400 Chestnut Street Tower II

May 20, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. L. S. Rubenstein, Acting Chief
Light Water Reactors Branch No. 4
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Rubenstein:

In the Matter of the Application of) Docket No. 50-327
Tennessee Valley Authority)

Reference: Letter from A. Schwencer to H. G. Parris dated May 5, 1980

Enclosure 1 of the referenced letter requested additional information on the special test program and procedures for Sequoyah Nuclear Plant. Enclosed for your review are 10 copies of the TVA response.

The additional information requested in Enclosure 2 will be supplied on or before May 21, 1980. The additional information requested in Enclosure 3 will be supplied by May 29, 1980.

If you have any questions, please get in touch with D. L. Lambert at FTS 854-2581.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

L. M. Mills, Manager
Nuclear Regulation and Safety

Enclosure (10)

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REQUEST FOR ADDITIONAL INFORMATION

EVALUATION OF SPECIAL TEST PROGRAM

Question 1

Pre-test predictions have been made to guide the operating crews in the performance of the tests. For example, see Table 1 attached to each procedure. This guidance should be supplemented by providing the following information to the operating crew and the NRC staff.

1. The predicted variation of hot leg temperature, core delta T and flow rate versus time. A single transient calculation, initiating at perhaps 3% power, using FSAR analyses should be sufficient for tests 1, 2, 5, and 7.

2. An estimate of the rate of pressure decrease with time for test 3 in which the pressurizer heaters are turned off.

3. Bounding analyses for test 6 to determine the maximum rate of cooldown and the maximum rate of heatup that may be expected.

4. The predictions in Table 1 of the procedure for test 4 (and the other tests) indicate that analyses were performed for operation with various numbers of isolated loops. Provide the assumptions and results of these analyses regarding the variation of RCS and steam generator shell side temperatures versus time in the isolated and non-isolated loops. Also provide your predictions of the direction of primary side flow in the isolation steam generator(s).

5. In test 8 you specify a maximum rate of power increase of .75%/min. What is the minimum rate of power increase needed for a meaningful test? For the maximum rate of power increase, provide an analysis which shows the variation of T_{hot} with time and shows the 65°F loop delta T criteria will not be exceeded.

Response

In the questions provided, the assumption was made that pretest predictions have been performed for each of the special tests (referencing Table 1 in the special test procedures). This is not the case, however, since bounding analyses were performed for the tests as described in the Special Test Safety Evaluation Report.

The operating crews have performed the tests on the Sequoyah simulator and have reviewed the procedures to ensure they contain adequate guidance.

1. The analysis described (i.e., the transient calculation initiating at 3 percent power using FSAR analyses) has not been performed; however, engineering judgement has been used to predict the approximate trends in the specified parameters.

Since depressurization is gradual, we do not expect significant variations with time for hot leg temperature, core delta T, or flow rate during test 5. The variations of these parameters with time in tests 1, 2, and 7 are expected to be essentially the same. Test 1 will be performed first. During the performance of test 1, the operations personnel will become familiar with the transient response prior to performance in tests 2 and 7. To aid the operator in performing test 1, general guidelines or trends are given to indicate to the operator what he should expect during the transient.

2. The rate of pressure decrease is estimated to be 20 psi/hour. This value is determined using full temperature pressurizer heat losses and treating the pressurizer as a closed system (no insurges or outsurges).

3. The estimated heat loss from the RCS using full power temperatures is less than 2.6 megawatts (including maximum letdown). The approximate RCS liquid mass is 555,000 lbm. Using a liquid specific heat of 1.2 BTU/lbm^{o}F yields a cooldown rate of roughly 13°F/hr if no pumps were running. This does not include the latent heat of the metal and the pressurizer vapor and therefore should be a conservative estimate of the cooldown rate. The maximum heatup rate, if minimal thermal losses occur and letdown is secured, should be less than 15°/hr.

4. Table 1 in the test procedure is an estimate only. It was derived using the following approximations:

   a. Core flow driving force is proportional to the core delta T.

   b. Resistance to core flow is proportional to the loop flow squared.

   c. The core delta T is proportional to reactor power and inversely proportional to the core flow.

   d. The core flow is approximately equal to the number of operating loops (non-isolated) times the flow in each loop.

