GPU Nuclear

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August 11, 1982
5211-82-188

Office of Nuclear Reactor Regulation
Attn: John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Reactor Building Spray System Modifications (RBSS)

Recent discussions with your staff indicate that two items remain outstanding before our RBSS modifications and Technical Specification Change Request No. 107 can be approved.

Item 1

The concentration of NaOH should be minimized (and therefore, the level in the NaOH tank should be maximized) to reduce corrosion and the potential health hazard in the case of inadvertent actuation.

Response:

Results of Attachment 1 indicate that by reducing the NaOH concentration to 10 weight percent and raising the initial NaOH tank level (i.e., reducing the difference in height between the BWST and the NaOH tank from 10 feet to 8 feet) optimum system performance is achieved. Under worst case single failure LOCA conditions the pH levels in containment are reduced and the dynamic consideration of the two tank system are improved over previously supplied values. Technical Specification Change Request 107 will be submitted to include these revised tank levels.

Item 2

The performance of this two tank gravity feed system should be demonstrated because of the sensitivity of the dynamic equilibrium and flow restriction piping network.

A001

Response:

Attachment 2 is provided to benchmark the model used in our analysis and thereby show that the model accurately predicts tank drawdown at TMI-1. Comparison of the result with Crystal River 3 and Virgil C. Sumner nuclear plants are provided. Based on this information, we do not believe a drawdown test of the BWST and NaOH tank is warranted.

Sincerely,

H. D. Hukill
Director, TMI-1

HDH:LWH:vjf
Enclosures
cc: R. C. Haynes
R. acobs
W. Pasedag

ATTACHMENT 1

# NaOH CONCENTRATION

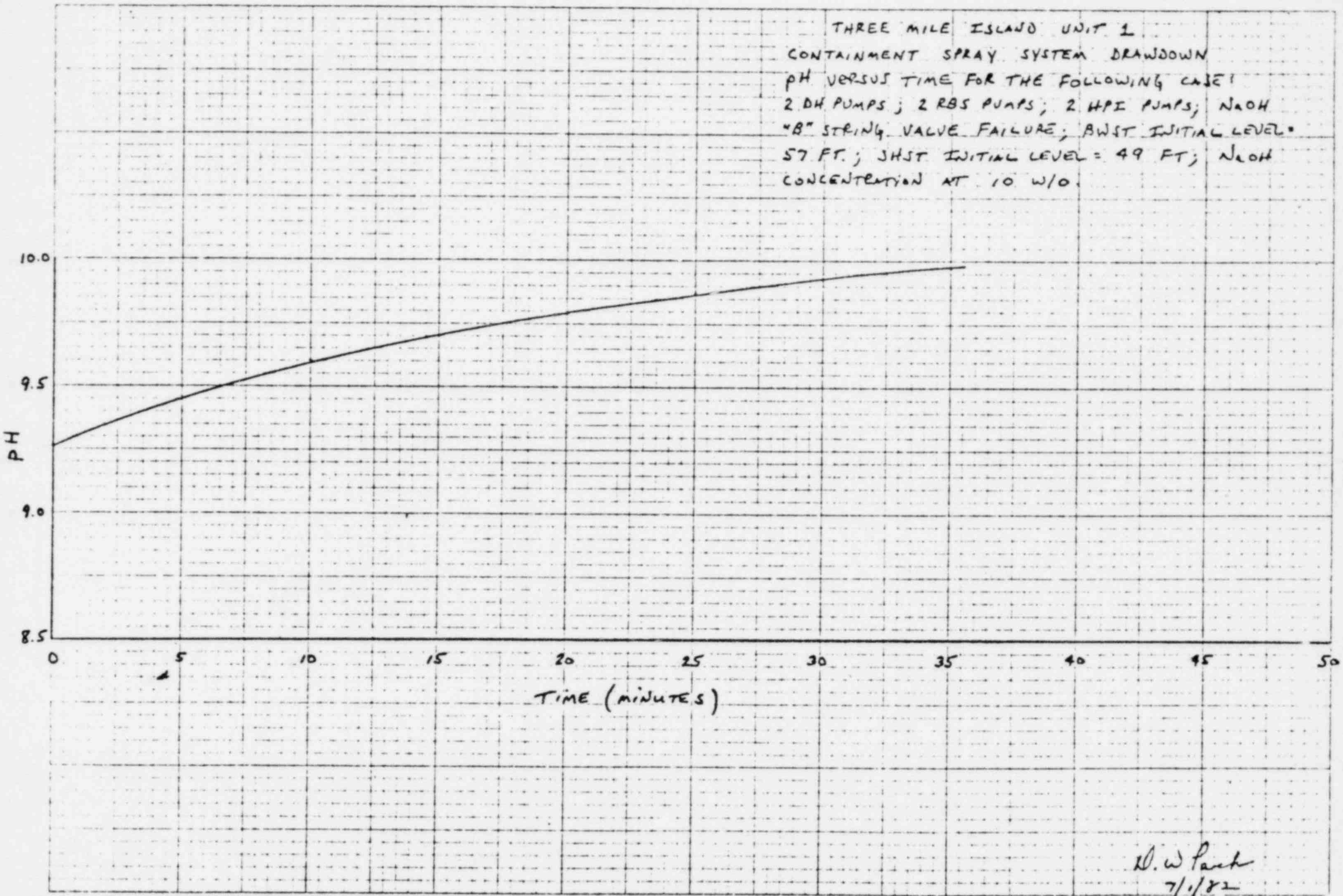
# REDUCTION ANALYSIS

# FOR TMI-1

Three Mile Island Unit 1
Tabular Data Used to Generate Curves for the
Following Case: 2 DH Pumps; 2 RBS Pumps; 2 HPI
Pumps; NaOH "B" String Valve Failure; BWST Initial Level =
57 ft; SHST Initial Level = 49 ft; NaOH Concentration
at 10 w/o

| Time (Minutes) | Tank Level Above ℄ Outlet Nozzle (feet) BWST | SHST | Molarity at Injection Point (Moles/Liter) | pH* |
|---|---|---|---|---|
| 0 | 57.00 | 49.00 | .1209 | 9.26 |
| 5 | 49.34 | 45.05 | .1379 | 9.45 |
| 10 | 41.70 | 40.54 | .1513 | 9.60 |
| 15 | 34.07 | 35.59 | .1600 | 9.70 |
| 20 | 26.46 | 30.36 | .1667 | 9.79 |
| 25 | 18.85 | 24.90 | .1739 | 9.87 |
| 30 | 11.26 | 19.22 | .1786 | 9.93 |
| 35 | 3.67 | 13.38 | .1822 | 9.97 |
| 36 Minutes 6 seconds | 2.00 | 12.07 | Extrapolate | |

* The pH Values are read from the curves attached.


THREE MILE ISLAND UNIT 1
CONTAINMENT SPRAY SYSTEM DRAWDOWN
pH VERSUS TIME FOR THE FOLLOWING CASE:
2 DH PUMPS ; 2 RBS PUMPS ; 2 HPI PUMPS ; NaOH
"B" STRING VALVE FAILURE ; BWST INITIAL LEVEL = 57 FT. ; SHST INITIAL LEVEL = 49 FT ; NaOH
CONCENTRATION AT 10 w/o.
10.0
9.5
9.0
8.5
pH
0
5
10
15
20
25
30
35
40
45
50
TIME (MINUTES)
D. W. Pack
7/1/82

Three Mile Island Unit 1
Tabular Data Used to Generate Curves for the Following Case: 2 DH Pumps; 2 RBS Pumps; 2 HPI Pumps; NaOH "B" String Valve Failure; BWST Initial Level = 57 ft; SHST Initial Level = 47 ft; NaOH Concentration at 10 w/o

| Time (Minutes) | Tank Level Above ℄ Outlet Nozzle (feet) BWST | SHST | Molarity at Injection Point (Moles/Liter) | pH* |
|---|---|---|---|---|
| 0 | 57.00 | 47.00 | .1104 | 9.15 |
| 5 | 49.33 | 43.39 | .1304 | 9.35 |
| 10 | 41.68 | 39.13 | .1434 | 9.50 |
| 15 | 34.04 | 34.44 | .1538 | 9.63 |
| 20 | 26.42 | 29.41 | .1630 | 9.75 |
| 25 | 18.81 | 24.08 | .1699 | 9.83 |
| 30 | 11.21 | 18.53 | .1761 | 9.90 |
| 35 | 3.62 | 12.77 | .1806 | 9.95 |
| 36 Minutes 4 seconds | 2.00 | 11.514 | Extrapolate | |

