Docket No. 50-289
License No. DPR-50

THREE MILE ISLAND NUCLEAR STATION

UNIT 1

INITIAL STARTUP REPORT

METROPOLITAN EDISON COMPANY

SUBSIDIARY OF GENERAL PUBLIC

UTILITIES CORPORATION

50-289

11-27-74

PREPARED BY

GENERAL PUBLIC UTILITIES

SERVICE CORPORATION

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THREE MILE ISLAND NUCLEAR STATION
UNIT 1

INITIAL STARTUP REPORT

METROPOLITAN EDISON COMPANY
SUBSIDIARY OF GENERAL PUBLIC
UTILITIES CORPORATION

PREPARED BY

GENERAL PUBLIC UTILITIES
SERVICE CORPORATION

1414 002

METROPOLITAN EDISON COMPANY

THREE MILE ISLAND NUCLEAR STATION

UNIT I

Docket No. 50-289

License No. DPR-50

INITIAL STARTUP REPORT

November 26, 1974

Prepared by

General Public Utilities
Service Corporation

1414 003

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

Three Mile Island Nuclear Station Unit I was issued operating license DPR-50 on April 19, 1974 and the first fuel assembly was inserted into the core on April 20, 1974. Fuel loading was completed on April 25, 1974. Initial Criticality was achieved on June 5, 1974, upon completion of a Post Fuel Load Pre-Critical test program.

Zero Power Physics testing began on June 5, 1974 and was completed on June 10, 1974. The zero power measurements were performed at a Reactor Coolant System temperature of 532°F and a pressure of 2155 psi.

Initial escalation of the reactor above zero power commenced on June 15, 1974 and the turbine generator was initially loaded on June 19, 1974. Further increases in power level were made as testing was successfully completed at each of the four major power plateaus defined in the power escalation sequence. The four major power levels and the dates they were first attained are as follows:

Power Level	Date
15%	June 16, 1974
40%	June 29, 1974
76%	July 14, 1974
100%	August 3, 1974

The power escalation test program was completed with successful performance of the unit acceptance test on August 26, 1974 and on September 2, 1974, TMI Unit I was declared commercial. Figure 1.0-1 gives the power history of Unit I from Initial Fuel Loading to the completion of startup testing.

This report is submitted in accordance with Technical Specification 6.7.3 and summarizes startup test program results and unit performance from fuel loading through 100% full power operation as of 1200 on August 27, 1974. The integrated burnup on the core at this date and time was 26.3 effective full power days (EFPD). Figure 1.0-2 shows the integrated core burnup for core 1.

