TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401

400 Chestnut Street Tower II

February 12, 1982

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Director of Nuclear Reactor Regulation
Attention: Ms. E. Adensam, Chief
Licensing Branch No. 4
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Ms. Adensam:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

By your letter to H. G. Parris dated January 22, 1982, TVA was requested to provide additional information on Watts Bar Nuclear Plant. Enclosed are TVA's responses to questions Q212.123 - Q212.132 concerning reactor systems. This information should close out items 24, 71, 72, 73, 74, 75, and 76 designated as open items in the draft Safety Evaluation Report.

If you have any questions concerning this matter, please get in touch with D. P. Ormsby at FTS 858-2682.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Sworn to and subscribed before me
this 12th day of Feb, 1982
Bryant M. Lowery
Notary Public
My Commission Expires 4/4/82

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

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212.123 Question:
(212.107)
(5.4.7)

The response to Q212.107 dated 9/25/81 references Sequoyah natural circulation startup tests to address the RSB 5-1 natural circulation test requirement. These tests were not performed under conditions satisfying all natural circulation test requirements. Reference and justify other tests to meet this requirement.

Response:

In response to Question 212.93, TVA provided a detailed comparison of the Watts Bar and Sequoyah natural circulation. Since this comparison indicates that Sequoyah is phototypical of Watts Bar, the basis for applicability of Diablo Canyon tests to Sequoyah is also applicable to Watts Bar.

Diablo Canyon is similar to Watts Bar and tests conducted at Diablo Canyon would demonstrate the ability to cooldown to RHR initiation on natural circulation. TVA will address any unforseen safety concerns identified during the Diablo Canyon tests.

212.124 Question:

The analyses of a locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in the FSAR assumes the availability of offsite power throughout the event. In accordance with Standard Review Plan 15.3.3 and GDC 17, we require that this event be analyzed assuming turbine trip and coincident loss of offsite power to the undamaged pumps.

Appropriate delay times may be assumed for loss of offsite power if suitably justified.

Steam generator tube leakage should be assumed at the rates specified in the technical specifications.

The event should also be analyzed assuming the worst single failure of a safety system active component. Maximum technical specification primary system activity and steam generator tube leakage should be assumed. The analyses should demonstrate that offsite doses are less than the 10 CFR 100 guidelines values.

Response:

Accident Scenario

The locked rotor followed by a loss of offsite power transients is postulated to occur in the following manner:

1. Reactor coolant pump rotor locks (or shears) and flow in that loop begins to coast down.

2. The reactor is tripped on low RCS flow in one loop.

3. Turbine/generator trips.

4. Generator trip causes power grid unstability.

5. Grid failure causes three remaining reactor coolant pumps to coast down.

Method of Analysis

The method of analysis used is the same as the cases presented in Section 15.4.4. The following case is analyzed:

Four loops operating, one rotor locks followed by coast-down of other three reactor coolant pumps.