   These approximations were used to extrapolate in a hand calculation, the core flow and delta temperature over several power levels and isolation conditions. Table 1 contains estimated values only (as indicated in the heading) and is intended only for use as an aid to the test director and operating personnel.

   We did not feel that it would be of significant benefit to make pretest predictions of all the various configurations shown in Table 1, We did, however, make some engineering evaluations of the expected trends of the parameters mentioned. We expect the steam generator shell side temperatures to remain constant in the unisolated loops and to approach T_{hot} in the isolated ones. The direction of primary side flow should remain positive in the isolated loop, at least until the steam generator reaches T_{hot}.

5. The objective of tests is to verify that natural circulation can be established from stagnant conditions using reactor power to simulate decay heat conditions. This objective is independent of the rate of power increase; therefore, there is no minimum rate of power increase stipulated for a meaningful test.

This maximum rate of power increase was stipulated to prevent the reactor power from significantly leading the flow.

The bases for the 65°F loop delta T criterion is that this is the full power delta T, which means that T_{hot} should not exceed the normal full power T_{hot}. We have no absolute assurance that the core delta T will not reach the 65°F value, however, if it does, the reactor will be tripped manually with no expected damage to the clad. This transient is bounded by the analysis presented in the special test safety evaluation section 4.2.

Question 2

Since system pressures may vary appreciably during the tests, the core exit and average temperature limits of Section 2.1.1 are not appropriate throughout the test. The margin to saturation appears to be the key limiting parameter. It appears that errors in temperature and pressure readings under low flow conditions could be such that a subcooling of less than 10°F would not be indicated even if an actual subcooling of 0°F occurs. With respect to core exit thermocouples, the following is noted: (a) thermocouples are located within sampling chambers in the UHI support columns, (b) with all pumps operating, flow inside the support columns and above the thermocouple is from the upper head region (T close to cold leg T) for outer locations and in upward direction for central region; (c) flow directions and temperatures for natural circulation conditions have not been tested. Provide a tabulation and discussion of potential inaccuracies and uncertainties in the core exit thermocouple and pressure readings that affect the subcooling margin.

Response

The 10°F margin to saturation temperature as the point for SI actuation is based on the following:

1. Pressurizer Pressure - Nominal steady-state errors of 3% span which corresponds to an error in pressure of 24 psi. This results in an error of approximately 2°F in the pressure range of 1700 to 2200 psig.

2. RCS Wide Range Pressure - Nominal steady-state errors of 3% span which corresponds to an error in pressure of 90 psi. This results in an error of 7-10°F in the pressure range of 1400 to 2200 psig.

3. Thermocouples - Nominal steady-state errors of 2% per point. Using a temperature of 600°F, the steady-state errors in temperature would be 12°F.

Using random readings from thermocouples in conjunction with a random Pressurizer Pressure channel, the indicated temperature will be within ±12.2°F of the actual temperature during 95% of normal operation. Use of the highest thermocouple and lowest pressurizer pressure channel results in even closer agreement with the actual saturation temperature. This is felt to be adequate to warrant the use of an indicated margin to saturation of 10°F for SI initiation. It should be noted that the test procedures restrict operation to no less than 20°F indicated margin and test termination with reactor trip at an indicated 15°F margin to saturation. This would eliminate the heat source reducing the possibility that SI will be required.

Question 3

Although the hot and cold leg pipe Reynolds numbers indicate turbulent flow under natural circulation conditions, flow stratification can exist. Provide information on the immersion depth, size, number and location of T_{hot} RTD's for hot and cold leg temperature measured during the test program. Discuss the capability of this instrumentation to give representative values for hot and cold leg temperatures and provide an estimate of the potential inaccuracies and uncertainties in the readings.

Response

There are four Wide Range T_H and four Wide Range T_C RTD's (one T_H and one T_C per loop) in the RCS. The T_H RTD's are in thermowells in the hot legs and the T_C RTD's are located in thermowells in the cold legs. Nominal steady-state errors of 4% span, with an allowance of 3% for flow/temperature stratification, were assumed. This corresponds to a possible error in temperature of 28°F. Efforts are made to cross check the Wide Range RTD's with the Narrow Range RTD's when RCP's are operating and in the Narrow Range span to minimize the steady-state errors.