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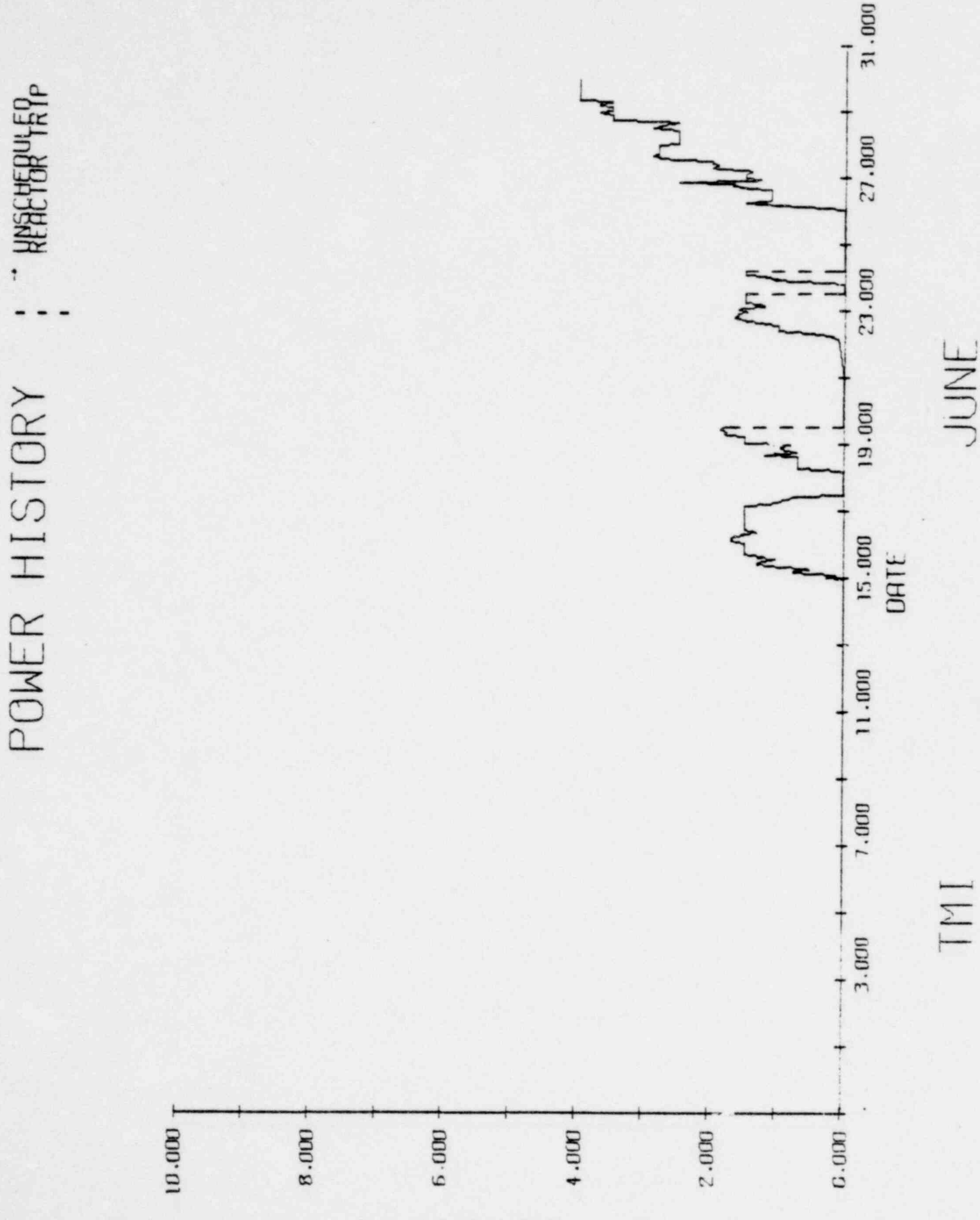

