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FFTF PRIMARY SYSTEM TRANSITION TO NATURAL CIRCULATION FROM LOW REACTOR POWER

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ABSTRACT

Plans for reactor and primary loop natural circulation testing in the Fast Flux Test Facility (FFTF) are summarized. Detailed pretest planning with an emphasis on understanding the implications of process noise and model uncertainties for model verification and test acceptance are discussed for a transition to natural circulation in the reactor core and primary heat transport loops from initial conditions of 5% of rated reactor power and 75% of full flow.

1. INTRODUCTION

The FFTF has been designed to provide shutdown heat removal utilizing natural convective cooling in the event that the normal forced flow cooling capability is not available. [1] A series of tests are planned during the FFTF startup program to confirm plant decay heat rejection capabilities by natural circulation. [2] The primary loop and whole plant testing is focused on satisfactory transition to natural circulation assuming lo of all electrical power to the pumps during operation at full power, end-o.-life burn-up conditions. This paper briefly summarizes this series of planned tests. It describes the instrumentation to be used in these tests and provides detailed test plans and analyses for the first test in this series; a reactor scram to natural circulation from 5% power and 75% full flow. Tests in an FFTF secondary loop of transition to natural circulation from refueling conditions, a second major objective of the natural circulation plant acceptance test program, have been completed and are described in more detail in Reference [3].

2. NATURAL CIRCULATION TEST PLANS

The principal objective of the FFTF natural circulation tests is to confirm adequate FFTF shutdown heat rejection capability utilizing a series of tests which culminate in prototypic transients that require minimal extrapolation to the limiting design cases. To facilitate required extrapolation and to ensure that the relevant processes are understood, analytical models will be sufficiently validated to establish their acceptability for safety evaluation. The progression of tests has been designed to assure that transient fuel cladding temperatures are maintained within normal steady state operating values to minimize the potential for damage to FFTF fuel; protection of plant components has been given similar attention. A secondary objective of the testing is to collect data for further refinement of analytical models used in the prediction of natural circulation performance.

The planned sequence of tests for confirming an acceptable transition to natural circulation from full power, maximum burn-up conditions consist of (1) a reactor scram to natural circulation from 5% power, 75% flow, (2) low power (<5%) steady-state natural circulation tests, and (3) high power scram transients to natural circulation conditions.

The first test (scram from 5% power) is the principal subject of this paper and will be conducted prior to reactor operation at higher power. This test is designed to provide early confirmation of sufficient shutdown heat removal capability in the reactor vessel and the primary heat transport loops to assure adequate decay heat removal during early phases of plant power testing. The detailed test design and planned evaluation is addressed below.

Following the initial ascent to power, a series of natural circulation tests is planned at low fission power (<5%). These tests will measure loop and core performance during steady-state operation utilizing natural convection in the primary and secondary Heat Transport Systems. These steady-state tests are designed to provide key model verification data including loop pressure drop/thermal head correlations, in-core inter- and intra-subassembly flow redistribution information, and unbalanced loop heat rejection data.

A final series of reactor scrams to natural circulation is planned from the following initial power, flow, and decay heat conditions:

- 35% power, 75% flow, 1 hour at power
- 75% power, 75% flow, 1 hour at power, and
- 100% power, 100% flow, 25 hours at power.

The final transient is highly prototypic of the limiting case total loss of pump electrical power design transient, providing a rather direct confirmation of the adequacy of FFTF decay heat removal capability. Other tests in the series will support extrapolation from this test to the maximum burn-up condition.

Final pretest predictions have not been completed for the low power scram, the steady-state, or the high power transient tests. Preliminary predictions were previously reported in Reference [2]. Detailed analyses for the 5% scram are reported later in this paper. Final pretest predictions will be issued approximately one month prior to actual test performance.

3. NATURAL CIRCULATION TEST INSTRUMENTATION

In-core instrumentation for the primary system natural circulation tests includes two Fueled Open Test Assemblies (FOTAs), a Vibration Open Test Assembly (VOTA) and an Absorber Open Test Assembly (AOTA). The core map of Figure 1 shows the location of these assemblies in the core. The FOTAs, one in row 2 and the other in row 6, are prototypic of the FFTF driver fuel assemblies with thermocouple temperature instruments provided in pin wire wrap, on the duct walls and in the outlet stalk which extends above the sub-assembly outlet. Each FOTA has approximately 44 thermocouples distributed throughout the sub-assembly (e.g. see Figure 2). A specially calibrated eddy current flowmeter is located in each FOTA sub-assembly stalk. Several of the FOTA thermocouples will be used to provide a high temperature scram signal during the steady-state natural circulation tests; thereby providing automatic test termination if unexpected cladding temperatures are measured. The AOTA contains absorber pins prototypic of a control rod assembly and is also instrumented with numerous thermocouples and an eddy current outlet flowmeter. The VOTA is a structural assembly which contains thermocouples, an outlet eddy current flowmeter, and self-powered fission chambers for measuring neutron flux levels. The two FOTAs and the VOTA are currently installed in the core. The AOTA will be loaded after the initial ascent to power and will not be available for the 5% scram test.