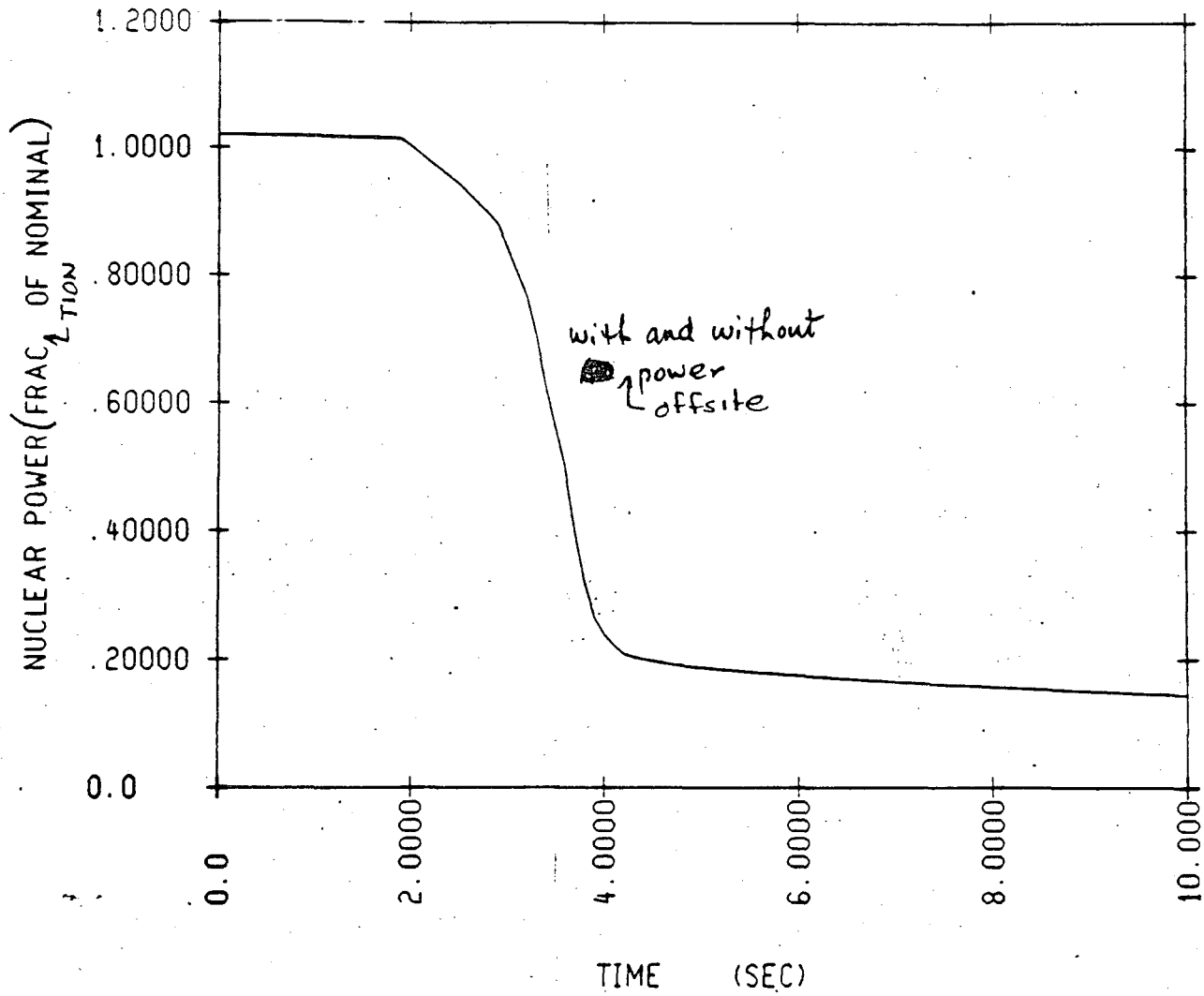

Figure 212.124-1 Nuclear Power Transient, Locked Rotor With and Without Offsite Power