Question 4

Using Tables 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," and 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Setpoints," of the Technical Specifications, provide, for each of the proposed tests, a matrix identifying each functional unit that will be bypassed using the permissive interlocks. For functional units otherwise defeated, identify the trip and describe the method, time, and manpower requirements for restoring its operability. Verify that the analyses and results presented in the "Evaluation of Special Test Program" did not take credit for any functions that were bypassed or defeated using the interlocks or by other means.

Response

Table 1 notes the operability of the various protection channels for the special tests. All channels that are noted as blocked by permissive or defeated were assumed to be inoperable when the safety evaluation was performed, thus no credit was taken for their operation. No assumptions were made as to time or manpower required to restore the operability of the defeated channels, but rather an overall assessment of the test period relative to plant life was made. It should be noted that while automatic actuation of SI is blocked, reactor trip and the alarms from an SI channel will be operable with the operator utilizing them as necessary.

Question 5

Section 2.1.3 refers to a positive moderator temperature coefficient that was considered in the analyses. Provide the following information: predicted coefficient as a function of temperature; a description of measurements of the moderator temperature coefficient to be made before the special test program starts to verify the prediction, including the temperature range the measurements will cover.

Response

The moderator temperature coefficient (MTC) will be verified to be within FSAR analysis assumptions as a part of the zero power physics test program. This verification will be made at approximately 547°F. The tests will be performed such that the MTC will be approximately 0 pcm/°F. At various temperatures the MTC may be slightly positive (at low temperatures) or slightly negative (near T_{no} load), but efforts will be made to keep the value of approximately 0 pcm/°F. The expected MTC under all test conditions is less than the value of +2.7 pcm/°F used in the Safety Evaluation.

Question 6

From the discussions during the meeting on 4/23/80, it was understood that the auxiliary spray would be used for pressure control for some tests. If either of the two valves in the charging lines, which are in parallel with the auxiliary pressurizer spray line, are open, there may be no auxiliary spray flow. If the valves in the charging lines are closed to permit pressure control with the auxiliary pressurizer spray, how is pressurizer level controlled if normal letdown is used?

Response

Since both the normal charging path and the auxiliary spray line go into the reactor coolant system, there is no effect on pressurizer level as long as charging and letdown balance. If auxiliary spray is needed, the normal charging path will be isolated for as long as necessary to control pressure and then reopened. While using auxiliary spray, the flow to the spray nozzle can be adjusted by opening the normal spray valves and allowing some of the flow to bypass the spray nozzle and directly enter the RCS cold legs. In the way the normal charging flow rate can be maintained yet not all will be directed to the spray nozzle. Using this method, charging and letdown will remain balanced and no pressurizer level problems are foreseen.

Question 7

Explain item (5) in Section 2.1.4 of the Safety Evaluation.

Response

The "at power" reactor trips are those blocked by permissive P-7 when less than 10% RTP. These trips are: loss of flow in two or more loops, under-voltage - RCP, underfrequency - RCP, pressurizer pressure - low, pressurizer water level - high, and reactor trip from turbine trip.

Question 8

You proposed to block the automatic actuation of safety injection during the performance of the low power test program. Provide a discussion of the safety considerations (both the benefits and possible adverse impacts) of the approach you propose relative to changing setpoints during the test program.

Response

The safety evaluation was performed based on operator initiation of Safety Injection when required. In those instances where SI is necessary, there is sufficient time for the operator to respond. There is sufficient reactivity bank worth with the control rods above the insertion limits such that SI is not necessary to provide shutdown margin after reactor trip. The main function of SI for the special tests is to provide RCS inventory, for which it has been determined that adequate time is available for operator response. Westinghouse has determined that at the low flow rates experienced during the special tests it is advisable to reduce the possibility of a spurious SI because of temperature stresses to the primary system. The special tests will be performed in an operation area where spurious SI may be likely. In addition, there are two tests where the pressurizer pressure is expected to decrease below the lower limit of the instrument span (1700 psig). To prevent these unnecessary thermal transients, it is recommended that the automatic actuation of SI be blocked.

