop operating data from another BWR which could be compared to that recorded at Browns Ferry.

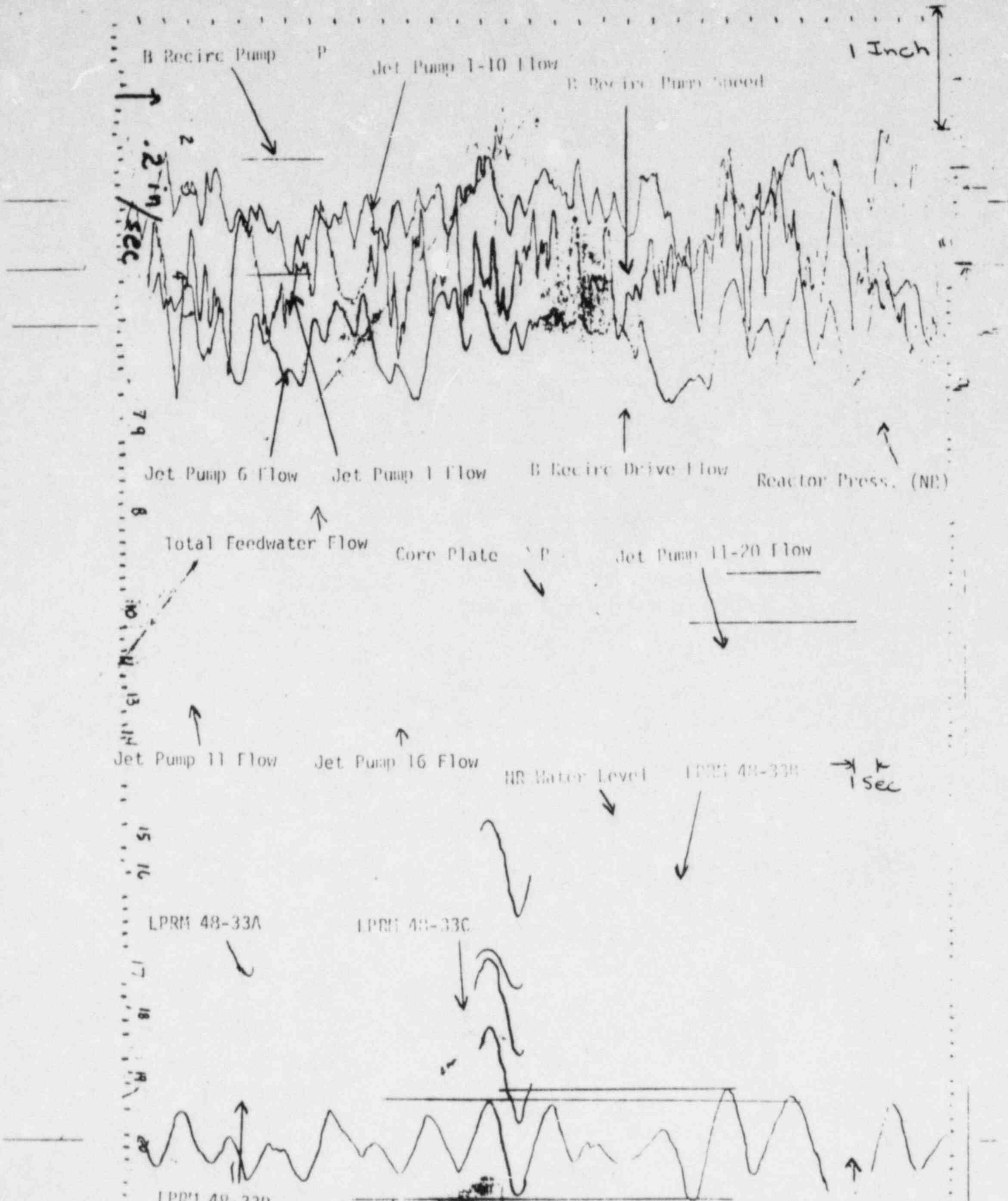

FIGURE 1: Single loop data at 50.7% flow, 61.1% power.