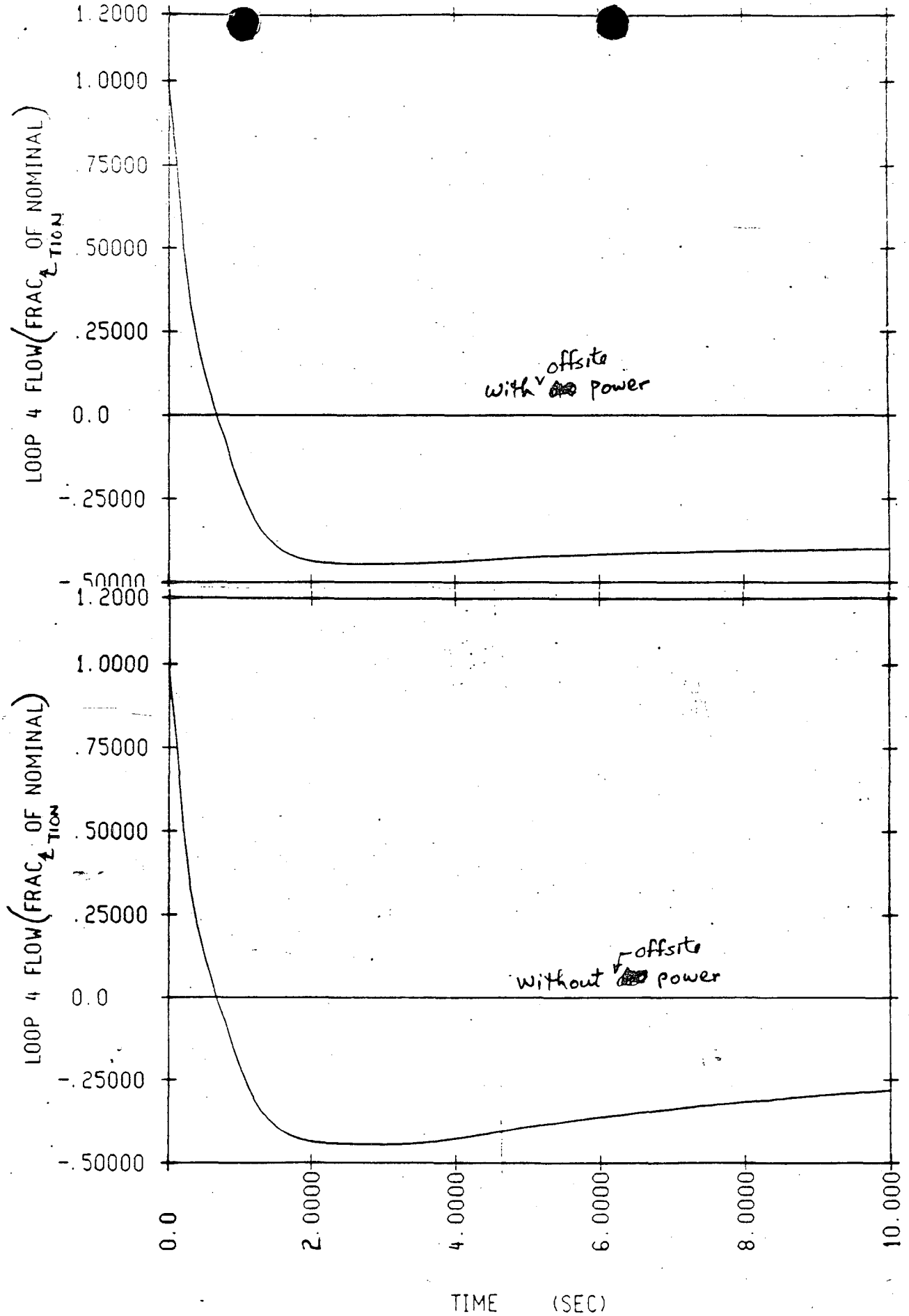

Figure Q212.124-2 Faulted Loop Flow Transient, Locked Rotor With and Without Offsite Power