Question 9

For tests when pressurizer heaters are turned off or rendered inoperable by loss of power, it is stated in Section 2.1.7 that pressurizer pressure will be controlled by changing pressurizer level. The recent North Anna incident showed that the pressurizer pressure is very sensitive to level under these conditions. How has this effect been included in the pre-test analysis and test procedures?

Response

The conditions in the special tests are not the same as those existing at North Anna. The problems that North Anna experienced were caused by:

1. Subcooled liquid in the pressurizer causing sensitivity to level reduction because of the loss of water-to-steam flashing as pressure was reduced. This sensitivity continued until the pressurizer heaters heated the water back to saturation temperature.

2. A safety injection causing isolation of all letdown and a rapid addition to the RCS water inventory. This resulted in the pressure rising to the PORV setpoint and repeated cycling of the PORV to compensate for the inventory addition.

After the SI was terminated, normal letdown and charging reestablished, and the pressurizer water temperature heated back to saturation condition, their sensitivity problems did not exist. In our natural circulation tests, we will not have the pressurizer conditions that North Anna had and therefore, we do not anticipate pressure control problems.

Question 10

You propose to lock out the UHI system during the performance of the low power test program. Provide a discussion of the safety considerations (both the benefits and possible adverse impacts) of the approach you propose. Clearly list the tests and postulated accidents that may cause the system pressure to decrease sufficiently to result in UHI injection if the system were not locked out.

Response

The transients which reduce the RCS pressure sufficiently to actuate UHI have been reviewed (Table 2) to determine the effects of locking out UHI during the low power test program. The only transients requiring UHI for mitigation are the loss of coolant accidents. These two transients are discussed separately in the response to questions 13 and 15. Even under normal full power operation, UHI is not required to reduce the consequence of a steam line break. A number of the remaining transients become less likely due to the methods of operation which will be employed during the low power test operation. With the pumps tripped, the opening of a pressurizer spray valve becomes inconsequential. During tests, the power operated relief valves will be blocked. Thus, further depressurization due to a controlled malfunction or inadvertant opening becomes highly unlikely.

The special low power test program requires a large amount of operator action for manual pressure control. Using, in most cases, auxiliary spray and steam dump. It could then be expected that the probability of UHI being actuated due to one of these operations being carried to an extreme would be increased. The activation of UHI with the pumps tripped may require removal of the upper head for inspection due to thermal stress effects in the guide tubes. Since the addition of UHI is not increasing the margin of safety under these test conditions, it is desirable to lock out the UHI system during the performance of the low power tests.

Question 11

For each instrument that provides an automatic reactor trip, engineered safety feature actuation, or visual indication of a parameter monitored (Section 3.0, Operational Safety Criteria) for manual reactor trip or safety injection, submit an evaluation of the methods used to compensate for this during operation outside the Technical Specification limits. For example, T_{ave} is as low as 425°F (Section 2.1.4). Are the temperature readings accurate? Are the measurements of other parameters (e.g., steam generator level) affected?

Response

The instruments that are affected by reduced temperature operation are.

1. Neutron Flux - Both power and intermediate ranges, due to increased shielding from colder water in the downcomer. Test 9A will determine the errors introduced from this effect. As a base assumption, 1% error in the power level per °F in downcomer temperature was assumed.

2. Water level - Pressurizer water level and steam generator water level indications will be impacted. Under extreme conditions, steam generator level may be expected to indicate about 2% below actual level, below 18% level, and about 5% above actual level at 80% level. In addition, pressurizer level may be expected to indicate about 7% below actual level below 35%, and about 10% above actual level at 80% level. These errors are in the conservative direction in all cases.

3. RCS flow - Since this trip function is blocked and operation will be at very low flow rates, this parameter was not considered in the evaluation.

Question 12

Section 4.1.1.1 indicates that at least one RCS pump will be started following every reactor trip. Explain the basis of this action in light of the following considerations: effects on heat transfer, additional heat source from pump heat, boron mixing, and system pressure control. Specify that all four pumps should be operating prior to rod withdrawal from a subcritical position.