.2 In./Sec. CHART SPEED

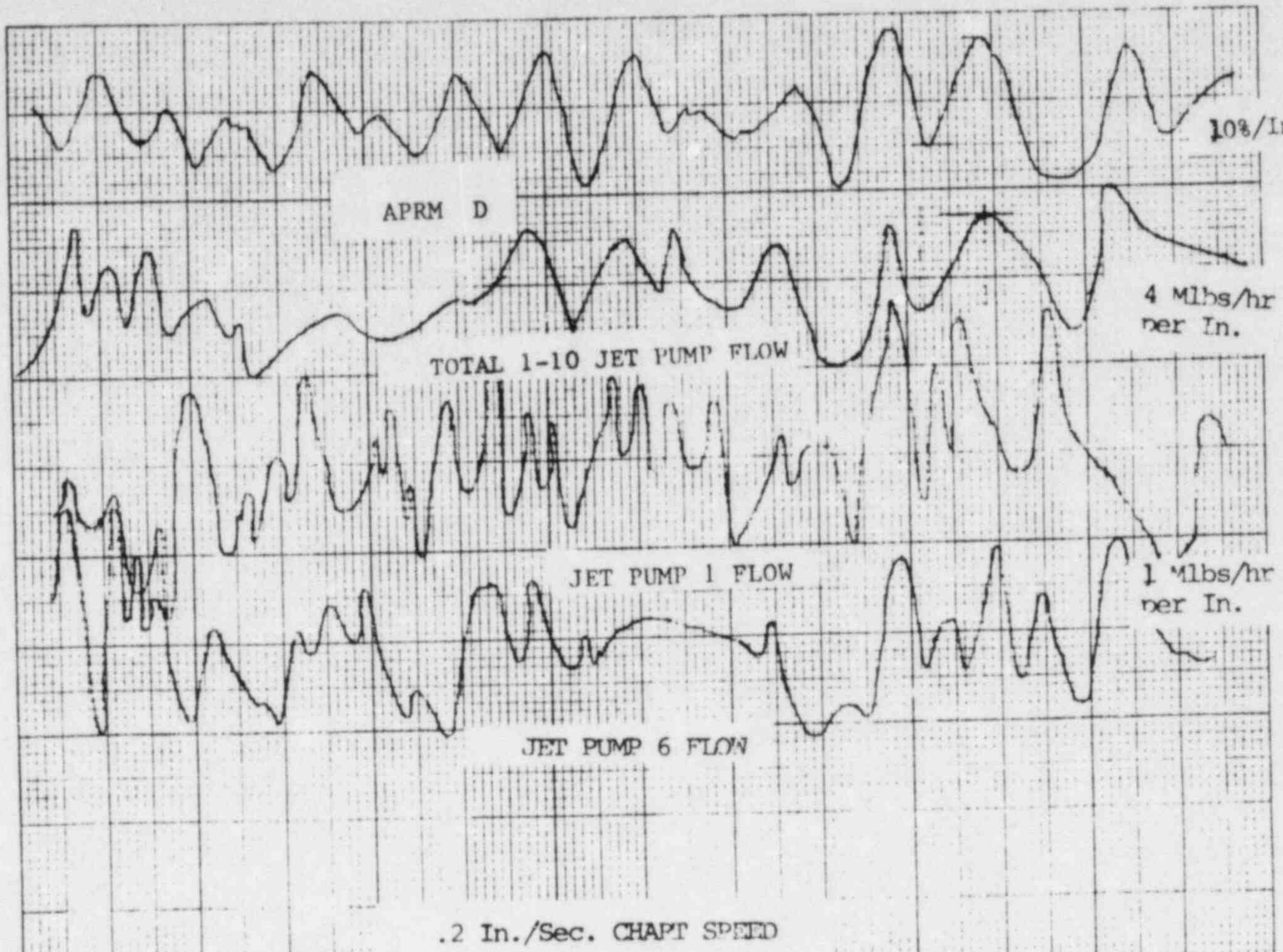

FIGURE 2: Single loop data at 50.7% flow, 61.1% power, w/o 2nd order effects.