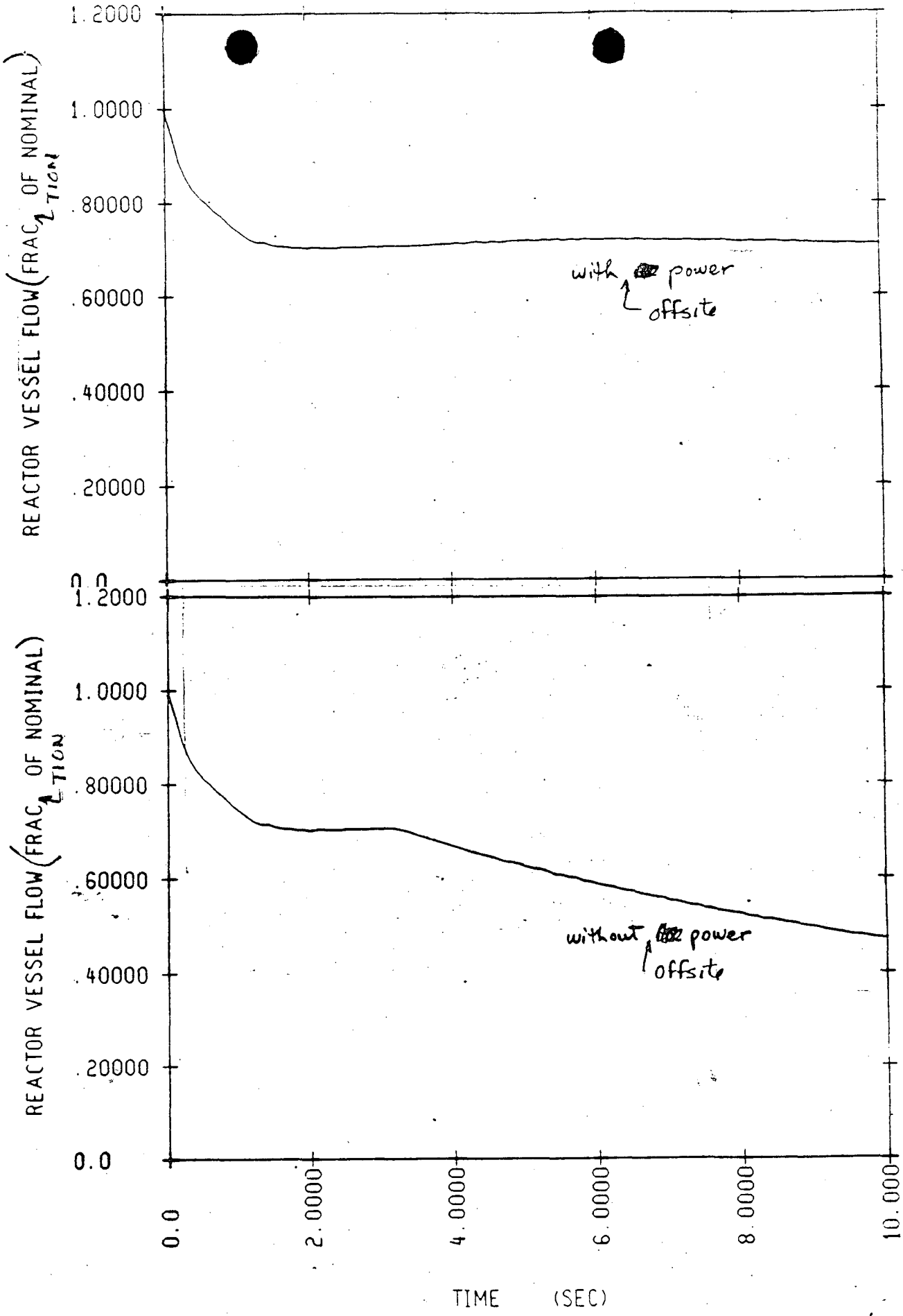

Figure Q212.124-3 Reactor Vessel Flow Transient, Locked Rotor With and Without Offsite Power