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FIGURE 1.0-1

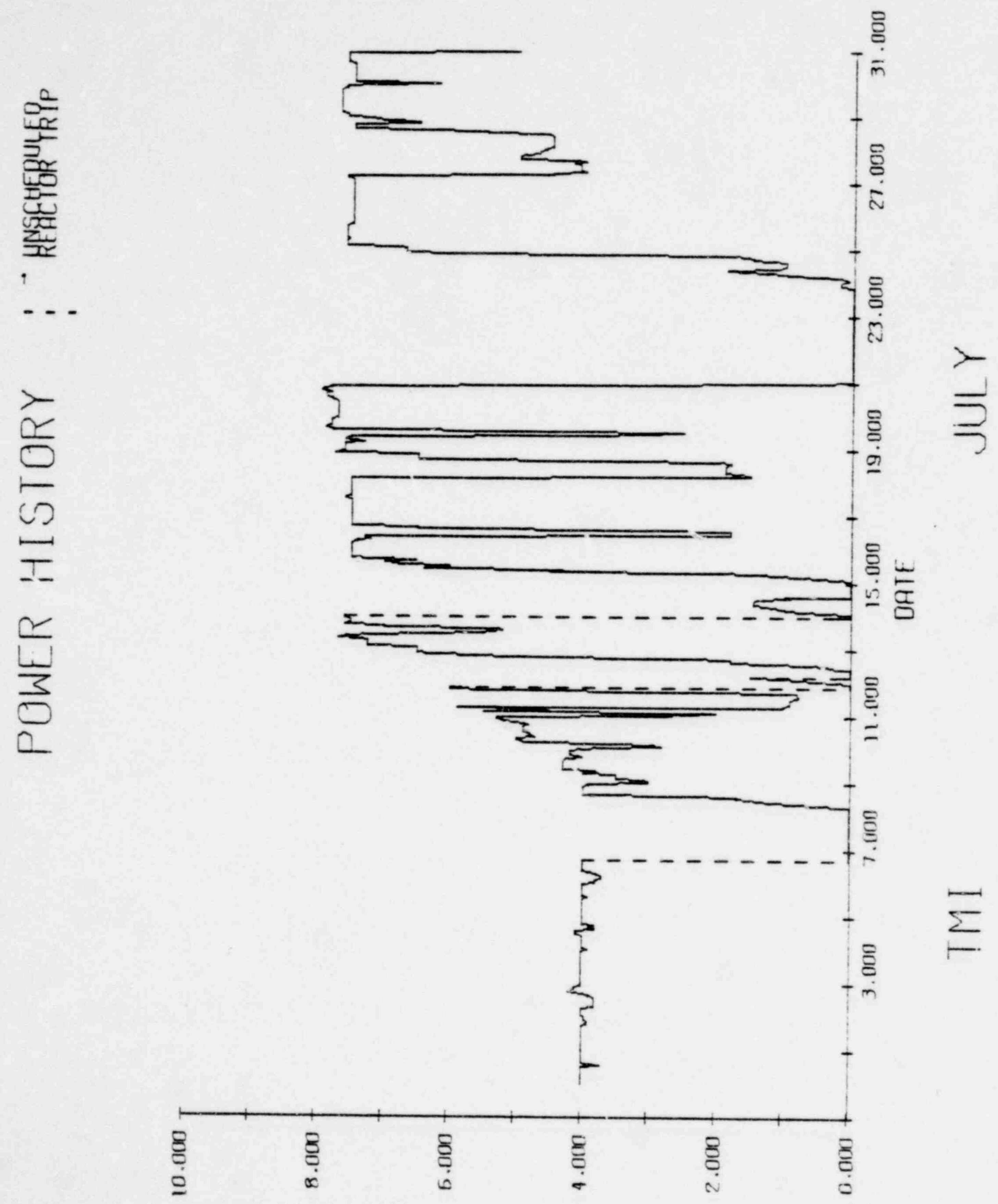

FIGURE 1.0-1 (Cont'd)