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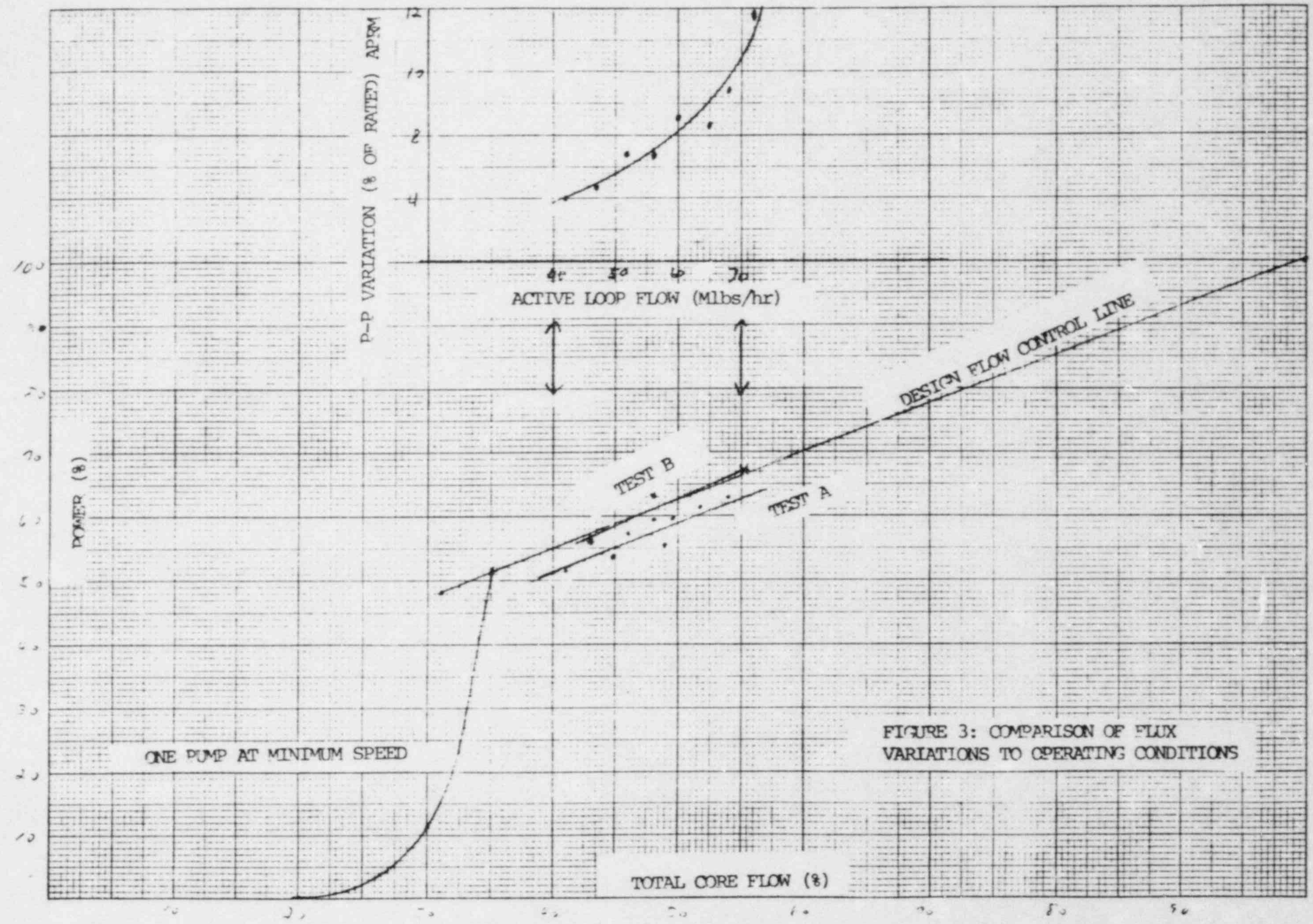

FIGURE 3: COMPARISON OF FLUX VARIATIONS TO OPERATING CONDITIONS

BROWN'S FERRY-I VIBRATION MEASUREMENTS

ABSTRACT

The vibration response of the Brown's Ferry-I prototype 251 product line reactor internals subject to a wide range of flow and power conditions was measured during the pre-operational and startup testing program conducted between December, 1972 and March, 1974. This report contains the vibration data and evaluation. Results show that all vibration amplitudes are within acceptable limits, thus, demonstrating the adequacy of the BWR/4 and BWR/5 internals designs in the plants of 251" vessel size.