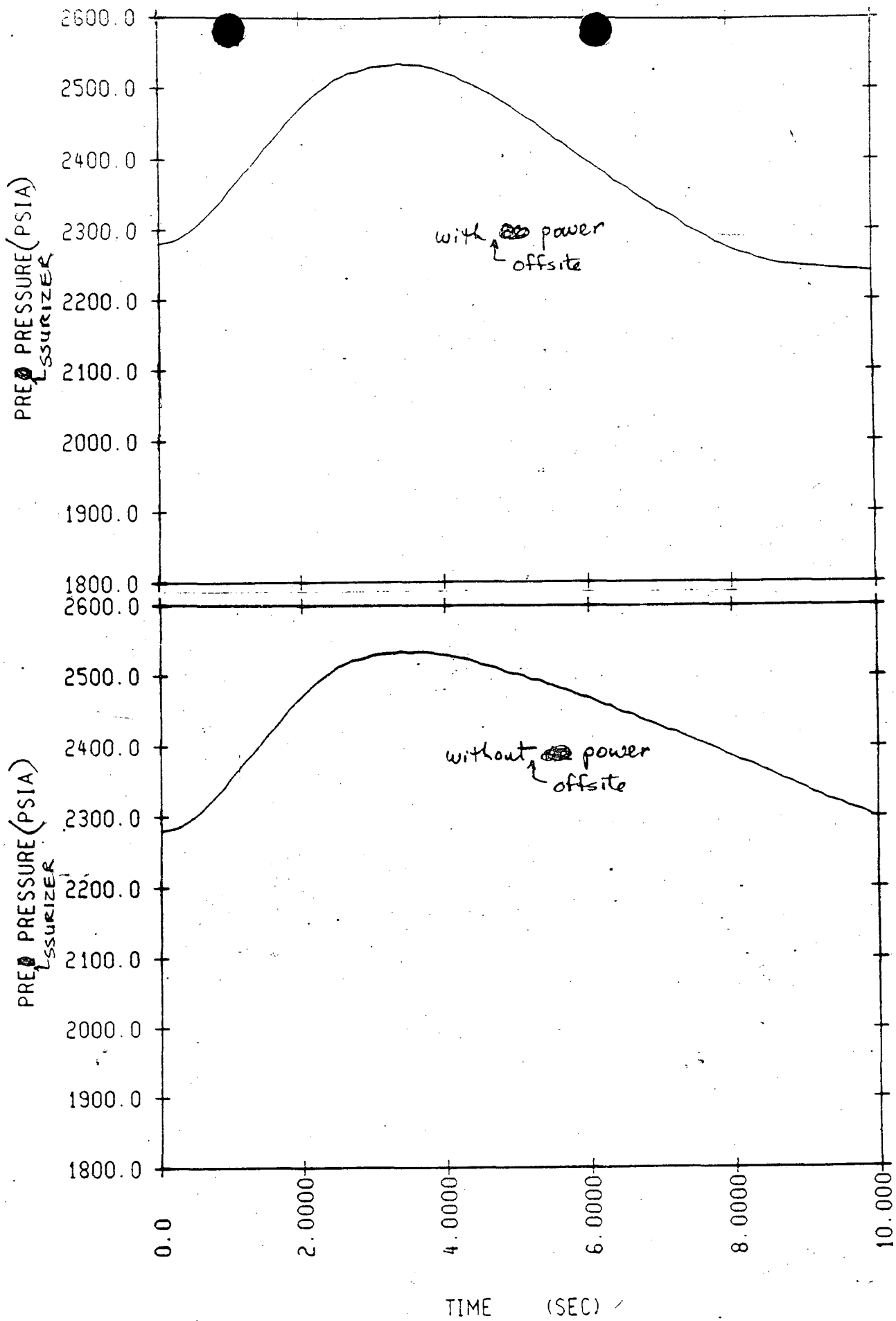

Figure Q212.124-4 Pressurizer Pressure Transient, Locked Rotor With and Without Offsite Power