Response

The object of starting at least one RCP following each reactor trip is to ensure positive coolant flow through the core during subsequent RCCA withdrawal maneuvers which will be required to return the core to a critical condition and to operating power levels for the tests. Without this precaution, each occurrence of reactor trip, whether inadvertent, deliberate, or required for protection, would be followed by a startup transient similar to that proposed in test B but with less control of initial conditions. In the event of an accident or inadvertent transient during the return to critical and power, Westinghouse design correlations indicate that positive core flow would tend to reduce the probability of DNB relative to the case with no flow. For example, an evaluation has indicated that in the event of uncontrolled rod bank withdrawal at normal RCS temperatures with one RCP operating, DNB would be precluded. In the event that DNB has occurred, positive flow would tend to improve post-DNB heat transfer and thus reduce the resultant clad temperature and degree of clad damage. The heat source from the pump, which is much lower than the test design power levels, would tend to offset system heat losses and at most heat the system at a few degrees per hour. Any non-uniformity in boron concentration existing in the RCS loops would be eliminated by the operation of one RCP before criticality is attained. Operation of the pump on the loop connected to the pressurizer surge line also permits use of normal pressurizer spray, which allows both mixing of pressurizer water with the RCS pressure. If the RCP is started shortly after a trip from normal test conditions, a drop in pressurizer level and pressure will be expected, with level returning to the normal no-load level. Return to normal RCS pressure can be accomplished within a few minutes using pressurizer heaters.

Question 13

Provide a description of the analyses which indicate that the core will remain covered "for at least 6000 seconds" following a small LOCA. (Section 4.1.2.)

Response

The small break loss of coolant transient was evaluated by examining the case of a 1-inch break with no SI but with UHI at 5 percent power. For this case, the core began to uncover at 24,000 seconds. If it is assumed that UHI flow left the system as saturated liquid, then the core would uncover approximately 1800 seconds earlier, or at approximately 22,200 seconds. If it is assumed the UHI flow left the system as a vapor, then the core would uncover approximately 8000 seconds earlier, or at approximately

16,000 seconds. An analysis of the case for full power with no UHI, the core begins to uncover at approximately 8000 seconds. Thus, the advantage of 5 percent power operation is 8000 seconds or a factor of two in uncovery time. A third analysis shows that at full power a 2-inch break with no SI or UHI core uncovered at approximately 4000 seconds. Using the factor of two from the previous case, it would be expected that the 5 percent power case would remain covered for 8000 seconds. The low power-low stored energy operation causes a much faster depressurization than the full power case. This results in activation of the cold leg accumulators sooner for the low power transients. The earlier addition of the cold leg accumulators will provide additional cooling for the low power transients. The 6000 second core uncovery time mentioned in the submittal represents an added conservatism to the 2-inch break 8000 second uncovery time.

Question 14

Low probability alone does not seem sufficient reason to dismiss the single rod withdrawal (Section 4.1.2.3) and rod ejection (Section 4.1.3.6) accidents without analysis. Provide the maximum core heat flux (or fraction of normal heat flux) for this event. Determine whether the conclusion stated in the last sentence of Section 4.2.3.2 is applicable to these events.

Response

For the single rod withdrawal transient, the maximum core average heat flux would be 10 percent of the core average heat flux of full power, assuming reactor trip at 10 percent power and no DNB. If the transient occurs during a period (in test 9A or 9B) following a reduction in reactor downcomer temperature and before compensating corrections to the NIS channel gains or flux trip setpoints have been made, reactor trip could occur at a power higher than 10 percent and thus the core heat flux would be higher. For example, assuming a 17/^{o}F reduction in indicated power and an uncompensated temperature reduction from 557°F to 450°F, the maximum core average heat flux would be 21 percent of normal full power heat flux.

Although no calculations of rod ejection transients have been made for the test conditions, the results for the FSAR case at hot zero power conditions are indicative of the order of magnitude of core heat flux. This calculation assumed all RCP's operating, neglected the effect of local boiling on heat transfer, and used a delayed neutron fraction appreciably below the cycle 1 BOL minimum value. The resultant peak core average heat flux is 25 percent of the core average heat flux at full power.