I. INTRODUCTION

Since 1966 it has been the practice of General Electric Company to make vibration measurements on BWR internals during pre-operational and start-up testing of new reactor designs, to demonstrate and confirm the mechanical integrity of critical components under actual operating conditions. This vibration measurement program constitutes the final verification of the design. The vibration measurements together with the inspection of internals following pre-operational testing are designed to meet the intent of NRC Regulatory Guide 1.20 (Vibration Measurements on Reactor Internals).

The vibration measurement program broadly consists of: 1) Developing a set of limiting criteria for each component based on its anticipated vibration response characteristics, and 2) Conducting flow and power tests in the reactor under the full range of operating conditions to ensure that the maximum vibration responses are within acceptable limits.

Brown's Ferry-1 is the prototype reactor design for subsequent 251-in. size BWR/4's and BWR/5's including Brown's Ferry-2 and Brown's Ferry-3. These plants will be subjected to a vibration measurement program of reduced scope, or to a flow test and inspection program, in response to provisions of RG 1.20 for plants similar to the prototype. The scope of this confirmatory program will be extended when needed to evaluate specific components which may differ from the prototype components in some respect.

Appendix A is the engineering specification which defines testing objectives, test conditions and procedures, and requirements for vibration sensors, read-out equipment, and other hardware. Additional detailed information on sensor locations and instrumentation specifications is given in Appendix B.

Results presented and discussed in this report were obtained from the following test series:

1. Pre-operational cold flow test without fuel (Appendix A, Section 6.2.2) December 18-21, 1972

2. Pre-operational cold flow check test with fuel (Appendix A, Section 6.2.4) August 29-30, 1973

3. Startup Tests (Appendix A, Section 6.3) November 15, 1973 through June 1, 1974.

The pre-operational cold flow test included 48 hours of running at rated volumetric core flow, in addition to other test conditions. Following this test, a vessel internals inspection was conducted in accordance with procedures defined in Section 6.2.3 of Appendix A. The inspection revealed no wear or damage to reactor internals or other evidence of vibration.

The vibration measurement procedure consists of recording the time-history response of the vibration sensors at each test condition on a multi-channel chart recorder. The data is also recorded on magnetic tape for subsequent use on the spectrum analyzer and other computational equipment when special data reduction is needed. The chart data is then reduced by visual inspection to yield frequencies and response amplitudes for comparison with the criteria. In addition to ensuring that the limiting criteria are not exceeded, evaluation of the data also provides an additional insight into the basic mechanisms that cause the observed response.

The evaluation of the test results and a brief discussion on the identifiable causal factors for the internals response are included in the following section.

II. RESULTS AND DISCUSSION

The maximum observed vibratory responses in the BF-1 internals are within the acceptance criteria for all test conditions. Since the test conditions cover a wide range of possible operating conditions of the reactor, it is concluded that major components of the BF-1 reactor internals that have been instrumented may be expected to retain their mechanical integrity during the operating life of the plant.

Results are presented first for balanced flow conditions, followed by unbalanced flow conditions. Normal reactor operation is at balanced conditions, with both recirculation loops at the same flow. Unbalanced flow conditions include one-loop operation, flow transients (pump trips) and dissimilar loop flows.

II-1. RESULTS FOR BALANCED FLOW

Table II-1 lists the maximum responses of each component as a percent of the acceptance criteria for the balanced flow conditions tested. The upper bolt guide ring had a peak displacement of 21% of the criteria during hot balanced operation. The jet pump assembly (riser brace) reached 16% of the criteria at 113% of rated volumetric flow. The in-core guide tubes measurement peak strain was 14% of the criteria while the control rod drive stub tube only reached 3% of its criteria.

The hot flow results summarized above are the most significant with respect to plant operation, but it is noted that generally the cold flow conditions result in higher vibration amplitudes. Section II-6 discusses the cold and hot comparison more extensively.

II-2. RESULTS FOR UNBALANCED FLOW

Table II-2 lists the maximum responses of each component as a percent of the steady state balanced flow acceptance criteria for unbalanced flow conditions.

During hot testing the riser brace motion reached 45% of the criteria during a transient test. The jet pump riser top maximum motion was 37% of the criteria and the incore guide tube reached a maximum of 5% during 85% and 53% pump speeds on A and B pump respectively. The shroud-separator assembly reached a maximum of 20% of the criteria during one pump operation.