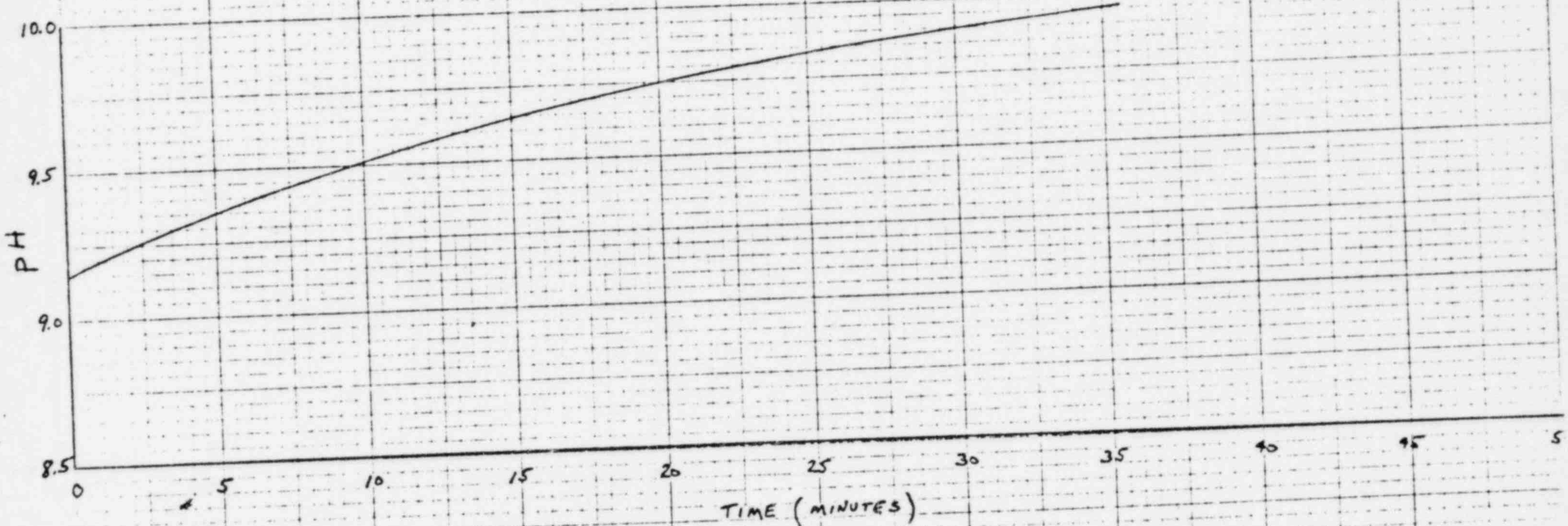
* The pH values are read from the curves attached.

THREE MILE ISLAND UNIT 1

CONTAINMENT SPRAY SYSTEM DRAWDOWN

pH VERSUS TIME FOR THE FOLLOWING CASE:

2 DH PUMPS; 2 RBS PUMPS; 2 HPI PUMPS; NaOH "B" STRING VALVE FAILURE, BWST INITIAL LEVEL = 57 FT.; SHST INITIAL LEVEL = 47 FT; NaOH CONCENTRATION AT 10 W/O.




D. W. Pueck
7/1/82

ATTACHMENT 2

# BENCHMARK FOR

# THE TWO TANK DRAWDOWN MODEL

# FOR TMI-1

Objective:

This document justifies the use of the PIPF Computer Program to predict that the drawdown transient during post-accident operation of the containment spray/decay heat systems.

The justification is divided into four parts:

A) An explanation of the applicability of the code is presented;

B) An explanation demonstrating that either a two tank or three tank configuration doesn't change the basic methodology;

C) A description of the TMI-1 water test results compared to the computer mode results;

D) A discussion of the validation of the computer models for other plants.

A. The PIPF code was developed for the purpose of calculating the steady state distribution of flow to all paths of any given incompressible fluid flow system. Two topical reports, one proprietary (1), the other non-proprietary (2), describing the computer code, its physical basis, its verification, and its use were submitted for review to the NRC in December 1976. A topical report evaluation issued in March of 1978 approved the use of PIPF as an acceptable program for evaluating the flow distribution and pressure balance of hyraulic networks. In addition, the NRC determined that the application of the code to system transients by the quasi-steady state method of analysis is also acceptable (3, 4).

The appplication of the code transient analysis is based on the assumption that flow rates change slowly and smoothly with time. This limitation on the application of the PIPF code must be addressed for each licensing application. Two other limitations enumerated by the NRC for each licensing calculation are (1) verifying that the assumption of incompressible, low temperature fluid conditions is justified and (2) justifying the omission of energy conservation.

The simulation of the borated water storage tank (BWST), sodium hydroxide storage tank (SHST), and sodium thiosulfate storage tank (STST) drawdown transient following activation of the containment spray/decay heat system pumps is the licensing application for which PIPF was written. Before a prediction of system performance under operating conditions can be made, it is necessary to justify the applicability not only of PIPF but also the system model. The procedure for verifying the accuracy of the system model is comparison to a water test.

The site water test is the benchmark for the PIPF model. Using system drawing and equipment data, an analytical PIPF model was created. The model is executed using initial tank and pump parameters that correspond with initial water test data. The transient drawdown was simulated by performing a series of steady state distributions with a suitably small interval between the steady state distributions. The results of the analytical prediction were compared to the water test. Good correlation between results confirmed the analytical model. After the water test comparison confirmed the model, a multitude of variations to the system were evaluated using the PIPF model.

The drawdowns performed on Three Mile Island Unit 1 using the PIPF program satisfy the three limitations imposed by the NRC as noted: 1) the fluids used for the drawdowns were low temperature and incompressible; 2) the fluid temperatures were comparable; 3) since pumps were operating at a constant flow rate and were taking suction from large tanks, the flow rate changes were slow and smooth.

The results of the TMI-1 water test/analytical prediction comparison were good. These results were included as Appendix F to the reference 1 and 3 topical reports and were evaluated by the NRC as partial evidence of PIPF's analytical accuracy.

B. The primary intent of the water test/analytical prediction exercise was to correlate physical piping to a computer model of the piping. If the performance of a constructed system can be predicted by PIPF and confirmed by a water test, it is reasonable to assume that alterations to the model can justifiably represent alterations to the system.

The water test for TMI was performed on a three tank system. The computer model predicted successfully the drawdown transient and thereby established itself as a benchmark for the system. Variations were then performed on the model with the confidence that PIPF needed only to iterate using the basis laws of fluid mechanics to evaluate the effects.

Some of the variation that have been evaluated using the PIPF model for TMI are valve failures, tank starting level fluctuations, fluid density and viscosity studies, pump failures, and proposed piping changes. Each of these studies performed using the PIPF model were based on the confidence gained by the water test comparison.

A two tank drawdown model is again a variation that the PIPF model can realistically evaluate. Since the entire three tank system has been succesfcully modeled and flow tested, a model that simply deletes existing piping will not introduce uncertainties.

D. As noted earlier, the results of the TMI water drawdown comparison were includes as Appendix F in the reference 1 and 3 topical reports. The pertinent parts of the Appendix are discussed below.

Attached are Figures F-1 to F-9 from Appendix F. Figures F-1 is schematic to the decay heat and reactor building spray system. Figure F-2 is a model of decay heat and reactor building spray systems for full flow drawdown analysis while Figure F-3 is the model for half flow drawdown analysis. The remaining figures provide the comparison between test data and analytical data for each tank in the full flow and half flow models.

Test procedure 204/3 of TMI Unit 1 was completed during the TMI-1 initial startup program. This test provided the results of the site water test that were interpreted and plotted relative to the level tap for each tank.

Analytical predictions were made of the results of the drawdown starting at the identical tank levels and controlling the pumps to the identical flow rates.

The flow rate changes were slow and smooth such that a time increment of 5 minutes was sufficient to simulate the transient. In other words, a steady state balance was performed on the system at time = 0 minutes. The flow rate (in GPM) from each tank was assumed to be constant for 5 minutes. At time equal to 5 minutes, tanks are adjusted to the lower levels and the model is re-executed. A transient profile thus develops until a tank termination level is reached (see Appendix C of references 2 and 4 for further explanation of the proecedure).

Figures F-4 and F-9 show graphically the correlation between test and analysis. The tail at that end of test data in figures F-4, F-5 and F-7 is a result of pump throttling prior to shutdown and should be disregarded. Likewise, the test data of figures F-7 and F-9 demonstrates the effect of a pump flow rate change during the test; the close correlation is still obvious.

D. Water test comparisons have been performed for two other nuclear power plants, Florida Power Corporation's, Crystal River Unit 3 (CR3) and the South Carolina Electric & Gas, Virgil C. Summer Unit 1 (VCS).

Water tests were performed on the CR3 system in 1976. For CR3 a PIPF model of the three tank system was developed and successfully predicted the results. The comparison was included as Appendix G to the reference 1 and 3 topical reports. Apprendix G was submitted in the topical report as further evidence of the validity of the PIPF modeling technique.