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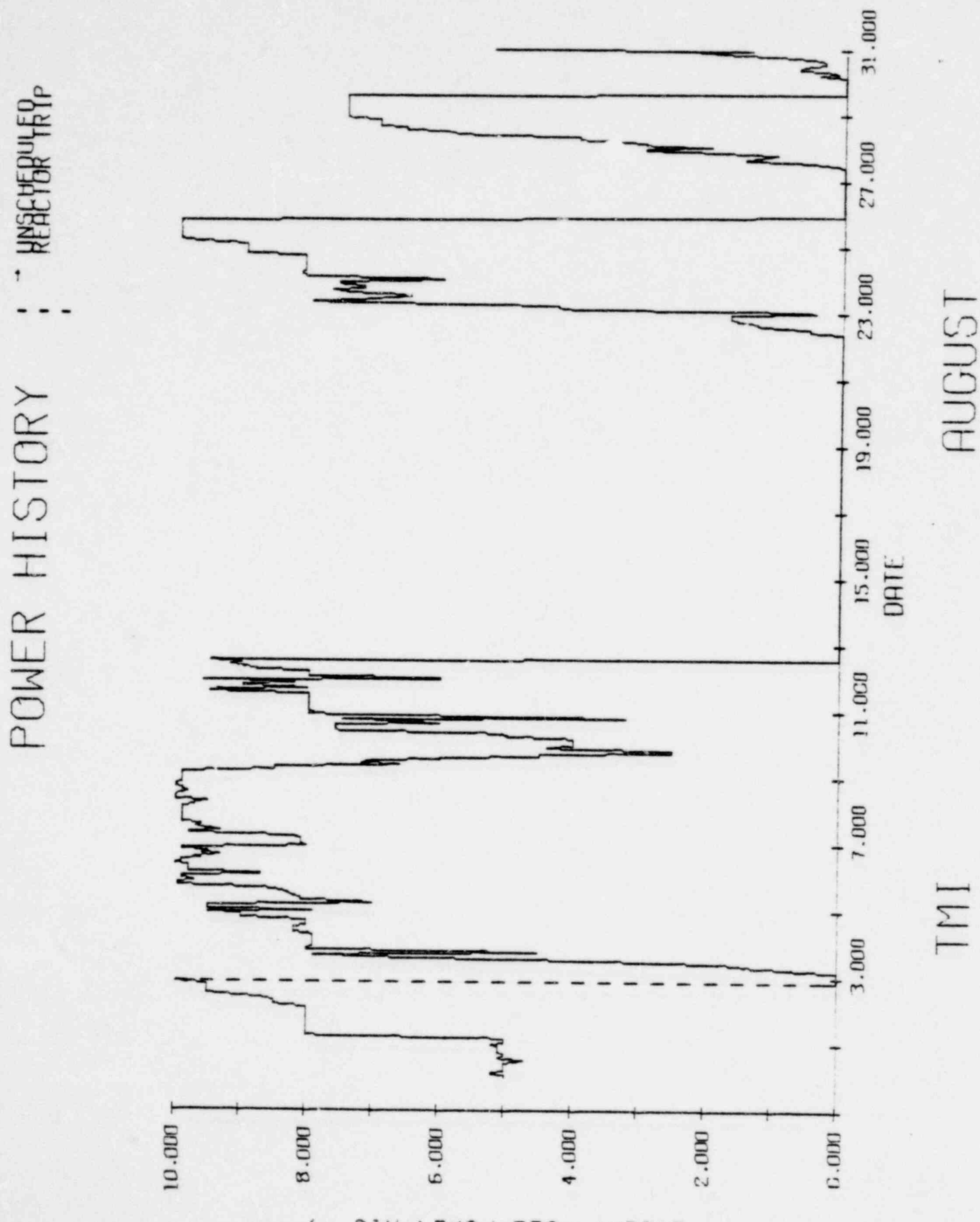

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FIGURE 1.0-1 (Cont'd)