It is noted that generally the cold flow responses are higher than the hot flow responses, as was the case for balanced flow conditions.

II-3. BASIS FOR ACCEPTANCE CRITERIA

The applicable criteria for the tabulated maximum measurements are listed in Table II-3. These criteria represent the maximum amplitude of vibration (displacement or strain) permissible at each sensor location. The criteria are developed as follows:

1) The reactor internals system is represented by a mathematical model consisting of a series of lumped masses connected by weightless beams with appropriate stiffness properties. The masses include the hydrodynamic component to account for the effect of water.

2) The mathematical model forms the input for the SAMIS computer program, which is capable of i) generating the stiffness matrix and the mass matrix, for the system, and ii) calculating the mode shapes (eigenvectors) and frequencies (eigenvalues) for the natural vibration modes.

3) Modal stresses are calculated from the element stiffness matrix and the mode shapes. This yields the distribution of stresses in the system for each modal frequency of vibration.

4) By limiting the maximum steady-state vibration stress in the system to an allowable alternating stress (S_a) of $\pm$ 10,000 psi, the limiting amplitude of vibration (displacement or strain) at the sensor location is obtained by linear scaling.

In view of the conservatism built into the calculation of allowable criteria and the fact that the only responses exceeding 30% of the criteria occurred at prohibited operating conditions, the BF-1 reactor internals for which vibration measurements were conducted are adequately designed to withstand flow-induced vibrations over the entire plant life.

In addition to satisfying the design criteria, the test data yielded some useful engineering information which may be used to improve vibration prediction methods and to modify the scope of prototype testing in future plants.

II-4. COMPARISON TO VIBRATION PREDICTIONS

The Brown's Ferry-1 vibration predictions (DAR 150, 383HA810) were determined through statistical techniques using available test data from differing plant designs. Correlation functions relating response to reasonable physical variables such as flow, power and mixing ratios were developed (by empirical means) and used to extrapolate data to a commom statistical base. The prediction is a range which will include the measured amplitude with 75% confidence, based on the Tchebycheff inequality in statistical analysis.

Table II-4 shows the maximum test measurements compared to the predicted range, for the selected component modes which are usually observed in all plants. This comparison is limited to rated flow conditions. Of the five modes for which data were obtained, four measurements fall in the range

predicted, which is consistant with the prediction basis.

II-5. DATA CORRELATIONS

Using the BF-1 test results, an attempt was made to correlate the jet pump response to nozzle velocity and shroud/separator response to the core flow. The jet-pump response in the tangential and radial directions shows a positive correlation to (nozzle velocity)^n where n varies from 1 to 4 for various operating conditions. The shroud response did not show any consistent correlation with core flow. However, a slight peaking of the already small response was noted in the 70-90% range of rated core flow.

The separator response showed the most interesting correlation to the core flow. The centrifugal excitation from the swirl phenomenon in the separators has been analyzed in the past. And, nomograms relating to the excitation frequency (from the centrifugal effect) to core flow and number of separators have been developed. Fig. II-1 shows the excitation frequencies plotted against core flow. The BF-1 data showed a marked peaking when the excitation frequency coincided with the fundamental frequency of the separator (5.2 Hz)- this occurs only during cold flow conditions. However, under hot conditions, with core flow ranging from 50-100% of rated flow, the excitation frequencies passed through the second and third natural modes of vibration of the shroud-separator assembly, yet there was no noticable magnification. A possible explanation could be that the two-phase flow is turbulent and, hence, has no single frequency for sustained periods.

Another possibility is that the swirling in the individual separators is not in phase long enough to build up significant amplitudes.

II-6. COMPARISON OF RESULTS FOR COLD AND HOT TESTING

The Brown's Ferry-1 vibration measurements show that in most cases the cold tests without fuel yielded the higher response compared to hot tests when all test conditions were considered. In the few cases where the hot tests yield a higher maximum response than the cold tests, the difference was less than 10%. Figures II-2 through II-10 illustrate this behavior. Comparisons are made on the basis of response amplitude as a percent of criteria. Table II-5 comparing the cold and hot data at rated flow again shows the cold test conditions to be the more severe. These comparisons indicate that the cold flow test provides substantial vibration excitation and is generally a conservative representation of normal reactor operation of this regard.