The VCS reactor building spray and decay heat system contains only two tanks. In response to an FSAR question from the NRC, a pre-operational water test was performed in 1979. Drawdown results were predicted using a PIPF model. The curves showing the comparison are included as attachment 10-13. It is noted that four combinations of pumps were operated during the water test drawdown. Each of these combinations is noted on a separate curve.

A good correlation between test results and analytical predictions on both plants was achieved. This further substantiates the validity of the PIPF Code.

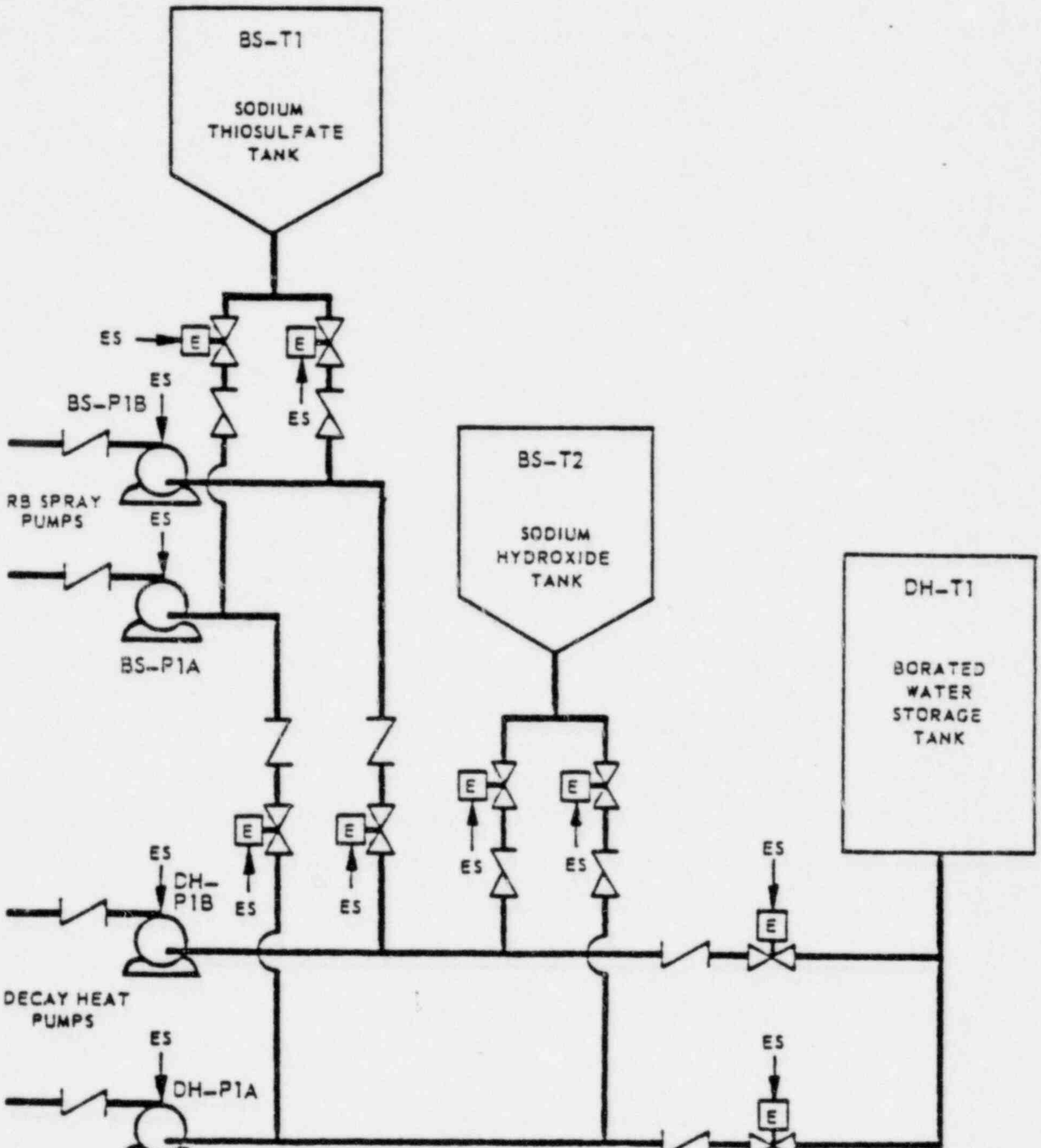



FIGURE F-1

THREE MILE ISLAND NUCLEAR STATION - UNIT 1
SCHEMATIC OF DECAY HEAT AND REACTOR BUILDING SPRAY SYSTEMS

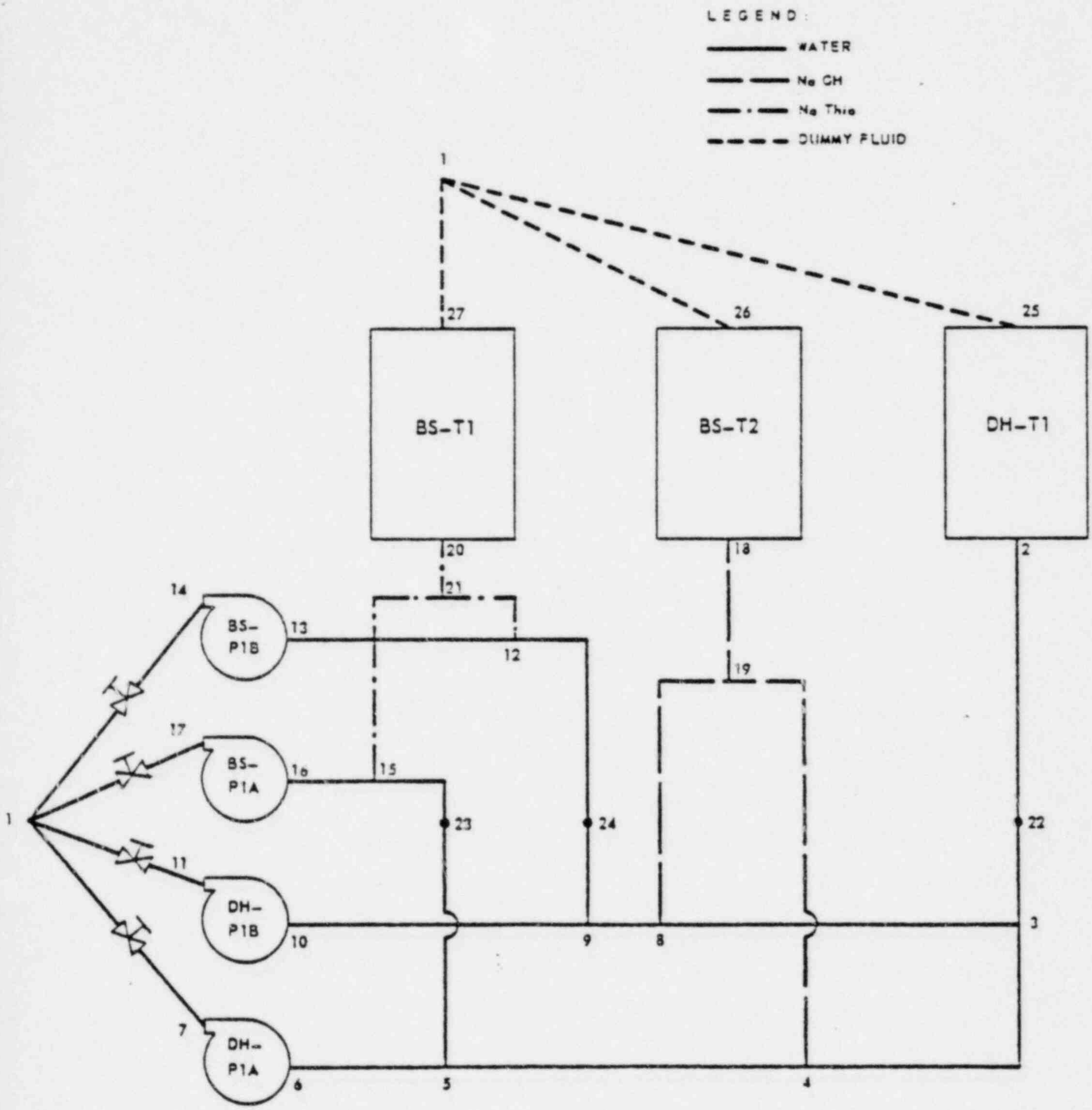



FIGURE F-2

THREE MILE ISLAND NUCLEAR STATION - UNIT 1

MODEL OF DECAY HEAT AND REACTOR BUILDING SPRAY SYSTEMS FOR FULL FLOW DRAWDOWN ANALYSIS