The in-core thermocouples are expected to have an accuracy of ±1° - 3°C depending on the success of planned averaging and isothermal calibration procedures. The FOTA eddy current flowmeters utilize phase detector electronics and have been calibrated in ex-reactor sodium loops at low flows to an accuracy of ±60 cc/sec. which is less than 26% of the expected fueled sub-assembly flow during the 5% scram test.

In addition to the instrumentation in the OTAs described above, each core position is instrumented with an eddy current flowmeter and thermocouples located in the instrument tree above the sub-assembly outlet. This instrumentation is expected to be of limited value for model calibration due to the slow response time of the instrument tree thermocouples and the possibility of cross flow between sub-assemblies through the gap between the sub-assembly handling sockets and the bottom of the instrument tree. Nonetheless, all of the flowmeters have been calibrated in an ex-reactor sodium test loop at high flows and selected flowmeters (as shown in Fig. 1) have been provided with phase detection electronics and have been calibrated in the natural circulation flow range. The selected assemblies include FOTA "twins" and neighbors to isolate effects attributable to the FOTA stalk or to cross flow.

Reactor power will be measured using installed neutron flux instrumentation (compensated ion chambers) located in the cavity outside of the reactor vessel. These chambers will be calibrated using calorimetric methods with an expected accuracy of (±7.4%) during the initial ascent to power. The linearity of the ex-vessel chambers down to the low powers used in the natural circulation tests will permit transfer of this calibration. The assumption involved can be checked using in-core self-powered fission chambers in the VOTA. Since the reactor thermal power calibration will not be done until after the 5% scram test, the reactor power level uncertainty is expected to be ±20% for this early test (though the subsequent more accurate calibration will be transferable to the test data). The decay heat computation has an uncertainty of ±12% for an overall power uncertainty of ±32%.

The instrumentation in the heat transport loops during the natural circulation tests includes permanently installed Resistance Temperature Detectors (RTDs) in the primary and secondary loops for measurement of loop ΔT. Primary loop flow is measured with the magnetic flowmeter located in the cold leg which has been calibrated using a Pulsed Neutron Activation (PNA) [4] flow meter (±5% of reading) which will be operating in a least one of the secondary loops. In addition, temperature data will be obtained from numerous process thermocouples located in both the reactor vessel and primary and secondary loops.

4. TEST DESCRIPTION, SCRAM FROM LOW POWER (5%)

The first test planned to address natural circulation in the FFTF primary loop and reactor vessel is a plant scram from 5% power (20 Mw), 75% flow. The primary pump pony motors will be de-energized just prior to reactor scram so that the reactor will undergo a transition to natural circulation following automatic trip of the pump main motors upon scram. The test will be condu-ted prior to power operation above 5% power and after steady operation for approximately 1 hour at 5% power. The calculated decay heat curve following the transient as shown in Fig. 3. The secondary loops will be operated with pump pony motors and with cold leg temperature controlled (via air flow modulation) during the transient.

This particular test was originally incorporated into the test program to provide experience in planning, conducting, and evaluating a transient natural circulation experiment at a sufficiently low reactor power level that risk to the fuel and the plant components would be minimal even if the plant response exhibited unanticipated behavior. Implicitly, a requirement was established that any significant unanticipated phenomenon be identified. The role of the test was extended somewhat when the project elected to co[illegible] subsequent higher power plant testing upon demonstration by this test [illegible] sufficient plant natural circulation capability to safely remove the decay heat associated with the low total burn-up planned through the subsequent test series. To meet both of these objectives, it was necessary to characterize the feasible range of normal plant response (hence defining an unanticipated result) and to establish a criterion for permitting follow-on tests to proceed as planned. These tasks necessitated studying the relevant uncertainties (in plant data, model parameters, test measurements, and model structure) and then determining whether the test, the instrumentation, and the planned evaluation, given the uncertainties, could be expected to meet the objectives. This process is the detailed focus of this paper; such focus providing direct information for those following the FFTF tests and serving as an exercise in the development of verification and validation technology.

The analyses performed to study this test have employed primarily the Westinghouse proprietary computer program, IANUS [5]. IANUS is a complete plant simulation similar in structure and purpose to the more widely available DEMO program. [6] The principal simplifying assumptions in the model pertinent to the low power test are characterized by the inclusion of only one fully dynamic thermal/hydraulic channel in the reactor and one-dimensional modeling of the fluid flow process. To compensate for the first assumption, which can readily be shown to be invalid with EBR-II data, the model is augmented with a quasi-dynamic treatment of the fuel-bypass flow (18% of total reactor flow at steady-state for FFTF) and the use of flow dependent factors to extrapolate to a particular assembly from the calculated core average assembly. The second assumption is not expected to be important for this test since the fuel assemblies will dominate the loop response and since a flow dependent multiplier is again employed to reflect the effect of intra-channel thermal/hydraulic mixing processes within an assembly. The two flow dependent factors are generated with FLODISC and COBRA as discussed in [1]. The sensitivity of the predictions for this test to the assumptions made in deriving the flow dependent factors and to the uncertainties in actual plant multi-channel behavior, are assessed with FLODISC and CORA [7], which model multiple hydraulically-coupled, adiabatic-boundary channels and fuel as[illegible] clusters with the ENERGY method for interchannel energy transfer, respectively. These programs require an IANUS-generated core driving pressure or total flow boundary condition, with attention required to ensure model compatibility. Sensitivity results are discussed below.