Table III-2

SENSOR LOCATIONS

SENSOR	LOCATION	ORIENTATION	ELEVATION	AZIMUTH
D1	Shroud To Vessel	Tangential	396.00	3° 45'
D2	Shroud to Vessel	Radial	404.75	3° 45'
D3	Shroud to Vessel	Tangential	396.00	183° 45'
D4	Shroud to Vessel	Radial	404.75	183° 45'
D5	Riser Top to Vessel	Tangential	344.00	90°
D6	Riser Top to Vessel	Vertical	345.75	150°
D7	Riser Top to Vessel	Tangential	344.00	150°
D8	JPI Elbow to Vessel	Radial	346.00	158°
D9	JP2 Elbow to Vessel	Radial	346.00	142°
D10	JPI Diffuser Top to Vessel	Tangential	207.70	160°
D11	JPI Diffuser Top to Vessel	Radial	209.25	160°
D12	JP2 Diffuser Top to Vessel	Tangential	207.70	140°
D13	JP2 Diffuser Top to Vessel	Radial	209.25	140°
A1	Upper Bolt Guide Ring	Tangential	587.12	0°
A2	Upper Bolt Guide Ring	Tangential	587.12	120°
A3	Upper Bolt Guide Ring	Tangential	587.12	240°
T1	Under Core Plate	-	179.87	Center
P1	Across Shroud Head	102.00 Radius	416.00	0°
P2	Across Core Plate	92.00 Radius	207.375	170°
P3	Across Shroud Wall	Not Applicable	153.00	135°
S1	JP Riser Brace on Top wide face at Outside edge	1 inch from Vessel Wall	299.25	30°

Table III-2 (continued)

SENSOR	LOCATION	ORIENTATION	ELEVATION	AZIMUTH
S2	JP Riser Brace on Bottom wide face at Inside edge	1 inch from Vessel Wall	299.25	30°
S3	JP Riser Brace on Top wide face at Outside edge	1 inch from Vessel Wall	299.25	60°
S4	JP Riser Brace on Bottom wide face at Inside edge	1 inch from Vessel Wall	299.25	60°
S5	JP Riser Brace on Top wide face at Outside edge	1 inch from Vessel Wall	299.25	90°
S6	JP Riser Brace on Bottom wide face at Inside edge	1 inch from Vessel Wall	299.25	90°
S7	JP Riser Brace on Narrow outside edge	Not applicable	299.25	120°
S8	JP Riser Brace on Bottom wide face at Center	Not applicable	299.25	120°
S9	JP Riser Brace on Narrow inside edge	Not applicable	299.25	120°
S10	JP Riser Brace on Narrow outside edge	Not applicable	299.25	150°
S11	JP Riser Brace on Bottom wide face at Center	Not applicable	299.25	150°
S12	JP Riser Brace on Narrow Inside edge	Not applicable	299.25	150°
S13	JP Riser Brace on Top wide face at outside edge	Not applicable	299.25	210°
S14	JP Riser Brace on bottom wide face at Inside edge	Not applicable	299.25	210°
S15	JP Riser Brace on Top wide face at Outside edge	Not Applicable	299.25	240°
S16	JP Riser Brace on Bottom wide face at Inside edge	Not applicable	299.25	240°
S17	JP Riser Brace on Top wide face at Outside edge	Not applicable	299.25	270°
S18	JP Riser Brace on Bottom wide face at Inside edge	Not applicable	299.25	270°
S19	JP Riser Brace on Narrow Outside edge	Not applicable	299.25	300°

Table III-2 (continued)