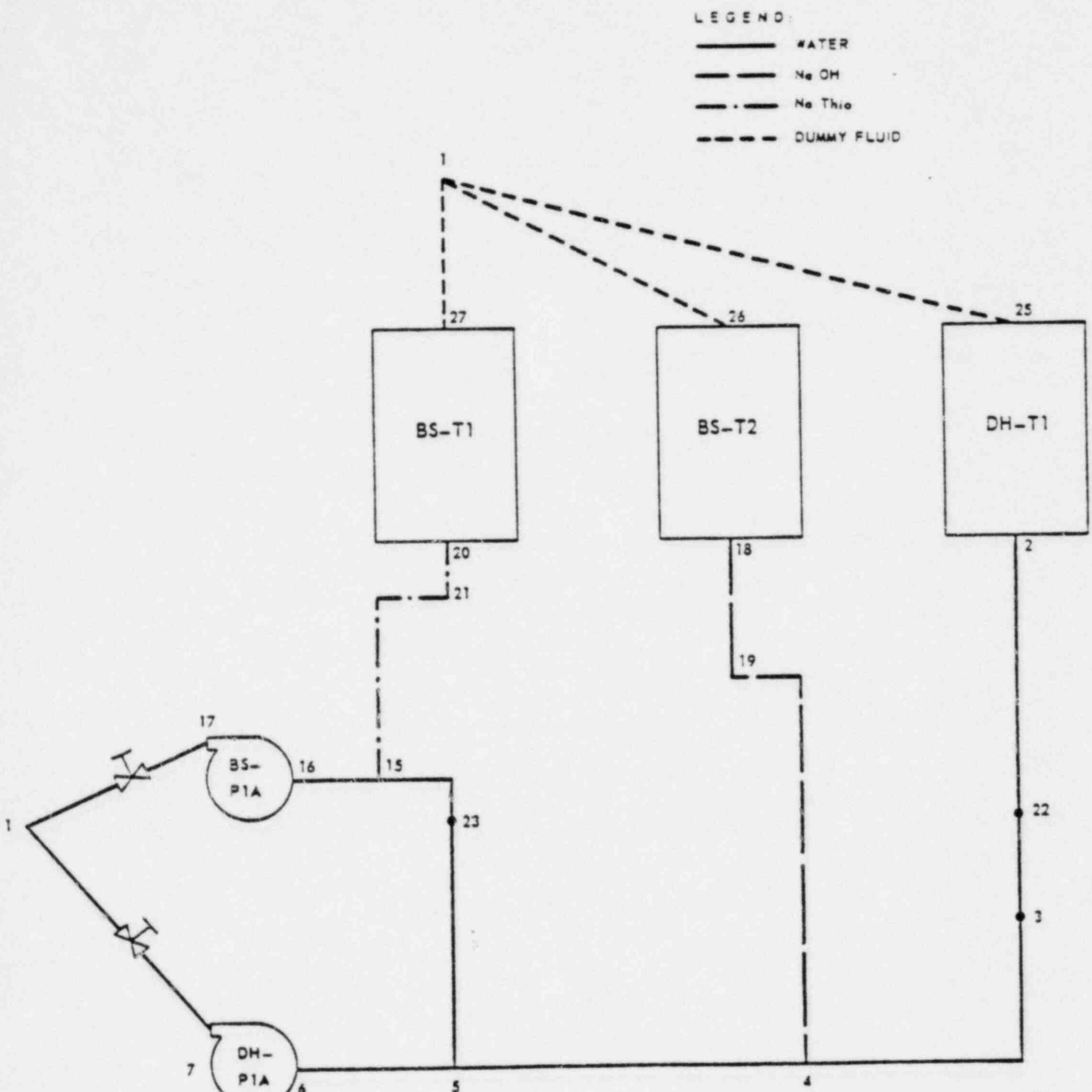



FIGURE F-3

THREE MILE ISLAND NUCLEAR STATION - UNIT 1

MODEL OF DECAY HEAT AND REACTOR BUILDING SPRAY SYSTEMS FOR HALF FLOW DRAWDOWN ANALYSIS

Attachment 4

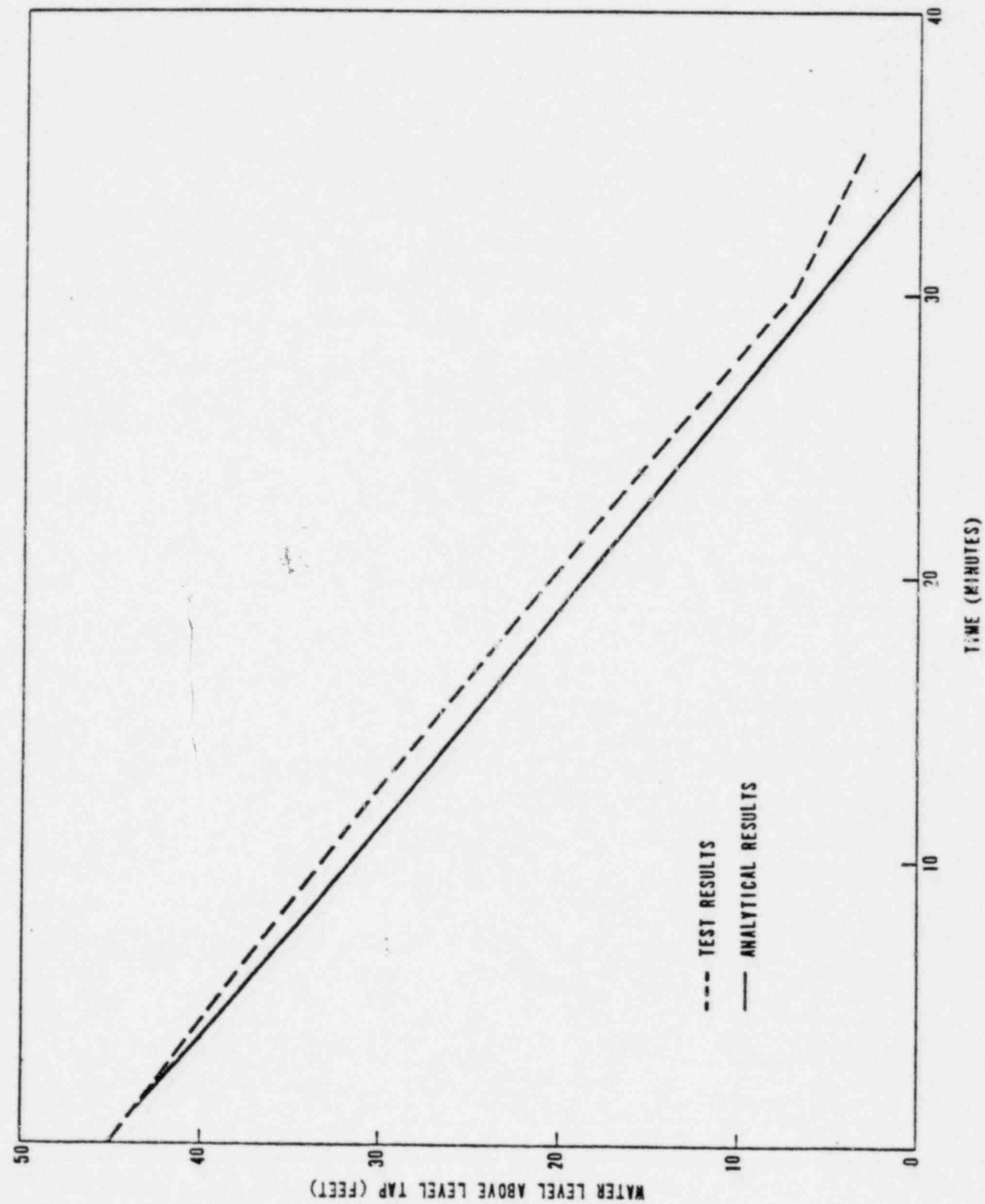



FIGURE F-4

THREE MILE ISLAND NUCLEAR STATION UNIT 1

COMPARISON OF ANALYTICAL RESULTS TO FULL FLOW WATER TEST
BORATED WATER STORAGE TANK DRAWDOWN

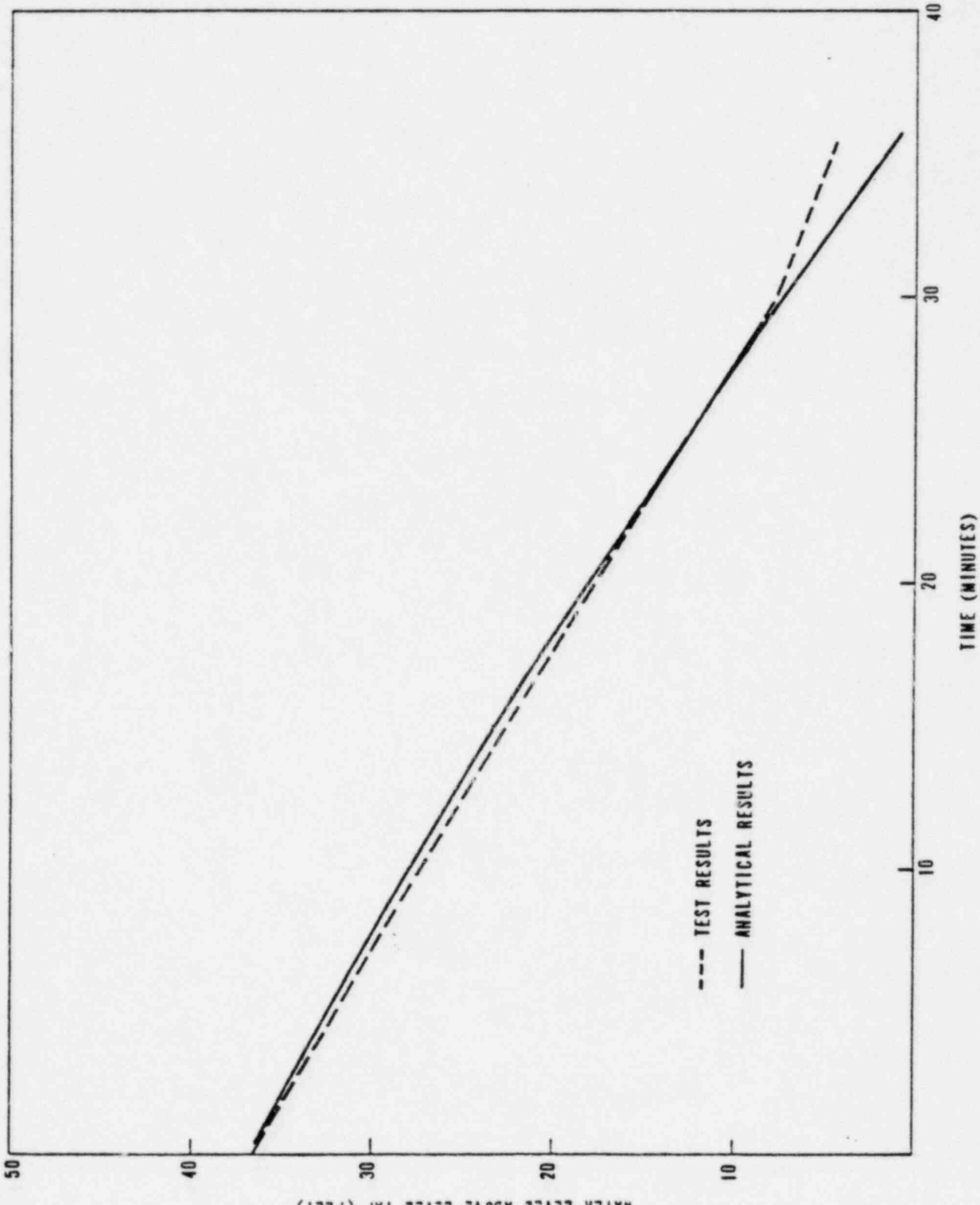



FIGURE F-5

THREE MILE ISLAND NUCLEAR STATION UNIT 1