As a baseline for later comparisons, the nominal IANUS predictions for the planned test are shown in Figures 4 and 5, with temperatures and flows respectively. Temperatures shown include average assembly mean coolant at top of active fuel zone, and row 2 FOTA (HF011) peak coolant at top of active fuel zone. Flows include the total loop flow and the estimated HF011 peak temperature rise is predicted to be 31°C occurring 243 seconds into the event. Extrapolation from the core average state variables calculated by IANUS to the HF011 peak measured value was performed using the same flow dependent analysis previously used to provide the hot channel estimate present in the hot channel model. The common model assumptions for the hot channel factor and the HF011 predictions helps assure that the accuracy (or conservatism) of the HF011 T/C prediction can be related the IANUS hot channel model. The nearly linear temperature rise rate from 60-120 seconds has also been selected for evaluation as it represents a significant measurable process characteristic. The heatup rate during this period is effectively a function of decay heat, core mass, and heat capacity with relatively low sensitivity to core hydraulics. A quasi-equilibrium "steady-state" is reached after 600 seconds though upper portions of the outlet plenum, irrelevant to loop thermal head, will still be heating.

5. SENSITIVITY

In order to assess possible alternate outcomes of the test, sensitivity studies were performed with the IANUS model. Two parameters selected for variation in the initial sensitivity study were decay power and pressure drop, the most sensitive design-event parameters. [1] The transient from 5% power is not sensitive to pump coastdown time since peak temperatures occur relatively long after the pump stops and, further, tests already completed have eliminated most coastdown uncertainty. In the period immediately following the pump coastdown a definite relationship is expected between power, flow and ΔT. Decay power, treated as a parameter, directly represents uncertainty in power in this relationship. Flow, in turn, is determined by well-known system geometry, ΔT, pressure drop, and some transient phenomena. Treating the pressure drop as a parameter emulates the flow uncertainty in the relationship. Consequently, it is inferred that results generated over a broad range of these two parameters encompass most foreseeable outcomes of the actual test. Two significant dependent variables which could be measured during a test are the peak temperature rise and the time of the peak temperature rise from the start of the transient in seconds. Figure 6 shows calculated points of peak HF011 temperature rise and elapsed time to the peak for various combinations of decay power and core pressure drop. Higher pressure drop causes a higher peak temperature later in time, while higher decay power leads to an earlier, higher peak. The test result on this grid will permit interpolation of pressure drop and decay power multipliers yielding a simply "calibrated" model, provided no significant deficiencies exist in other areas of the IANUS model.

Additional sensitivity runs were made to analyze the effects of other potential contributors to test uncertainty. The sensitivity to mass in the upper pin structure was obtained. The mass above the core was doubled in one run and was distributed very nonuniformly in a second run. The effects in both cases were small. Other sensitivities examined in this study were the sensitivity of the HF011 T/C reading to the degree of coupling to the mass of the duct wall, the sensitivity to fuel gap conductance, and the sensitivity to initial power. Predictions for the flow and temperature measurement for the test under each of these cases are listed in Table I. These predictions are for: (1) the temperature rise at the top of the active zone in HF011, ΔT_{HF011}, at its peak in the transient, (2) the time of the transient peak, τ_{max}, (3) the change in T_{HF011} between 1 and 2 minutes into the test, T_{1-2}, (4) the loop flow, W_{loop}, at τ_{max}, (5) the flow through HF011, W_{HF011}, at τ_{max}, and 10 minutes after the initiation of the test, (6) ΔT_{HF011}, (7) W_{loop} and (8) W_{HF011}. Table II consists of the results of the Table I analyses expressed as a matrix of sensitivities, reflecting predicted changes in each measured system state assuming a unit change (i.e. 100% change) in each of the uncertain parameters (except for Q_o where the uncertainty was bounded by a 20% change).

A most useful result of these studies is the insensitivity of the temperature rise between one and two minutes to all parameters except the decay power and the duct mass coupling. This insensitivity to flow means that one can estimate the decay power from the HFO11 T/C reading which, in turn, allows estimates of the reactor pressure drop from the flows at peak temperature or from the quasi-steady state temperature and flows. Estimates obtained from these temperature and flow measurements should be consistent with the values of Q_d and ΔP obtained from the grid in Figure 6. In the next section the appropriate degree of consistency is discussed.