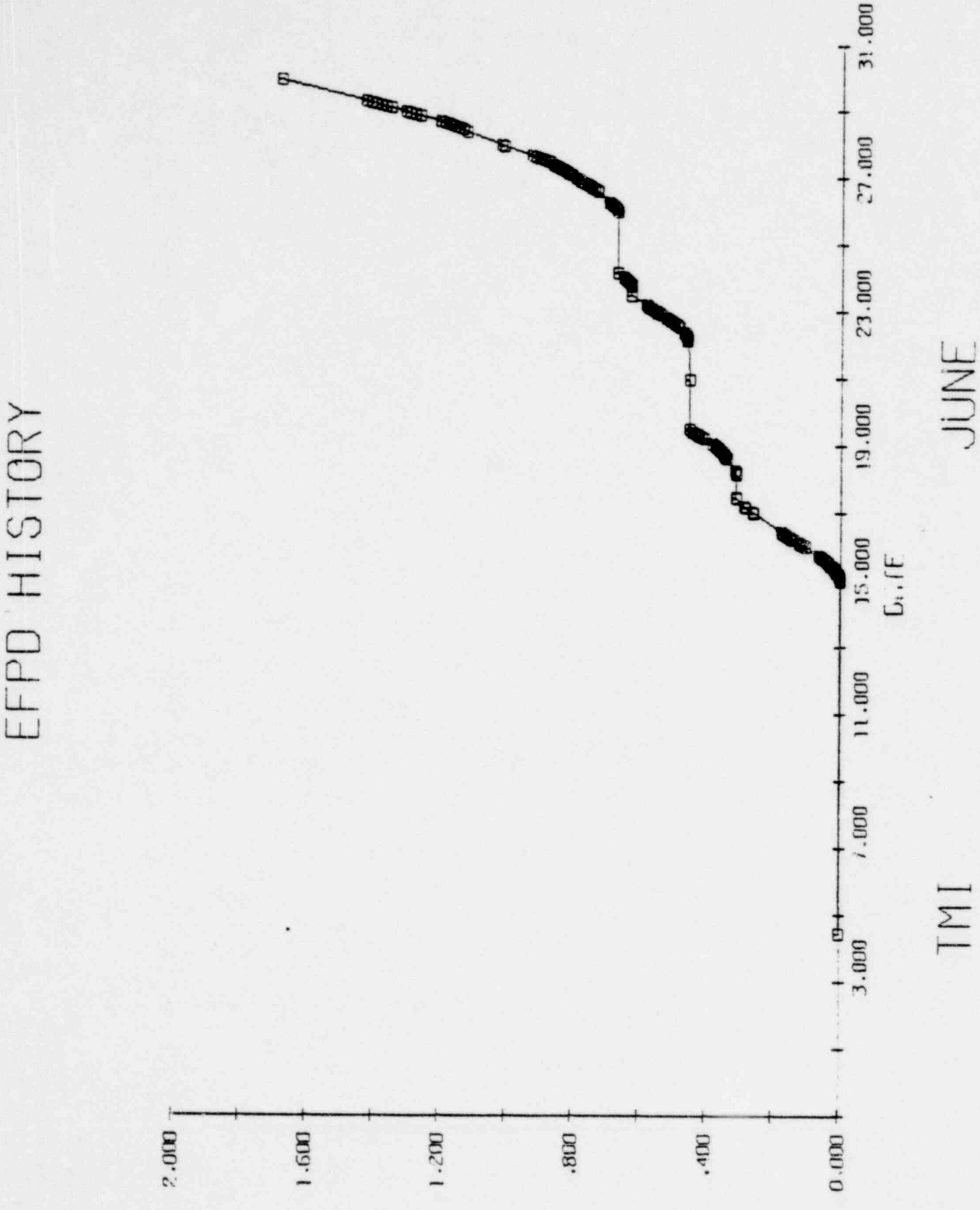

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FIGURE 1.0-2

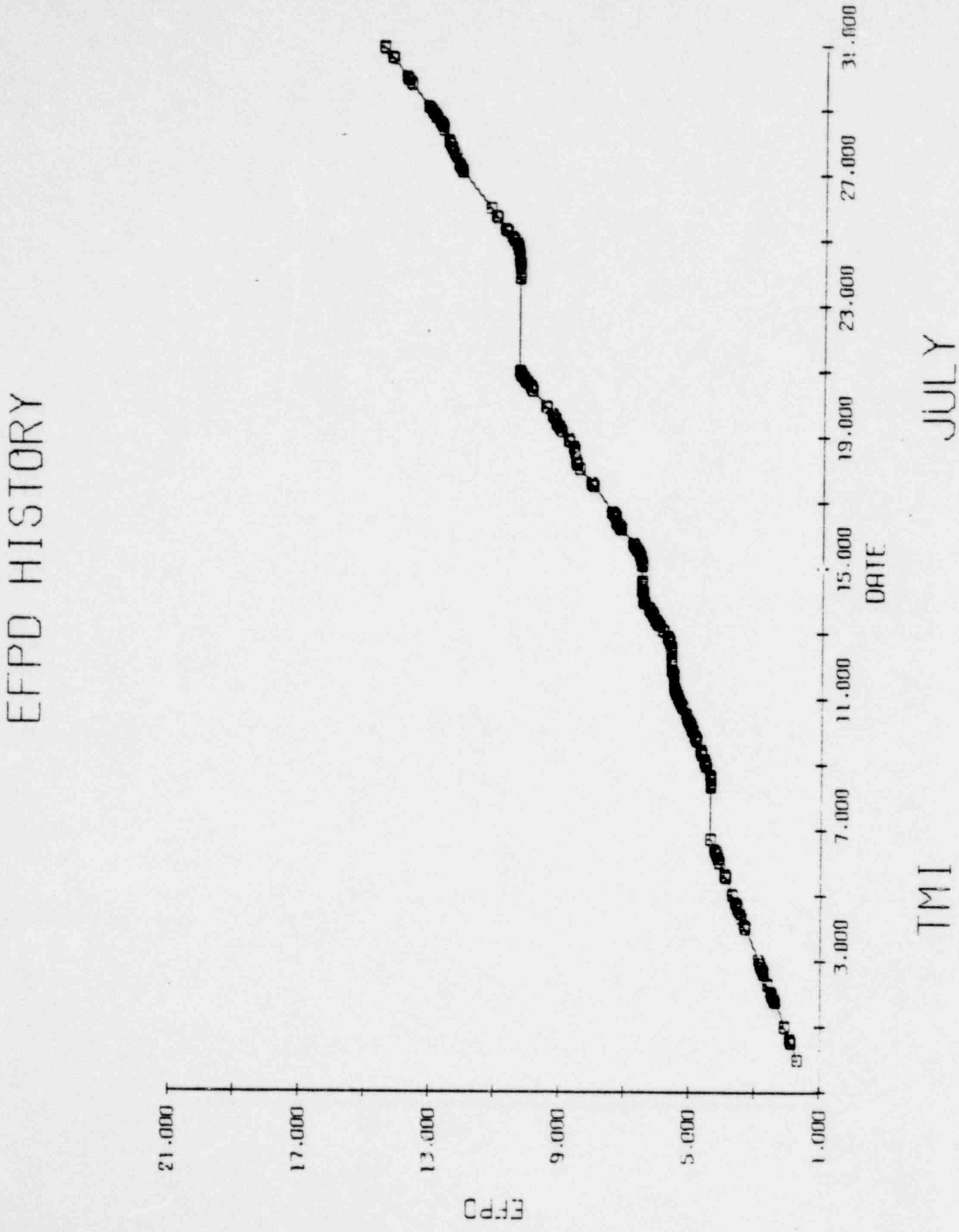