COMPARISON OF ANALYTICAL RESULTS TO FULL FLOW WATER TEST

SODIUM THIOSULFATE STORAGE TANK DRAWDOWN

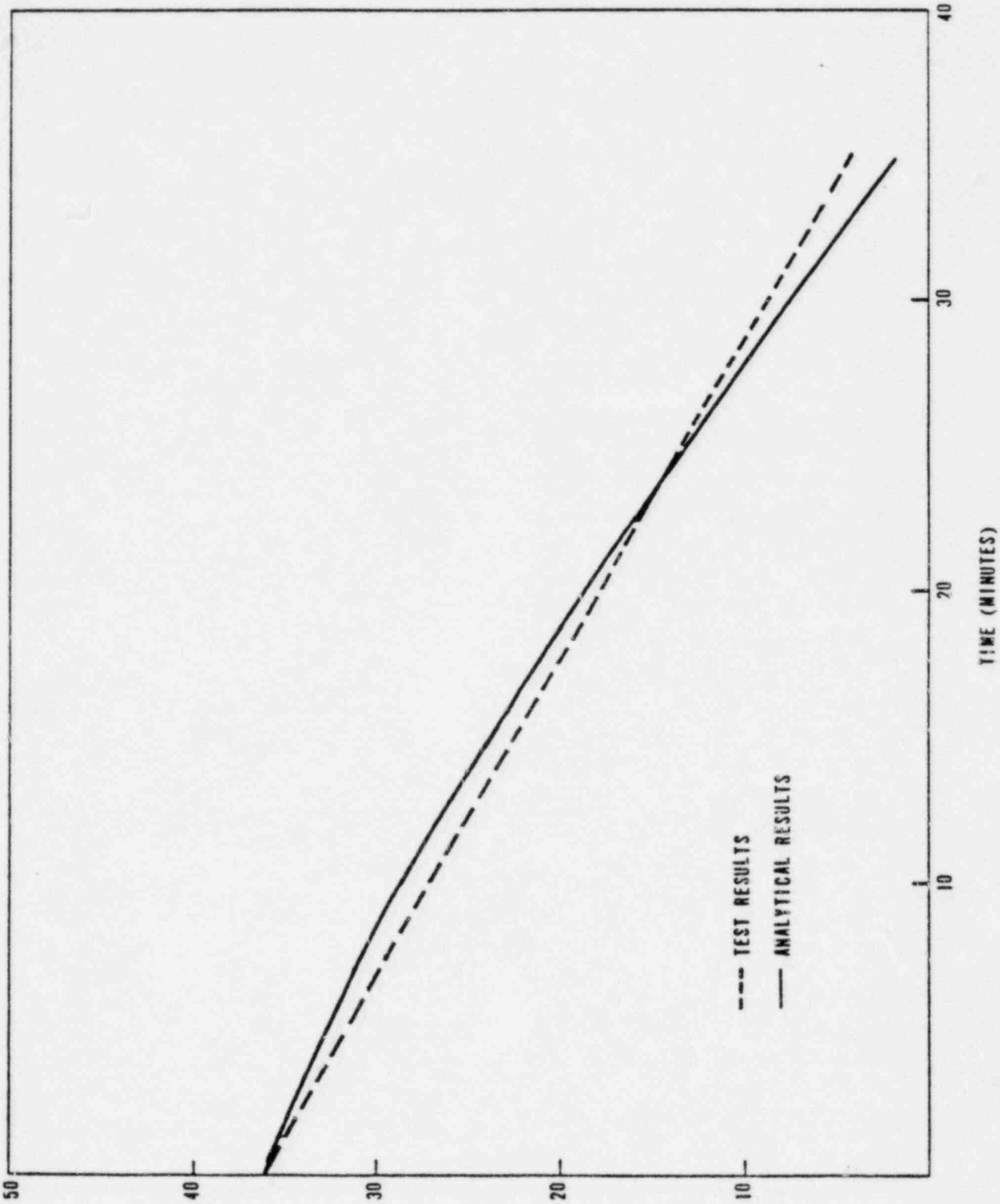



FIGURE F-6

THREE MILE ISLAND NUCLEAR STATION UNIT 1

COMPARISON OF ANALYTICAL RESULTS TO FULL FLOW WATER TEST
SODIUM HYDROXIDE STORAGE TANK DRAWDOWN

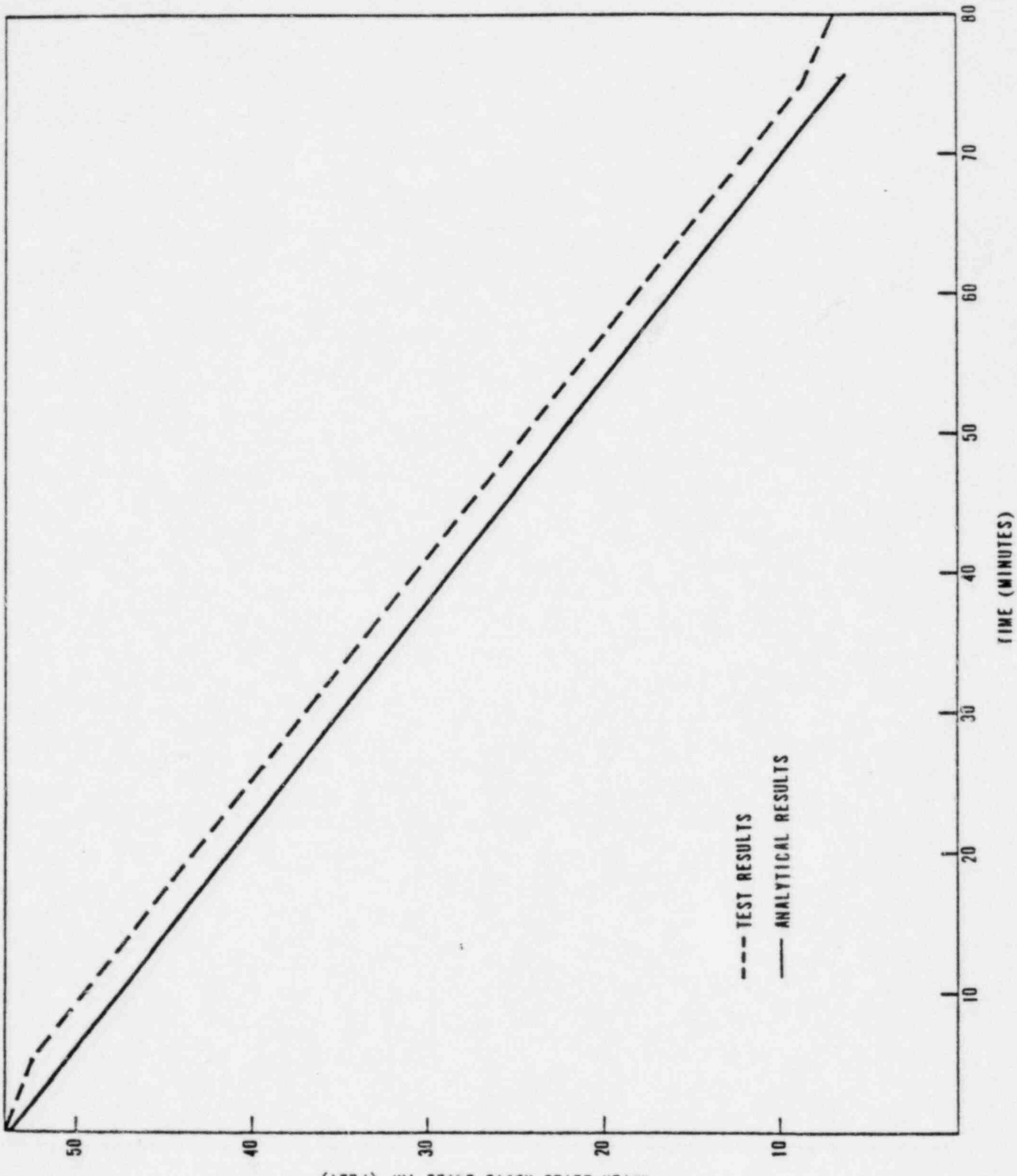



FIGURE F-7

THREE MILE ISLAND NUCLEAR STATION UNIT 1

COMPARISON OF ANALYTICAL RESULTS TO HALF FLOW WATER TEST
BORATED WATER STORAGE TANK DRAWDOWN