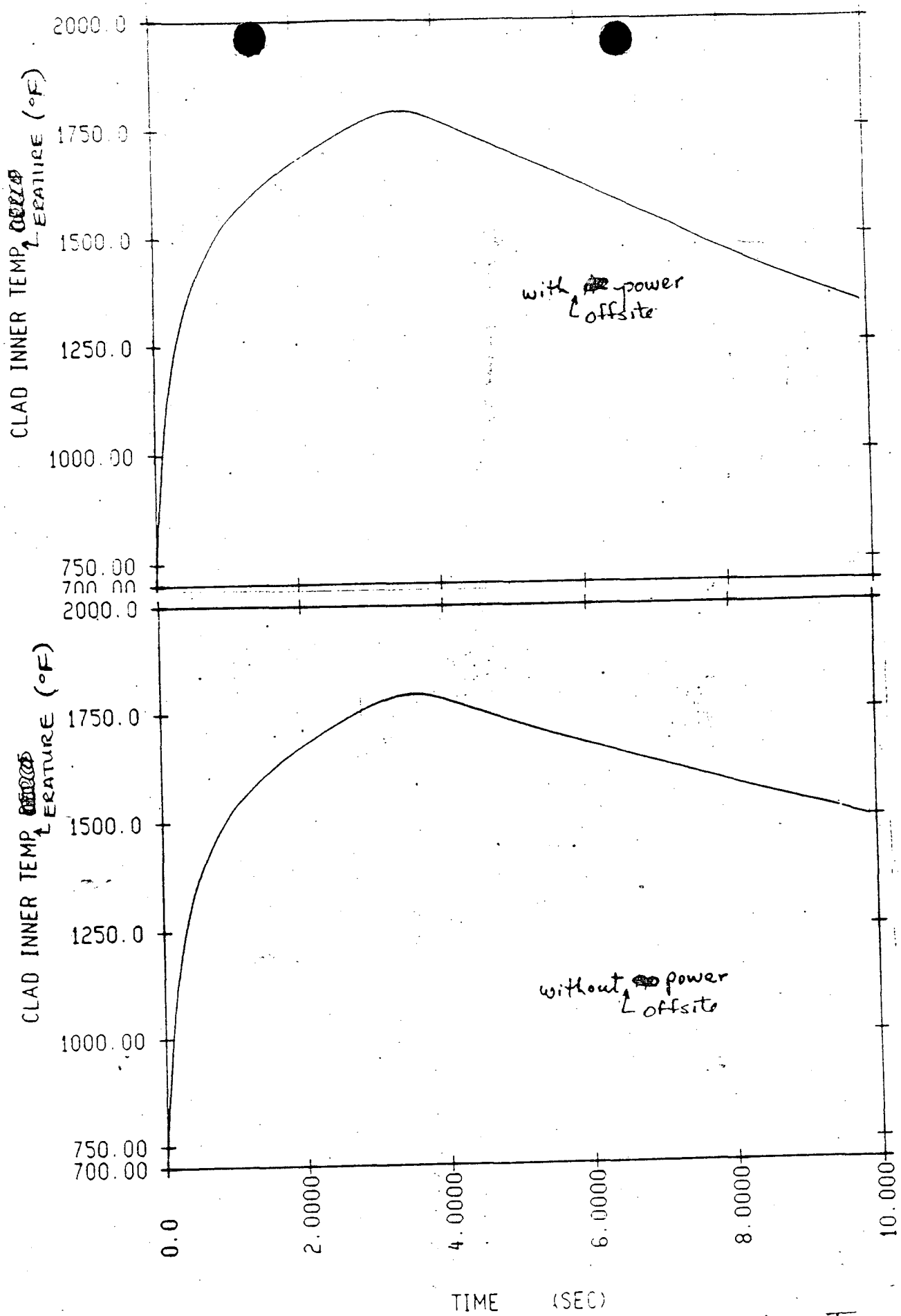

Figure 212.124-5 Clad Inner Temperature Transient, Locked Rotor With and Without Offsite Power

Results

Figures 212.124-1 through 212.124-5 show a comparison between the locked rotor transient without offsite power and the locked rotor transient with offsite power. As can be seen from the figures, loosing offsite power results in the same peak clad temperature and the same peak RCS pressure.

The calculated sequence of events for the case without offsite power is shown in Table 212.124-1.

Conclusion

The locked rotor without offsite power transient is no more limiting than the case presented in Section 15.4.4.

This analysis assumes a two-second delay between reactor trip and loss of AC power which results in no new close analysis requirements to meet 10 CFR 100 guidelines. Justification for allowing use of the two-second delay time is provided in FSAR Section 8.2 which shows grid stability for an extended period following a trip due to mechanical fault.

TABLE 212.124-1

SEQUENCE OF EVENTS LOCKED
ROTOR WITHOUT OFFSITE POWER

Event	Time (seconds)
Rotor on one pump locks	0.0
Low RCS flow trip setpoint reached	.06
Rods begin to drop	1.06
Maximum RCS pressure occurs	3.4
Maximum clad temperature occurs	3.61
Remaining reactor coolant pumps begin to coastdown	3.06

212.125 Question:

The NPSH analyses provided on 10/28/81 do not sum properly. Correct the apparent arithmetic error and provide a corrected NPSH.

Response:

TVA rechecked the addition of the calculated NPSH. No arithmetic error was identified.