FIGURE 1.0-2 (Cont'd)

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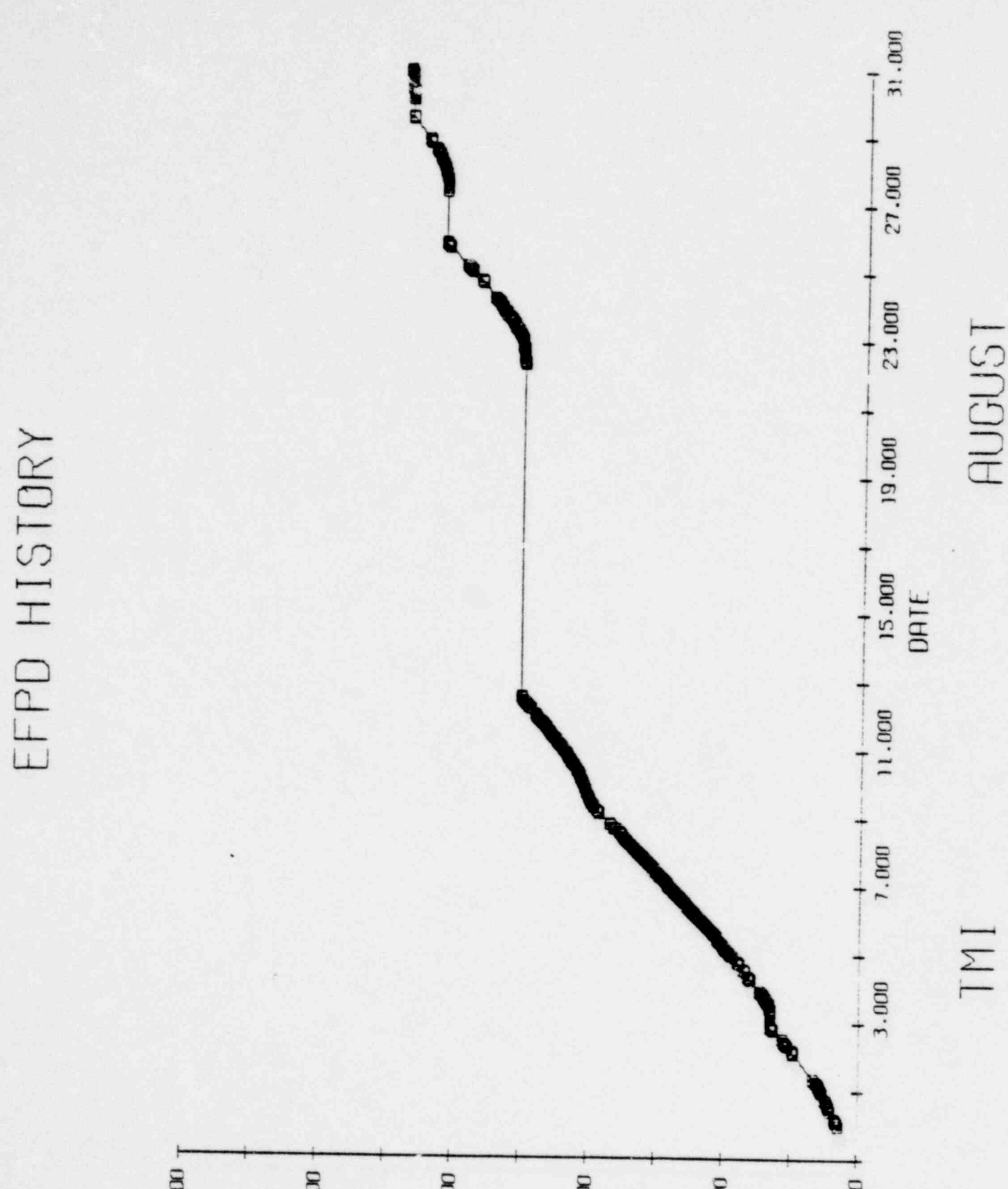