No analyses have been performed to determine whether the DNB criteria of the FSAR are met for these types of transients.

Question 15

Provide a description of the analyses which indicate that the core will remain covered for 1.7 hours following a large LOCA. (Section 4.1.3.)

Response

The large break loss of coolant accident was evaluated by determining how long it would require to begin to uncover the core if the vessel was

initially full of water up to the bottom of the hot legs. The water to fill the vessel to this level is supplied by the cold leg accumulators. The low power-low burnup condition of the core would allow the refill process to occur much more rapidly than if power operation had been the initial condition. A linear axial void fraction distribution was assumed in the core at the time of uncovery.

Question 16

For the analysis described in Section 4.2.1, provide a figure of core heat flux versus time corresponding to Figure 4.2.1.

Response

See attached Figure 1.

Question 17

The last sentence of Section 4.2.3.2 states that "Analyses of core conditions" indicate that the DNB criterion of the FSAR is met. Describe the data or correlations that were used to determine the critical heat flux for low flow conditions that will exist during the test program. Why doesn't the same conclusion regarding meeting of DNB criteria apply for the RCC bank withdrawal of transients described in Section 4.2.2?

Response

For certain cooldown transients, the conclusion that DNB is precluded was drawn based on use of the W-3 critical heat flux correlation. Although the analyses for the cooldown events discussed in Section 4.2.3.2 result in mass velocity below the range of direct applicability of the correlation, the reactor heat flux was so low relative to the predicted critical heat flux that even a factor of 2 would not result in serious concern for DNB for this event.

For uncontrolled RCCA bank withdrawal transients, analyses have shown that for cases similar to those presented in the FSAR, i.e., continuous insertion rate, a reactor trip will occur on the low setting of the high flux trip. Because of the difficulty of calculating core hydraulic behavior under test conditions, the potential for DNB resulting from inadvertant rod withdrawal has not been precluded by analysis. Once reactor trip is obtained, a rod assumed to be in DNB would have adequate cooling. Figure 4.2.4 of the Safety Evaluation shows the time of reactor trip as a function of insertion rate. This figure represents the maximum amount of time a rod could be in DNB as a result of the rod withdrawal.

As a consequence of limiting the excess reactivity and the fact that many normal trips are bypassed, there is another condition not considered in the FSAR. This condition is that a continuous rod withdrawal will not result in sufficient increase in power (10 percent assuming setpoint errors) to cause a reactor trip. Because of the difficulty of predicting DNB under the test conditions, it has not been absolutely demonstrated that DNB will not result with an ensuing clad temperature excursion until manual reactor trip is initiated. Again, due to analysis limitations, a reasonable low bound on required operator action time has not been demonstrated. Therefore, major reliance is placed on the low probability of inadvertant rod withdrawal events occurring during the limited test duration.

TABLE 1

PROTECTION CHANNEL	TEST	1	2	3	4	5	6	7	8	9A	9B
Power Range, Neutron Flux-High Setpoint	Rx.Tr.										
Power Range, Neutron Flux-Low Setpoint	Rx.Tr.	R	R	R	R	R	R	R	R	R	R
Power Range, Neutron Flux-High Positive Rate	Rx.Tr.										
Power Range, Neutron Flux-High Negative Rate	Rx.Tr.										
Intermediate Range, Neutron Flux	Rx.Tr.	R	R	R	R	R	R	R	R	R	R
Source Range, Neutron Flux	Rx.Tr.	P-6	P-6	P-6	P-6	P-6	P-6	P-6	P-6	P-6	P-6
Overtemperature Delta T	Rx.Tr.	D	D	D	D	D		D	D		D
Overpower Delta T	Rx.Tr.	D	D	D	D	D		D	D		D
Pressurizer Pressure-Low	Rx.Tr.	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7
Pressurizer Pressure-High	Rx.Tr.	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7
Pressurizer Water Level-High	Rx.Tr.										
Loss of Flow	Rx.Tr.	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7
Steam Generator Water Level-Low Low	Rx.Tr.	R	R	R	R	R	R	R	R	R	R
Steam/Feedwater Flow Mismatch	Rx.Tr.	D	D	D	D	D	D	D	D	D	D
Steam Generator Water Level-Low	Rx.Tr.	D	D	D	D	D	D	D	D	D	D
Undervoltage-RCP	Rx.Tr.	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7
Underfrequency-RCP	Rx.Tr.	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7
Turbine Trip	Rx.Tr.	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7	P-7
Containment Pressure-High	SI	D	D	D	D	D	D	D	D	D	D
Pressurizer Pressure-Low	SI	D	D	P-11/D	D	P-11/D	D	D	D	D	D
Differential Pressure Between Two Steamlines-High	SI	D	D	D	D	D	D	D	D	D	D
Steam Flow in Two Steamlines-High	SI	D	D	D	D	D	D	D	D	D	D
T_{avg}-Low Low	SI	D	D	D	P-12/D	D	D	D	P-12/D	P-12/D	P-12/D
Steamline Pressure-Low	SI	D	D	D	D	D	D	D	D	D	D
Containment Pressure-High High	ØB										
Steam Generator Water Level-High High	Fdwtr.Iso.										