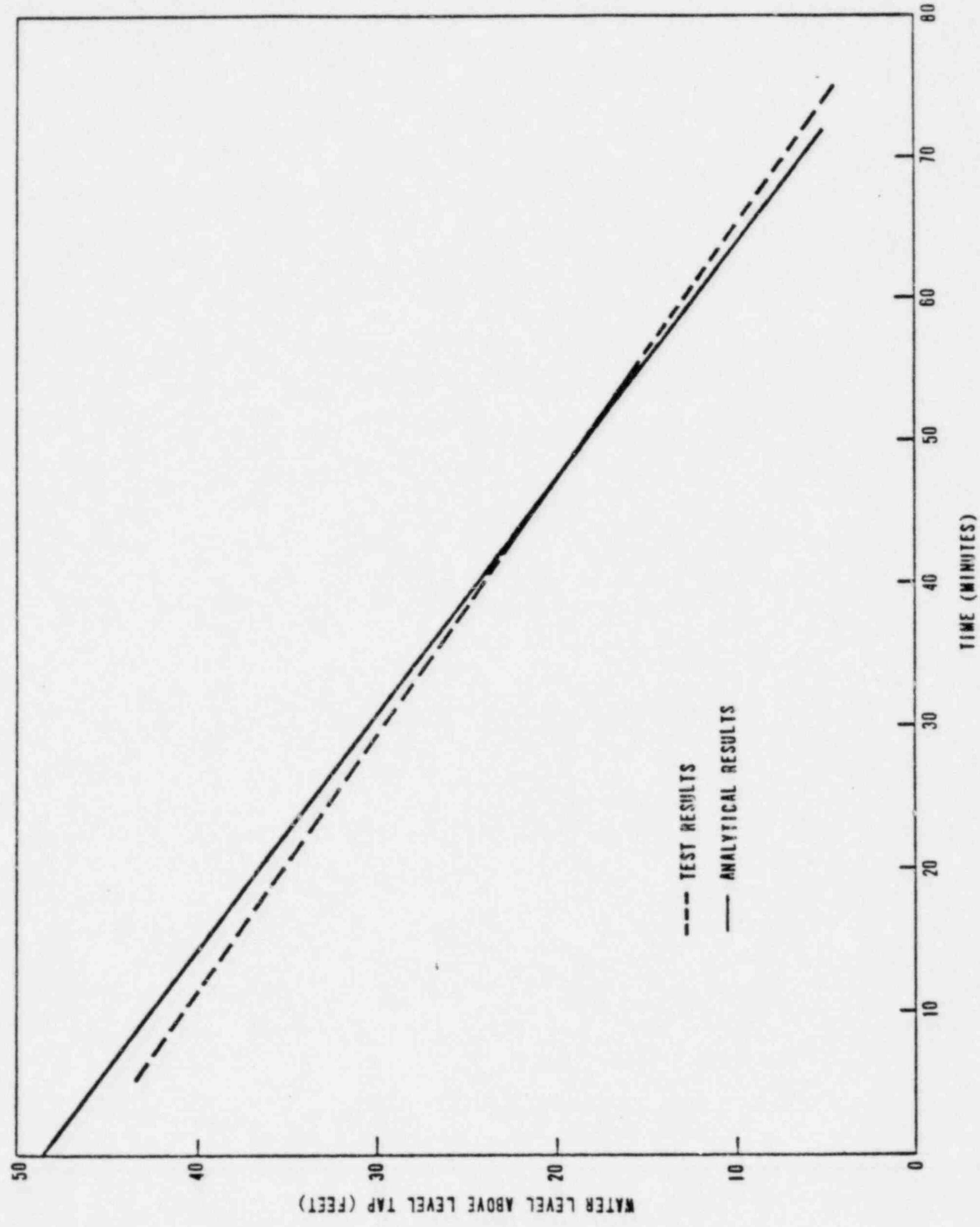



FIGURE F-8

THREE MILE ISLAND NUCLEAR STATION UNIT 1

COMPARISON OF ANALYTICAL RESULTS TO HALF FLOW WATER TEST
SODIUM THIOSULFATE STORAGE TANK DRAWDOWN

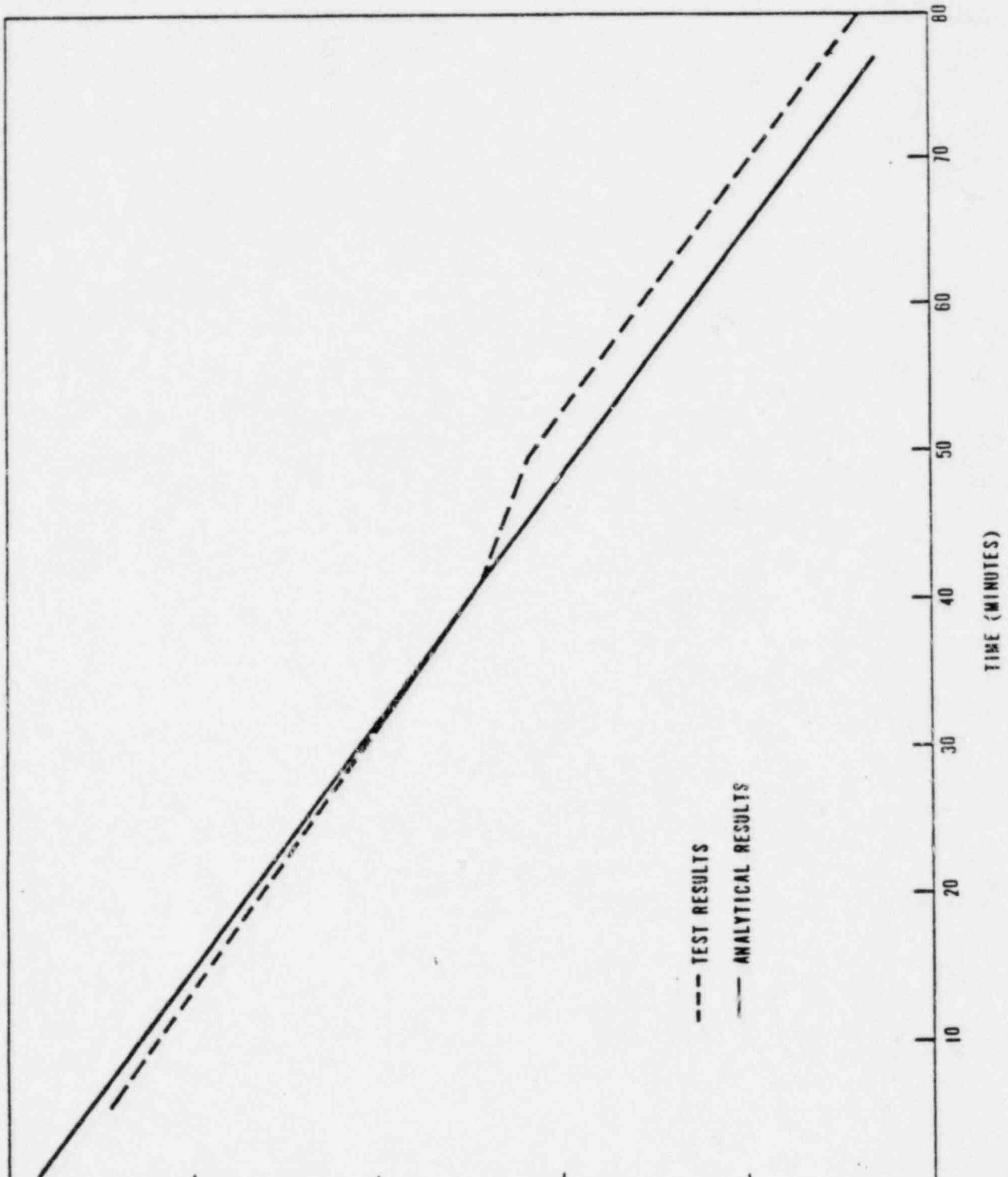
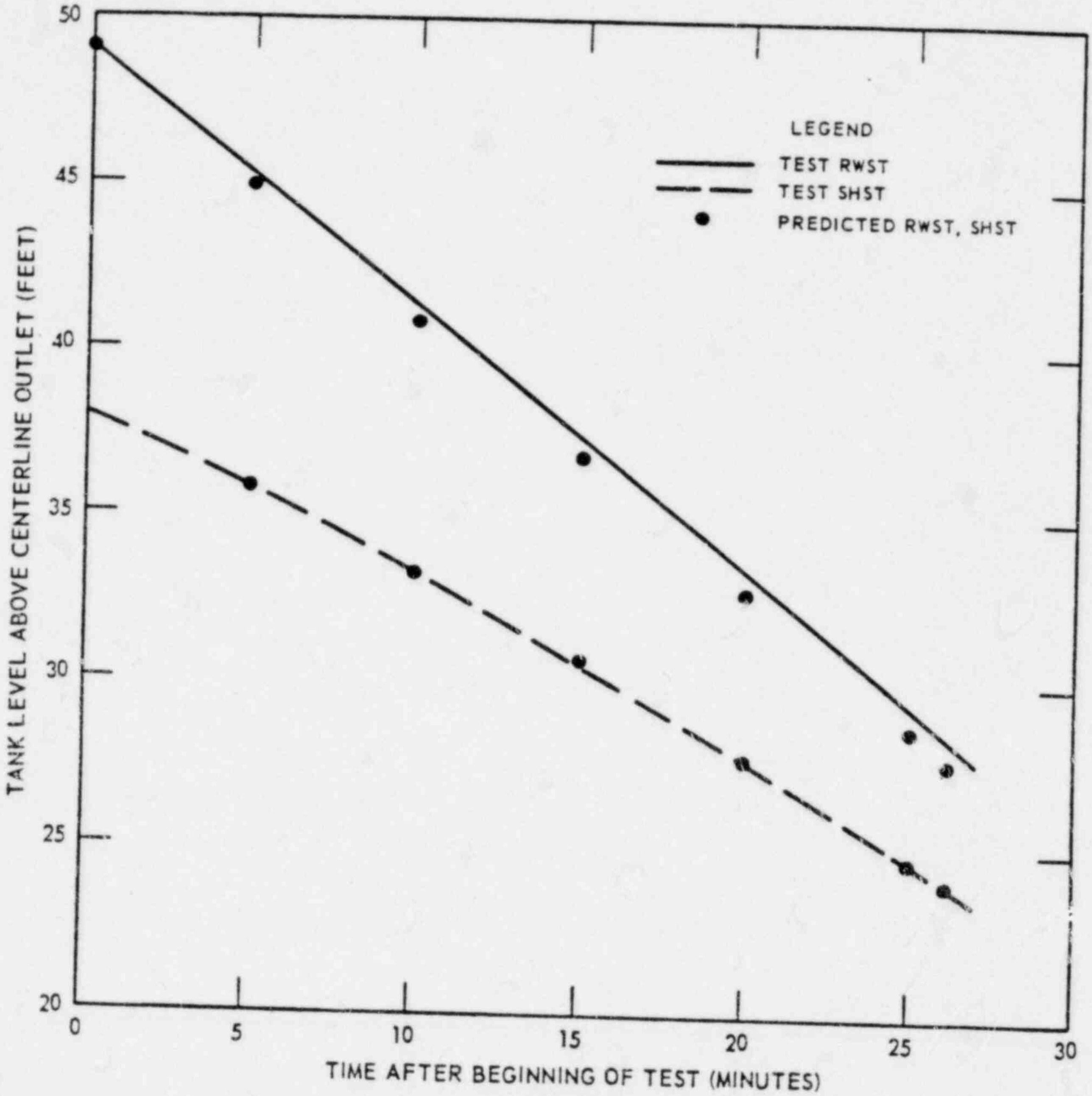



FIGURE F-9

THREE MILE ISLAND NUCLEAR STATION UNIT 1

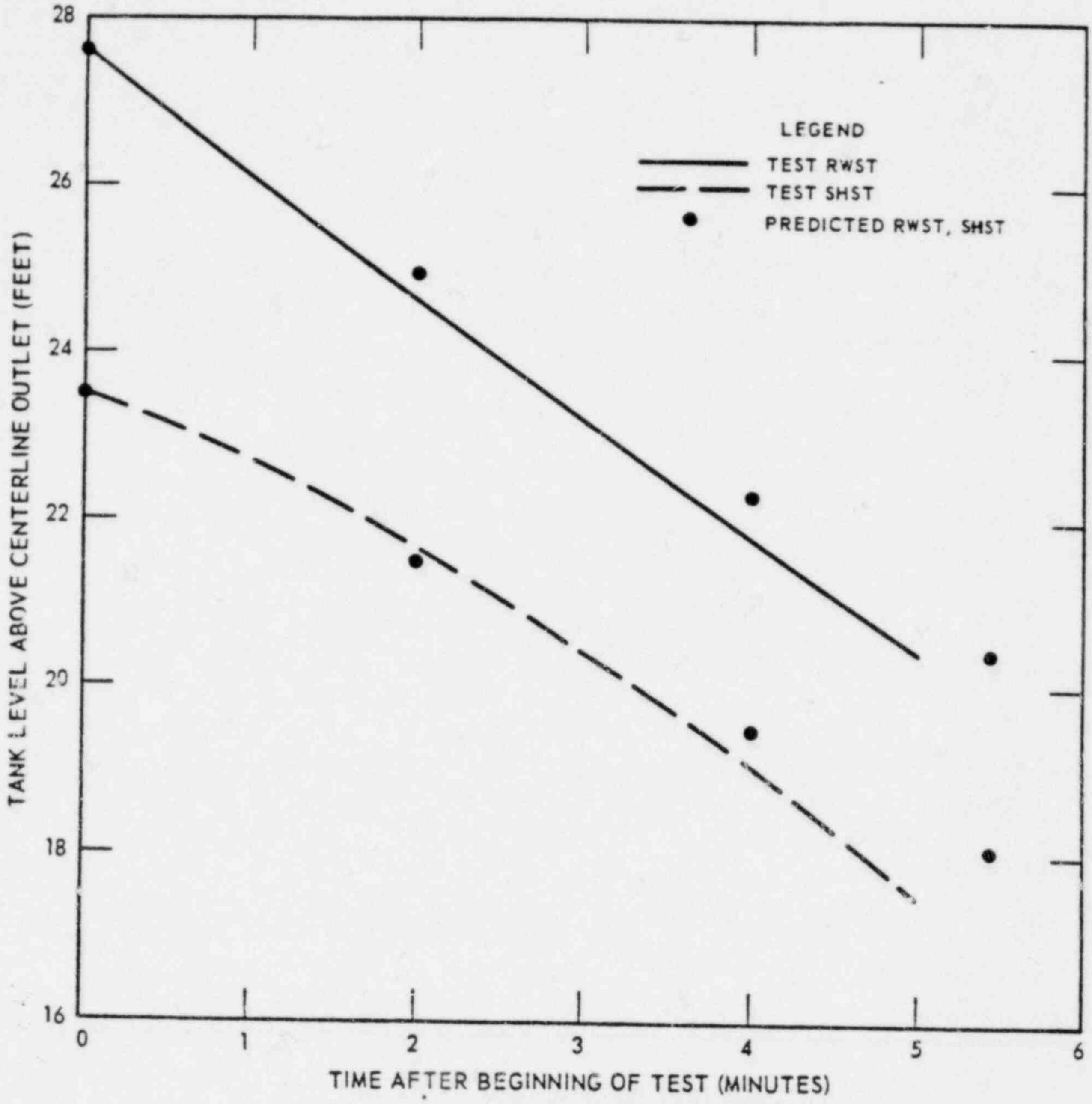
COMPARISON OF ANALYTICAL RESULTS TO HALF FLOW WATER TEST
SODIUM HYDROXIDE STORAGE TANK DRAWDOWN