The impact of the uncertainties in the parameter values was assessed by bounding them and then using the sensitivity matrix to calculate the resulting uncertainties in the interpretation of the measured values summarized in Table III. Column 3 in Table III presents the measured state uncertainties due to a possible 6% change in the shape of the decay heat curve between the time of the peak and the quasi-steady state at 10 minutes. The uncertainties in Columns 4-7 reflect the results of the Table II sensitivity studies. The estimated uncertainties in M_{1S} and in $M(Z)$ represent plausible bounds; the uncertainty in the duct mass effect was based on a CORA run which indicated a comparatively flat temperature profile across the assembly for the test. Column 8 in Table III consists of the additional uncertainties in the measured values due to statistical hot channel factors.

Column 9 in Table III is an estimate of the effects of three known model structural approximations whose effects were estimated without a direct comparison of IANUS sensitivity runs. The three model approximations are:

- no radial heat transfer in the reactor core or vessel,
- no flow mixing under the instrument tree,
- a single node upper plenum without a stratification model.

All three effects are expected to be small in the low power test. The reliance of the analysis of HFO11 temperatures for verification minimizes the effect of heat transfer since all of the sub-assemblies have power-to-flow ratios which are close to that of HFO11. HFO11 has a stalk which prevents cross flow for the assembly. Finally low flow and plenum ΔT should minimize stratification. Nevertheless, the effects are relevant for this test and in the design case and some uncertainty allowance was judged to be appropriate.

Radial heat transfer beyond immediate neighbor assemblies will act to effectively deposit more decay heat in the fuel bypass channels, resulting in a lowered flow through the core and a later peak temperature - much like increased ΔP and decreased Q_d. Coolant mixing under the instrument tree will tend to reduce the adverse chimney effect caused by the cold sodium in the instrument tree and reduce instrument tree pressure drop, both effects causing a reduction in core ΔP, but without benefit in the FOTA. Since flow will be low and the bypass sodium temperature will be quite close to the upper plenum temperature, plenum stratification causing bypass streaming into the outlet nozzle will be negligible. The simplified upper plenum model would be more significant in the design transient. All three model approximations affect only the core pressure boundary conditions. Hence, the rate of heatup is insensitive to these uncertainties, and an estimate of the decay power in HFO11 based on heatup rate would be unaffected by the uncertainties. Estimates for the effects of these structural approximations on flow sensitive measurements were obtained by treating them as effective ΔP and using the sensitivity matrix. The estimates are listed in Column 10 of Table III.

The subtotals of uncertainties in Column 11 represent the appropriate uncertainties in the measurements when the test results are to be used to estimate Q_d and ΔP. That is, these uncertainties will indicate the precision with which a determination of Q_d and ΔP can be made, given this test. To establish the best possible estimate, there are six methods of obtaining Q_d and ΔP given in Table IV. The uncertainty for each estimate of Q_d and ΔP is calculated from the sensitivites in Table II and the uncertainties in Column 11 of Table III.

6. EVALUATION OF TEST RESULTS

To satisfy the initial test objective, we need only observe that an unanticipated result is one which is outside the bounds specified by the total a priori prediction uncertainty. Alternately stated, the total a priori prediction uncertainty represents the resolution limit for this test to uncover unanticipated phenomena in the FFTF. Failure to uncover an unanticipated result will provide a technical basis for proceeding with the natural circulation test plan for the steady state tests. In these tests and subsequent transient tests, more precise screening for unanticipated phenomena can be performed. Should an unanticipated result occur in the low power test, a more detailed analysis and a closer examination would precede the next natural circulation test; some changes in the follow-on test plan might be made to resolve any outstanding anomaly.

The second test objective, verification of sufficient plant natural ciruclation capability to permit plant testing at higher power levels, can be met by performing an approximate calibration of the model by estimating the most sensitive parameters, Q_d and ΔP, using the low power test result. Since these are only two of many actual uncertainties (plant data parameters, model structure), the estimates for Q_d and ΔP are expected to be distored from real values. To compensate for the known approximations involved, the error bounds from the intersection of the six methods in Table IV will be used to generate an extreme, conservative estimate of Q_d and ΔP. This constitutes a "worst case" estimate of results of the low power test. Further, to allow for potential unkown error sources, we use this result as we would use a nominal value and we apply the conservatism of the "safety evaluation model". Even with this conservative procedure, the margin to the established limit is sufficiently large to allow an expectation of a successful test result. From this result we conclude that the low power test, while inherently imprecise, is suited to the intended purpose. Should the final estimate be beyond the established limit, additional evaluation or supplemental testing will be performed.