R - Setpoint changed to lower value
D - Automatic protection function defeated
P-6 - Reactor Trip blocked by permissive P-6
P-7 - Reactor Trip blocked by permissive P-7
P-11 - Safety Injection blocked by permissive P-11
P-12 - Safety Injection Blocked by permissive P-12

TABLE 2

TRANSIENTS ACTUATING UHI

Transient Case	Design Number of Occurrences
1. Reactor trip with cooldown, including inadvertent initiation of SI	10
a. Cooldown due to excess feedwater	
b. Cooldown due to excess steam dump	
2. Inadvertent RCS depressurization	20
a. Actuation of one pressurizer safety valve	
b. Opening of one pressurizer power relief valve	
c. Opening of one pressurizer spray valve	
d. Controller malfunction causing one power relief valve and two spray valves to open	
e. Inadvertent auxiliary spray operations	
3. Small loss of coolant accident (1-inch break)	5
4. Small steam break	5
5. Large loss of coolant accident	1
6. Large steam break	1

FIGURE 1

RESPONSE TO QUESTION 16

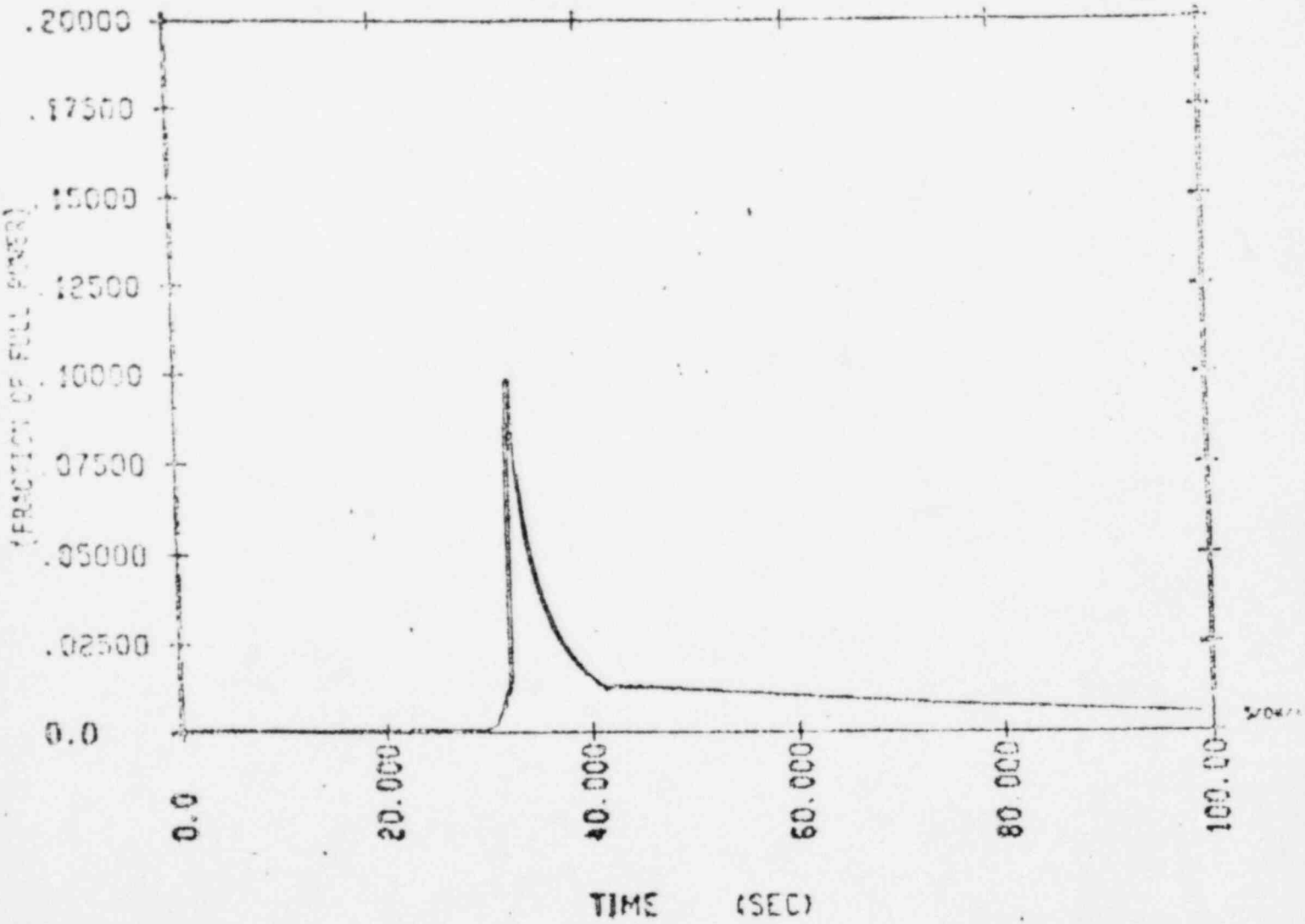

Figure 4.2.2.A Uncontrolled Rod Bank Withdrawal from a Subcritical Condition, Core Average Heat Flux vs Time

REQUEST FOR ADDITIONAL INFORMATION

PROCEDURES FOR LOW POWER TEST PROGRAM

Question

It is not clear what written instructions the operator will use in the event plant conditions go outside the limits listed in the special test procedures that only the test engineer will use. How will the operator know that he should manually trip the reactor or manually initiate safety injection?

Response

Criteria for manual trip and manual initiation of safety injection are specified in each of the special test procedures. The reactor operator will utilize these procedures during performance of the test.

Test 7 - Question A

Explain why the battery charger is not maintained connected to the DC bus as it would be in a real blackout condition. Any loading effect by the charger should be included in the test.

Response