212.126 Question:

NPSH, RWST sizing, and Watts Bar sump tests are based on the post-LOCA containment flooding level assumption. During the Sequoyah review, it was discovered that crane wall penetrations were not properly sealed to allow flooding to the assumed elevation and modifications were necessary to provide a proper seal. Verify that Watts Bar penetrations are properly sealed.

Response:

All penetrations in the crane wall at Watts Bar below elevation 716.0 will be sealed using installation procedures, inspection procedures, and sealing materials identical to those utilized at the Sequoyah Nuclear Plant. (See response to SQN FSAR Q6.61 for information on the sealing materials and the Sequoyah Safety Evaluation Report Supplement No. 1 for NRC discussion of the sealing methods.)

212.106 Question
(9.2.7)
(212.37)
(212.87)

An analysis of NPSH for RHR pumps in the recirculation mode of ECCS operation, dated April 1979 and the analysis summary presented in FSAR Section 9.2.7.1 (Amendment 35) are inconsistent in results and calculational assumptions. The 4/79 analysis employed techniques subsequently approved for Sequoyah; however, inputs for the Sequoyah analyses had to be corrected in order to make them acceptable and the resultant calculated NPSH margin reduced from 31.3 ft. to 2.8 ft. Explain the discrepancy between Watts Bar documentations (4/79 Analyses, and Amendment 35) and, in tabular form, list input assumptions (LPI flow, worst case, etc.), criteria (NPSH required at given flow, test rate 'as installed' LPI must meet, etc.), and summary of calculated NPSH available (specify term by term).

Response

The basic discrepancy between the April 1979 analysis and FSAR Amendment 35 is the assumption of maximum water temperature. An earlier analysis used the 160°F maximum of the April 1979 analysis, but the FSAR assumption was provided extra conservatism by using 190°F. This change effectively reduced the RHR pump excess head by 11-1/2 feet from vapor pressure and atmospheric pressure effects (33.8 feet in April 1979 and 23 feet in FSAR).

The pipe and screen loss calculations have been corrected for individual pump rated flows of 4000 gpm for containment spray and 5500 gpm for RHR.

Tabulation of Assumptions, Criteria, and Operating Data

	4/79 Analysis	FSAR Amend. 35	5500 GPM Analysis
Maximum water temperature - assumed	160°	190°	190°
Containment atmosphere - assumed	14.7 lb/in²a	14.7 lb/in²a	14.7lb/in¹
Containment flood elevation - assumed, RWST and ice melt only	716 ft.	716 ft.	716 ft.
RHR pump rated flow - criteria, 2 pumps operating concurrent	4500 gpm ea.	4500 gpm each	550 gpm each
Containment spray pumps rated flow - criteria, 2 pumps operating concurrent	9500 gpm total	9500 gpm total	8000 gpm total
Pipe configuration - data, identical	- maximum loss path -		-
RHR pump NPSHR - data, manufacturer's test curve	17.5 ft.	17.0 ft.	24.5 ft.

Summary of Calculated NPSH (+ Excess Head)

	4/79 Analysis	FSAR Amend. 35	5500 GPM Analysis
Static head 716' - 678' - 7-1/2'	+37'-4-1/2'	+37'-4-1/2'	+37' - 4-1/2'
Containment atmosphere	+34'-8-1/4'	+35	+35
Pipe loss	-8'-4-5/8'	-8'-4'	-10' - 3'
Sump and screen loss	-1'-1-7/8'	-0'-11-1/4'	-0'-11-3/4'
NPSHR	-17'-6'	-17'	-24' - 6'
Vapor pressure	-11'-2-1/2'	-22'-3-3/8'	-22' - 3-3/8'
Excess head resultant	33'-9-3/4' 33.8' shown	23'-9-7/8' 23' shown	14' - 4-3/8'

212.127 Question:
(212.111)
(6.3)

NPSH calculations for Watts Bar assume an LPI flow rate of 4500 gpm per LPI pump. Sequoyah NPSH analyses assumed 5500 gpm, and in the review of Salem, system modifications were necessary to meet NPSH analysis assumptions. Justify that the as-built Watts Bar LPI pump flows will not exceed 4500 gpm (this value must be verified in preoperational tests).