7. SUMMARY AND CONCLUSION

In the FFTF natural circulation acceptance test plan a low power scram test will be performed prior to operation of the plant at a power level above 5%. This test result will be used to determine effective values of decay power and reactor pressure drops; these values will be used as nominal imputs to the "safety evaluation model", providing an approximate, but sufficiently accurate calibration to the model to permit demonstration of safe decay heat removal through the ensuing test series, pending completion of additional, more precise natural circulation tests. The meaning of the low power test measurements will be partially obscured by IANUS model simplifications, parameters and plant data uncertainties, and the measurement inaccuracies. As a result the determination of decay power and pressure drop is less certain than measurement errors alone would indicate. This study establishes a priori bounds for total uncertainty and also prescribes the use of redundant measurements to help to improve the quality of the determination. An uncertainty-broadened criterion for model acceptance is also developed. The conclusion is that the proposed test is designed to satisfy the established objectives. Further the techniques applied here, when uncertainties are large, may also prove useful for designing and evaluating future tests planned for model verification and validation.

ACKNOWLEDGEMENTS

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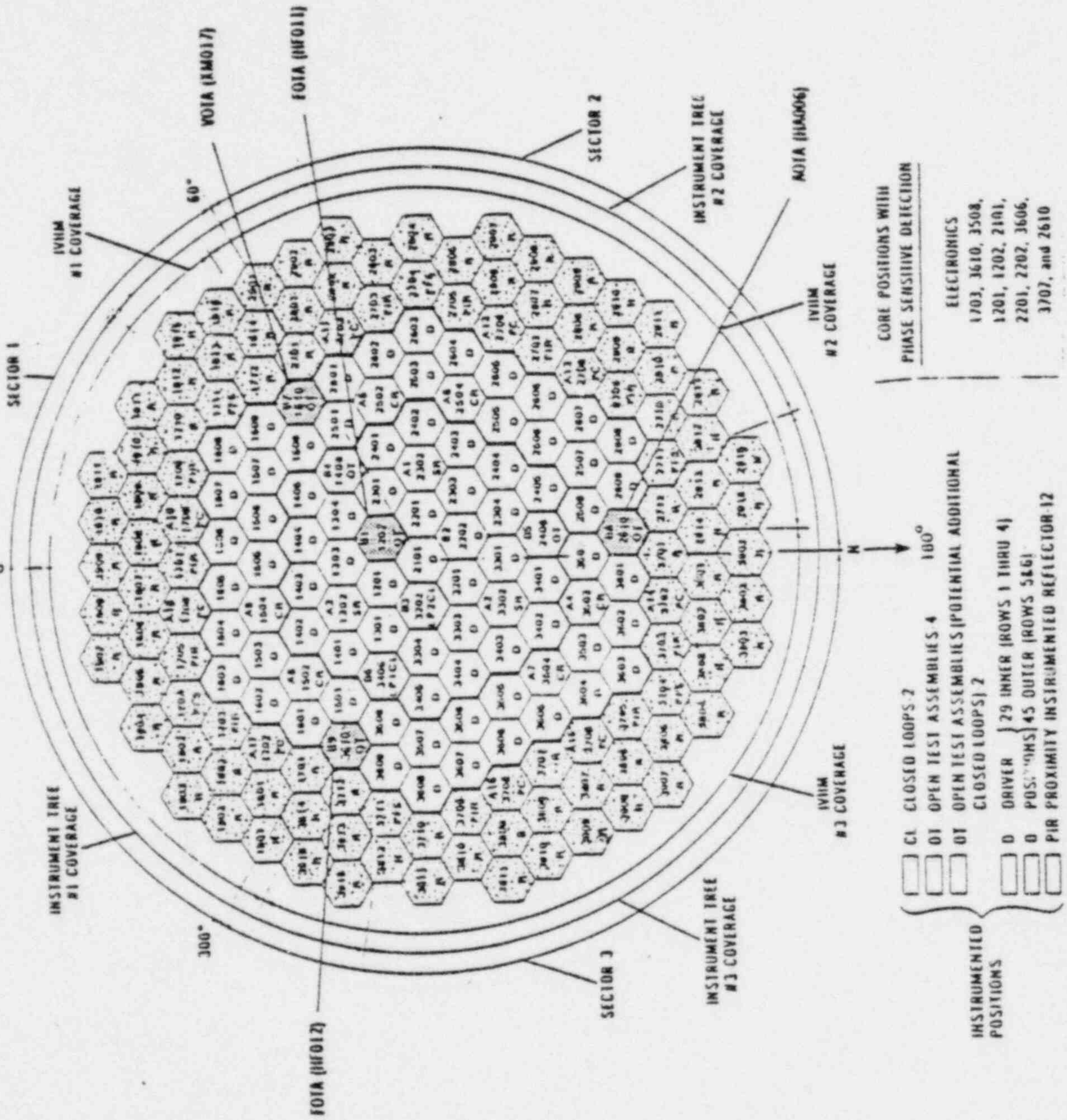
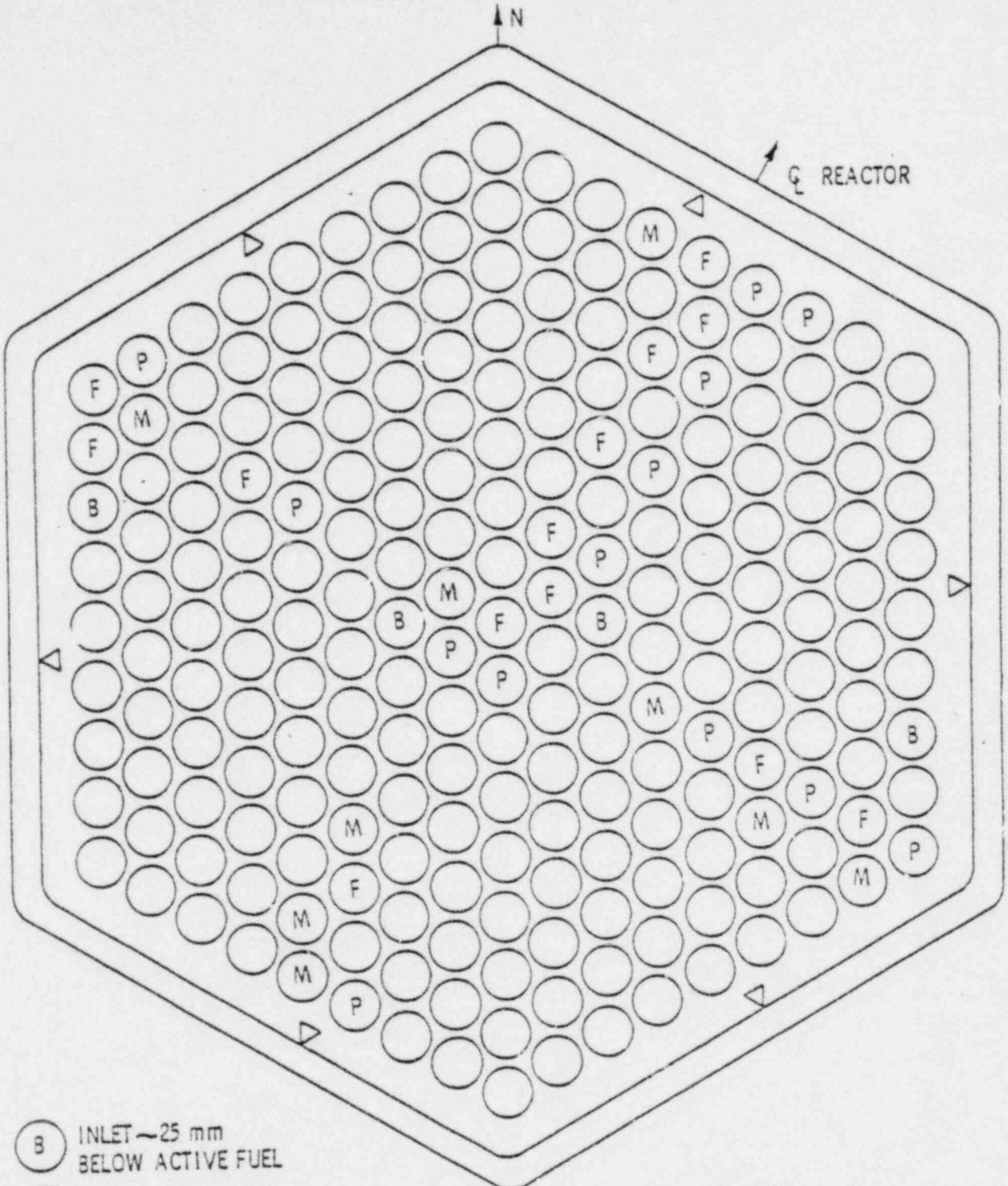