SENSOR	LOCATION	ORIENTATION	ELEVATION	AZIMUTH
S20	JP Riser Brace on Narrow Inside edge	Not applicable	299.25	300°
S21	JP Riser Brace on Bottom wide face at Center	Not applicable	299.25	300°
S22	JP Riser Brace on Narrow Outside edge	Not applicable	299.25	330°
S23	JP Riser Brace on Narrow Inside edge	Not applicable	299.25	330°
S24	JP Riser Brace on Bottom wide face at Center	Not applicable	299.25	330°
S25	In-Core Guide Tube	X48-Y53 45° CCW from Radial Vector	2.50 above Weld in Bottom Head	
S26	In-Core Guide Tube	X48-Y53 45° CW from Radial Vector	2.50 above Weld in Bottom Head	
S27	CRD Stub Tube	X50-Y51 on Radial Vector	2.50 above Weld in Bottom Head	
S28	CRD Stub Tube	X50-Y51 90° CW Radial Vector	1.62 above Weld in Bottom Head	
S29	Shroud Support Leg	Center Narrow Side	104.00	55°
S30	In-Core Guide Tube	X56-Y33 45° CCW Radial Vector	2.50 above Weld in Bottom Head	
S31	In-Core Guide Tube	X56-Y33 45° CW Radial Vector	2.50 above Weld in Bottom Head	
S32	CRD Stub Tube	X54-Y31 45° CCW Radial Vector	1.62 above Weld in Bottom Head	
S33	CRD Stub Tube	X54-Y31 45° CW Radial Vector	1.62 above Weld in Bottom Head	

Table III-2 (continued)

SENSOR	LOCATION	ORIENTATION	ELEVATION	AZIMUTH
S34	CRD Stub Tube	X58-Y31 45° CW Radial Vector	1.62 above Weld in Bottom Head	
S35	CRD Stub Tube	X58-Y31 45° CCW Radial Vector	1.62 above Weld in Bottom Head	
S36	In-Core Guide Tube	X56-Y17 45° CCW Radial Vector	2.50 above Weld in Bottom Head	
S37	In-Core Guide Tube	X56-Y17 45° CW Radial Vector	2.50 above Weld in Bottom Head	
S38	Shroud Support Leg	Center Narrow Side	104.00	115°
S39	CRD Stub Tube	X54-Y15 45° CW Radial Vector	1.62 above Weld in Bottom Head	
S40	CRD Stub Tube	X54-Y15 45° CCW Radial Vector	1.62 above Weld in Bottom Head	
S41	In-Core Guide Tube	X40-Y17 on Radial Vector	3.38 above Weld in Bottom Head	
S42	In-Core Guide Tube	X40-Y17 40° CW Radial Vector	3.38 above Weld in Bottom Head	
S43	CRD Stub Tube	X42-Y03 45° CCW Radial Vector	1.62 above Weld in Bottom Head	
S44	CRD Stub Tube	X42-Y03 45° CW Radial Vector	1.62 above Weld in Bottom Head	
S45	Shroud Support Leg	Center Narrow Side	104	185°
PE1	Recirculation Pump A Speed			
PE2	Recirculation Pump B Speed			

GENERAL ELECTRIC

NUCLEAR ENERGY DIVISION

DOCUMENT NO. 386HA219 REV. NO. 0
SHEET NO. 61

Table III-1 (continued) (TRIP?

REACTOR OPERATING CONDITIONS

IDENTIFICATION NO.		H-24	H-25	H-26	H-27
DATE		5-31-74	5-31-74	5-31-74	5-31-74
TIME		0445	0517	2100	2230
SPECIAL (SEE FOOTNOTES)		6		6	
POWER THERMAL	MWt	3246	—	3282	2357
	%	99.6	—	99.7	71.6
ELECTRICAL	MWe	1093	742	1090	758
TOTAL CORE FLOW	10^6 lb/hr	100	55	98	52
CORE DIFF. PRES.	PSI	16	4.0	15	4.0
% PUMP SPEED A	%	86	0	84	81
B	%	86	91	84	0
LOOP DRIVE FLOW A	10^3 GPM	41	0	39	40
B	10^3 GPM	38	40	37	3
LOOP TOTAL J.P. FLOW A	10^6 lb/hr	51.5	20	50	72
B	10^6 lb/hr	48	76	46	20
INDIVIDUAL JET PUMP FLOW # 1	10^6 lb/hr	4.5	6	4.2	2.0
# 6	10^6 lb/hr	4.4	6	4.2	2.0
#11	10^6 lb/hr	4.7	1.6	4.2	5.6
#16	10^6 lb/hr	4.9	2.0	4.6	6.6
TOTAL STEAM FLOW	10^6 lb/hr	14.0	9.2	14.2	9.6
FEEDWATER FLOW	10^6 lb/hr	13.8	9.6	13.0	9.0
REACTOR PRESSURE	PSI	1013	990	1020	1000
REC. LOOP WATER TEMP.	°F	525/523	516/512	[illegible]/520	512/509