AMENDMENT 19
JUNE, 1980

**SOUTH CAROLINA ELECTRIC & GAS CO.**
**VIRGIL C. SUMMER NUCLEAR STATION**

Analytical Prediction to Water Test
Case 1: Reactor Building Spray System Design Case
Figure 6.2-51cc



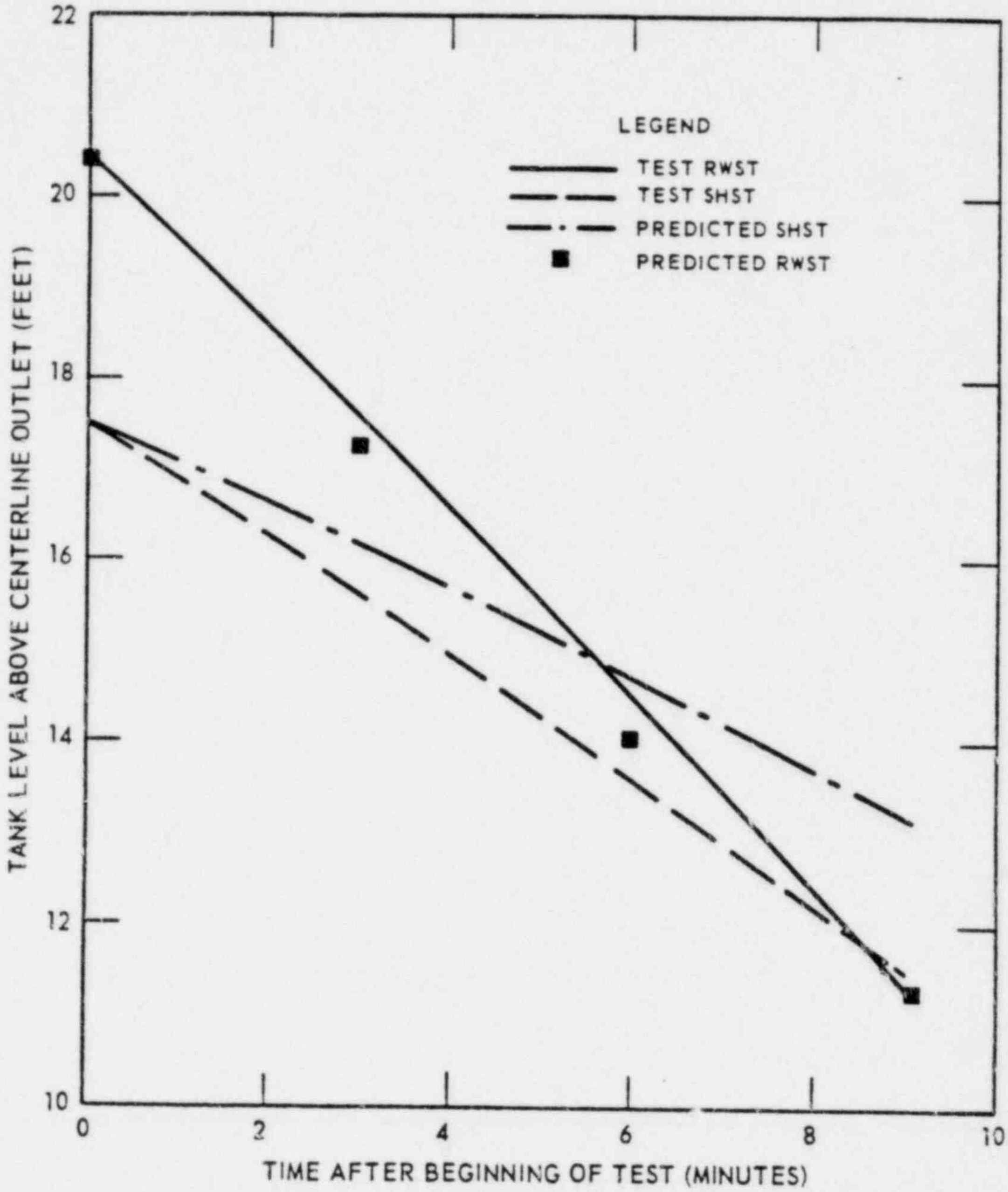

AMENDMENT 19
JUNE, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test

Case 2: Normal Case

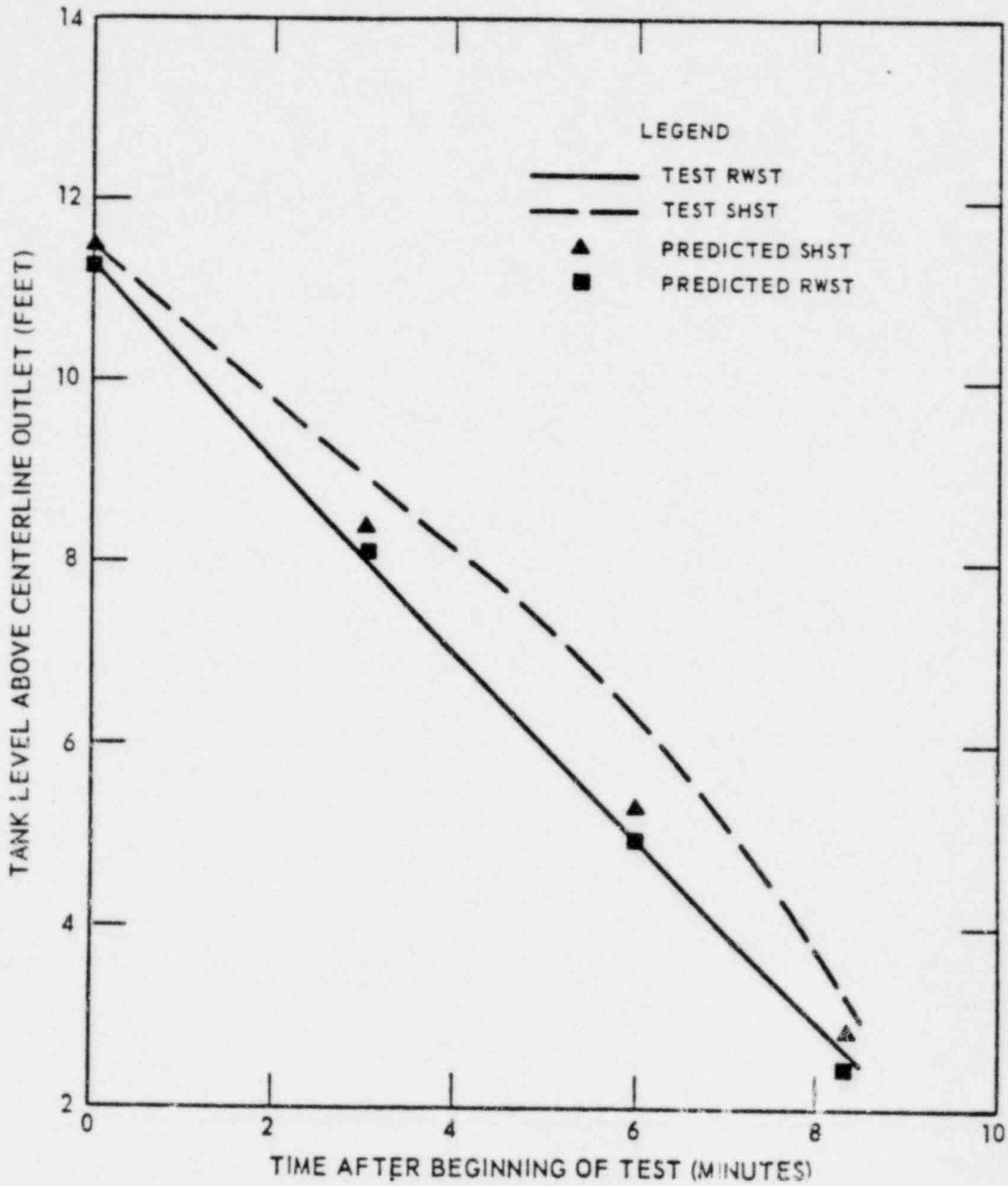
Figure 6.2-51dd




AMENDMENT 19
JUNE, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test
Case 3: Normal Case with One Reactor Building Spray Pump Inoperable
Figure 6.2-51ee




AMENDMENT 19
JUNE, 1980

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Analytical Prediction to Water Test
Case 4: Normal Case with One RHR Pump Inoperable

Figure 6.2-51ff