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FIGURE 1.0-2 (Cont'd)

1.2 SUMMARY

1.2.1 GENERAL

Three Mile Island Unit I commenced commercial operation on September 2, 1974 at a rated full power output of approximately 861 MWe (gross). The nuclear steam supply system was designed by the Babcock and Wilcox Company and was the third in a series of systems to be put into service. The tandem compound turbine generator was supplied by the General Electric Company.

The unit has been operated at power levels up to and including 100% FP since the completion of startup testing. In general, the performance of the unit has been satisfactory. Testing and operation of the nuclear steam supply system and the turbine-generator revealed a few conditions that were other than predicted and none which adversely affected plant safety. The problems encountered were of a nature that would be expected during the startup of a unit of this size. An evaluation of the unit startup and Power Escalation Test Program results concluded that the unit can be safely operated at full rated power.

A chronological log of the unit startup, beginning with post fuel load filling and venting of the Reactor Coolant System is given in Figure 1.2-1. A summary of the startup and power escalation test results addressed by the major sections of this report is given below.

1.2.2 INITIAL FUEL LOADING

Initial Fuel Loading began on April 20, 1974 and was completed on April 25, 1974. Loading of the core was accomplished under semi-dry conditions with the reactor vessel water level maintained six inches to three feet below the vessel flange. One major delay occurred during the loading sequence when power was lost to the main fuel handling bridge. Overall, Initial Fuel loading was completed in less than five days and was conducted in a safe and orderly manner.

1.2.3 POST FUEL LOAD PRECRITICAL TEST PROGRAM

Following Initial Fuel Loading and prior to Initial Criticality, a Post Fuel Load Pre-Critical Test Program was conducted from April 26, 1974 to June 4, 1974. During this period, the Reactor Coolant Pump Flow and Flow Coastdown Tests, Control Rod Drive Drop Time Test, Reactor Coolant System Leakage and Surveillance Procedure Verification Test and Pressurizer Operational Test were conducted. In all cases, the applicable Technical Specification requirements and test acceptance criteria were met. A brief summary of each test follows:

(a) Reactor Coolant Pump Flow Test

Reactor coolant (RC) flow measurements were conducted at 532°F, 2155 psi with the core installed and measured flow rates were within the range of maximum and minimum acceptable values for all RC pump combinations tested. RC flow with 4 pumps operating is 146.0×10^6 lbm/hr or 111.2% of the design flow rate.

(b) Reactor Coolant Pump Flow Coastdown Test

The reactor coolant flow coastdown characteristics were measured at system conditions of 532°F, 2155 psi with the core installed and met flow decay criteria for all RC pump combinations tested. RC flow decays to 66.5×10^6 lbm/hr in 10 sec when all 4 RC pumps are tripped.