Figure 1
FFTF Core Map for Natural Circulation Tests



(B) INLET ~25 mm BELOW ACTIVE FUEL

(M) MIDPLANE ~25mm ABOVE MIDPLANE

(F) ~25 mm ABOVE TOP OF ACTIVE FUEL

(P) TOP OF PIN, 0 TO 50 mm BELOW TOP OF PIN

◁ DUCT WALL ~25 mm ABOVE TOP OF ACTIVE FUEL

Figure 2
FFTF Fueled Open Test Assembly (FOTA), HF011 Thermocouple Locations

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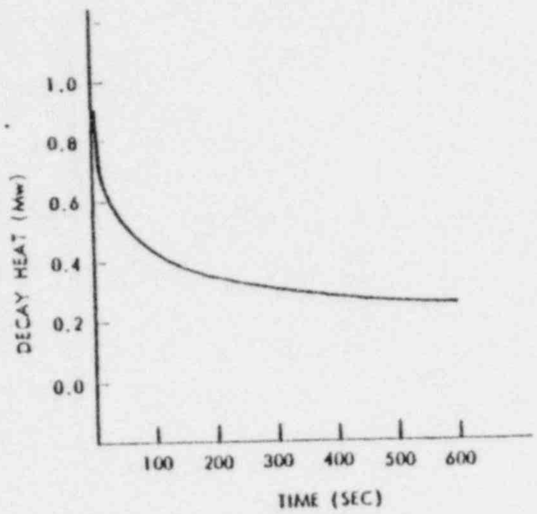