Response:

The results of the NPSH anlaysis assumping 5500 gpm was provided in a revised response to Question 212.106 (letter from L. M. Mills to E. Adensam dated December 10, 1981). The pump suction pressure will be verified in preoperational test W 3.1. Attached is an additional revision to Q212.106.

212.128 (212.115) Question:

The response (8/26/81) to our request for LOCA analyses did not provide or cite sensitivity studies to justify selection of the worst case.

Response:

The analyses performed for the Floating Nuclear Plant (FNP) and the McGuire UHI plants provide a set of cases that constitute an extensive background for consideration of break spectrum results. The general trends indicated for both these plants are consistent. The break spectrum results obtained for the FNP and McGuire plants support the following observations:

1. A break in the cold leg is clearly the most limiting break location.

2. For cold leg breaks, large breaks are much more limiting than small breaks.

3. For large cold leg breaks, guillotine type breaks are more limiting than split type breaks.

The results obtained for FNP and McGuire exhibit consistent trends, Since the FNP, McGuire, and Watts Bar plants are all generally similar, the above conclusions are also applicable to the Watts Bar plant. Thus a limited break spectrum need only be considered for Watts Bar to ensure that the worst case break has been identified.

212.129 (212.116) (6.3) Question:

The applicant should discuss housekeeping procedures to ensure that the containment sump will be maintained in an as-licensed state of cleanliness and free of loose debris.

Response:

The containment sump cleanliness surveillance requirements are addressed in technical specification Sections 4.5.2.c.1, 4.5.2.c.2, 4.5.2.d.2, 4.6.5.7.6, and 4.6.1.7.

212.130 Question:
(212.116)
(6.3)

The applicant should identify how emergency procedures address the possibility of ECCS degradation because of debris clogging the ECCS sump. He should also discuss Watts Bar design flexibilities which, though not credited in Chapter 15, might be available to restore long-term core cooling in the event of sump clogging.

Response:

TVA's response to this question was provided as a revised response to Question 212.116 (letter from L. M. Mills to E. Adensam dated December 10, 1981).

212.131 (212.119) Question:

The response to Q212.119 (boron dilution event) provided a shutdown alarm setting time at 30 minutes after plant shutdown. This schedule is inadequate since it does not provide a schedule for resetting the alarm count setting to stay within an acceptable margin over background. Provide a resetting schedule for the alarm.

Response:

In addition to providing a shutdown alarm setting at 30 minutes following reactor shutdown, when in hot standby, hot shutdown, and subsequent cold shutdown condition, and once below the P-6 interlock setpoint, the High Flux At Shutdown alarm setpoint will be maintained at one-half decade or less above the source range count rate by readjusting its setpoint at least once every 2 hours during the first 8 hours following shutdown and at least once per shift afterwards (as necessary).

212.132 Question:

Your response to Question 212.118 (VCT level instrumentation malfunction) is not sufficient to specifically justify the Watts Bar design for the scenerio postulated in Question 212.118. Justify the Watts Bar design against the Question 212.118 scenerio by providing a chronological timetable indicating initiation of the event, indications, and alarms (credit given for operator corrective action only upon receipt of a control room alarm) with the times at which they occur, the time available to take operator action (after alarm) before the charging pumps are damaged, and a subsequent scenerio analysis if an acceptable operator action time to avert pump damage cannot be justified. In the analysis, justify the number of charging pumps assumed to be operating. Discuss the auxiliary charging system and whether it could be used as a backup to assure safety for this scenerio.

Response:

The only transmitter that will cause the volume and control tank (VCT) to loose actual level is LT-62-130A. This would modulate divert valve (FCV-62-118) to full divert and would prevent automatic makeup. However, the MCR level indicator would indicate actual VCT level. Whenever the level transmitter failed, the control room Abnormal Hi/Low VCT level alarm would come in. The operator would have 11 minutes to identify the situation and take corrective action. During operation, one charging pump is operating. Since charging is 87 gpm and all three charging pumps are rated above 100 gpm, more than one pump is not used during normal operation.