The battery charger is not maintained connected to the DC bus because it was much more convenient to disconnect the normal charger at that point. The charger is designed to have no loading effect on the batteries when not charging. It accomplishes this by utilizing a diode network to prevent any current flow from the batteries.

Question B

Justify the use of temporary lights as mentioned in Step 2.14 NOTE. Unless all areas are blacked out, the test does not validate the assumption that all areas of importance have been identified.

Response

Any temporary lights will only be utilized in areas in which activities deemed important by operations are taking place, but which are not required in a loss of all power situations. At present, no areas have been identified that will need to stay in operation during the test, but the statement is left in the test to add flexibility if that situation should arise. The test was not designed to identify important areas in the plant. It was designed to verify that the important areas that have been recognized as necessary to shut the plant down safely have sufficient lighting and access to fulfill this requirement.

Test 9B - Question

In Step 3.4, identify the direction that the pressure will change so that the operator will know the type of "care" he must take.

Response

The prerequisite simply reminds the test director and operator that pressure control will not be as effective with auxiliary spray as it would be with normal spray.

Question

In Step 5.1.7, 100 ppm increase in boron concentrating should be determined by mass balance rather than by sampling. Sampling may not be valid because of stratification or other lack of uniformity throughout the reactor coolant system.

Response

A mass balance is used in determining the amount of boron to add to the system per the normal operating instructions. The sample simply verifies the mass balance. The sampling method used will show any lack of uniformity in the RCS because samples are taken from two RCS cold legs and the pressurizer liquid.