Figure 3. DECAY HEAT IN LOW-POWER TEST

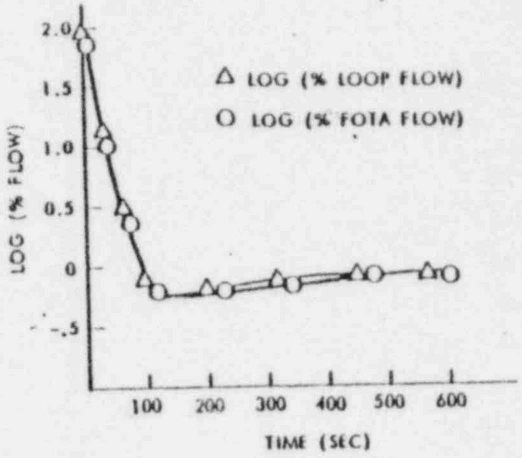

Figure 5. LOOP AND FOTA FLOW FOR LOW-POWER TEST

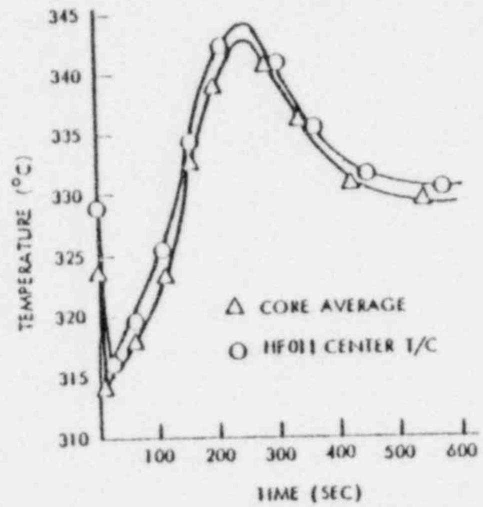

Figure 4. TRANSIENT TEMPERATURE AT THE TOP OF THE ACTIVE ZONE

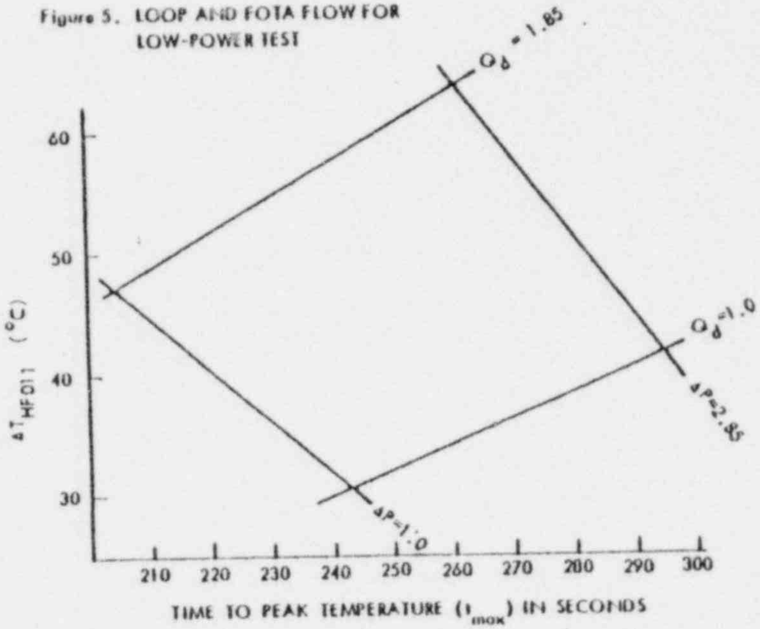

Figure 6. GRID FOR DETERMINING Q_d AND ΔP

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Table I

Sensitivity Analysis Runs for 5% Transient Natural Circulation Test

				TRANSIENT PREDICTIONS					STEADY STATE PREDICTIONS		
No.	Q_d	ΔP	Q_o(%)	ΔT_{HF011}(°C)	T_{max}(sec)	T_{1-2}(°C)	$W^1_{loop}(t=t_{max})$	$W_{HF011}(t=t_{max})$	ΔT_{HF011}(°C)	W_{Loop}(%)	W_{HF011}(%)
1	1	1	5	31	243	9	0.512	0.645	18	0.625	0.712
2	1	2.85	5	43	295	9	0.410	0.469	24	0.474	0.528
3	1.85	1	5.	49	205	17	0.572	0.871	26	0.718	0.887
4	1.85	2.85	5	66	262	17	0.469	0.563	37	0.528	0.614
5[5]	1	1	5	32	251	9	0.492	0.665	17	0.660	0.732
6[2]	1	1	5	32	218	11	0.490	0.600	16	0.612	0.689
7[2]	1	2.85	5	41	257	11	0.387	0.441	22	0.463	0.510
8[3]	1	1	5	30	242	9	0.520	0.646	17	0.629	0.715
9	0.83	1	6	33	235	9	0.572	0.675	17	0.687	0.757
10	1.25	1	4	34	250	9	0.444	0.630	18	0.557	0.670
11	1.25	2.85	4	44	305	9	0.377	0.452	26	0.440	0.511
12[4]	1	1	5	34	243	9	0.512	0.645	18	0.625	0.712

1 Flows are expressed as percent of design flow

2 This run had no coupling of the coolant at the top of the active zone to the duct.

3 Gap conductivity 3/16 nominal

4 Redistributed mass above active zone

5 Mass above core doubled

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POOR ORIGINAL

Table II

Sensitivity Matrix for 5% Transient Natural Circulation Test

MEASUREMENTS	PARAMETERS					
Transient	ΔP	Q_d	Upper Structure Mass	Duct	Q_o(%)	Axial Distribution of Mass
ΔT_{HF011}(°C)	6.3	21	0.83	1.1	3.4	-2.8
t_{Max}(sec)	28	-45	7.5	-25	-16	0
T_{1-2}(°C)	0	8.8	0	1.1	1.4	0
W^2_{Loop}(%)	-0.0541	0.071	-0.02	-0.02	0.079	0
W_{HF011}(%)	-0.0973	0.271	0.02	-0.05	-0.079	0
Steady State						
T_{HF011}(°C)	3.6	9.4	-0.3	-2.2	1.1	0
W_{Loop}(%)	-0.0811	0.106	0.035	-0.01	0.086	0
W_{HF011}(%)	-0.0973	0.212	0.040	-0.02	0.087	0

1 The variation in predicted values for a 100% change in the parameter (except Q_o).

2 All flows are expressed as percent of design flow.

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Table III

Uncertainty Summary for 5% Transient Natural Circulation Test

Measurement	Uncertainty[2]	Q(t)	M_{us}	Duct	Q_o	M(z)	Hot Channel[3] Statistical Factor	Other[4]	Subtotal[6] Prediction Uncertainty	Total[7] Prediction Uncertainty
		Anticipated Model Uncertainties[1]								
Transient										
ΔT_{HF011}(°C)	±3	0	±0.2	±0.1	±3.4	±1.4	±1.6	±2	±5.4	±6.5
t_{max}(sec)	0	0	±1.5	±2.6	±16.5	0	±4.4	±10	±20	±24
T_{1-2}(°C)	±1	0	0	±0.1	±1.4	0	±0.6	0	±1.8	±2.1
W_{Loop}(%)[5]	±0.05	0	0	0	±0.08	0	±0.01	0	±0.09	±0.09
W_{HF011}(%)	±0.18	0	0	0	±0.08	0	±0.02	0	±0.2	±0.21
Steady-State										
ΔT_{HF011}(°C)	±3	±0.6	±0.06	±0.02	±1.1	0	±0.06	0	±3.3	±3.7
W_{Loop}(%)	±0.05	0	±0.01	0	±0.09	0	±0.01	0	±0.1	±0.11
W_{HF011}(%)	±0.18	±0.01	±0.01	0	±0.09	0	±0.02	0	±0.2	±0.21

1. These estimates are based on the assumption that $\Delta Q(t) = \pm 6\%$, $\Delta M_{us} = \pm 0.2$, $\Delta Duct = \pm 0.1$, $\Delta Q = \pm 1\%$ and $\Delta M(z) = \pm 0.5$.
2. Represents 3σ values on the data acquisition.
3. Statistical factors due to inlet flow maldistribution, rod bowing, and power distribution.
4. Estimated uncertainty due to radial heat transfer, upper plenum modeling and sub-assembly cross-flows.
5. All flows are expressed as percent of design flow.
6. This is the root-mean-square of all the uncertainties relevant to the calculation of Q_d and ΔP.
7. Uncertainties of 0.4 in ΔP and 0.12 in Q_d are included.

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Table IV
Uncertainty in Decay Heat (Qd) and Pressure Drop (ΔP)

Method	Measurement #1	Measurement #2	Uncertainties Qd	Uncertainties ΔP
1	ΔT_{HF011} $[t_{max}]$	t_{max}	0.29	0.53
2	T_{1-2}	$W_{loop}[t_{max}]$	0.24	1.17
3	"	$W_{HF011}[t_{max}]$	"	2.2
4	"	$T_{HF011}[SS]$	"	1.0
5	"	W_{loop} [SS]	"	1.3
6	"	$W_{HF011}[SS]$	"	2.1

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APPENDIX C

PRETEST PREDICITONS OF THE THERMAL AND HYDRAULIC RESPONSES OF THE FUELED OPEN TEST ASSEMBLIES TO THE 5% POWER NATUAL CIRCULATION FFTF PLANT STARTUP TEST, HEDL-TC-1778, OCTOBER 1980