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(c) Control Rod Drive Drop Time Test

Control rod drop time measurements were conducted at 150°F and 532°F under flow and no flow conditions with the core installed to ensure that the control rod assembly trip insertion times from 100% withdrawn to three-fourths insertion will not exceed 1.40 seconds under reactor coolant no flow conditions and 1.66 seconds with reactor coolant flow. All acceptance criteria limits were met.

(d) Pressurizer Operational Test

The pressurizer spray flow was set to 190.5 gpm and the spray bypass flow was set to 0.99 gpm. Both settings were well within the acceptance criteria limits of 190.0 +19/-6 gpm and 1.0 +0.5/-0.25 gpm, respectively.

(e) Reactor Coolant System Leakage Test and Surveillance Procedure Verification

Reactor Coolant System hot leakage measurements were conducted during the hot functional and post fuel load pre-critical test programs. Measured results verify that the unidentified reactor coolant leakage does not exceed the Technical Specification limit of 1.0 gpm and that the normal control instrumentation is sensitive enough to perform leak rate measurements. The value of normal evaporative losses, as used in Technical Specification 3.1.6.2, was established as 0.51 gpm.

1.2.4 CORE PERFORMANCE - MEASUREMENTS AT ZERO POWER

Core performance measurements were conducted during the Zero Power Test Program which began on June 5, 1974 and ended on June 10, 1974. This section presents a summary of the zero power measurements. In all cases, the applicable test and Technical Specifications limits were met.

(a) Initial Criticality

Initial criticality was achieved on June 5, 1974 at reactor conditions of 532°F and 2155 psig. Control rod groups 1 through 6 and 8 were withdrawn to 100% and group 7 was positioned at 75% withdrawn. Criticality was achieved by deborating the reactor coolant from 2086 ppm to 1545 ppm at an average deboration rate of 73 ppm per hour. Initial criticality was achieved in an orderly manner and good agreement was found between the measured and predicted critical boron concentrations of 1609 ppm and 1625 ppm, respectively.

(b) Nuclear Instrumentation Overlap

At least two decades overlap was measured between the source and intermediate ranges and the linearity and absolute output of the intermediate and source range detectors were within specifications.

(c) Reactivity Calculations

An on-line functional check of the reactimeter[1] was performed after initial criticality. Reactivity calculated by the reactimeter was within ±2% of the core reactivity determined from doubling time measurements.

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(1) A discussion of the reactimeter is given in section 4.3.

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(d) All Rods Out Critical Boron Concentration

The measured all rods out critical boron concentration of 1617 ppm was in excellent agreement with the predicted value of 1634 ppm.

(e) Temperature Coefficient Measurements

The measured temperature coefficients of reactivity at 532°F, zero power were well within the acceptance criteria limit of $\pm 0.4 \times 10^{-4} \Delta k/k/^{\circ}F$ over the range of boron concentrations that the measurements were made.

(f) Soluble Poison Worth

The measured results for the soluble poison differential worth at 532°F were within 1.25% of the predicted values.

(g) Control Rod Group Worth Measurements

The measured results for the differential and integral control rod group reactivity worths conducted at zero power, 532°F using the boron/rod swap and rod drop techniques were in good agreement with predicted worths. The maximum deviation between measured and predicted worths was 8.33%.

(h) Ejected Control Rod Worth

Two methods were used to measure the pseudo ejected control rod worth at zero power, 532°F. The results from the boron-swap and rod drop techniques were in good agreement. The best estimate for the measured result was 0.688%Δk/k from the boron swap method and the Technical Specification limit of 1.0%Δk/k was not exceeded.

(i) Shutdown Margin

Minimum shutdown margin verification and stuck control rod worth measurements were completed at the zero power, 532°F condition. The measured value of the most reactive control rod stuck out of the core with all other control rods inserted was 3.84%Δk/k. The shutdown margin available under this condition was at least 1.8% Δk/k, which ensures that the Technical Specification limit is satisfied.

1.2.5 CORE PERFORMANCE - MEASUREMENTS AT POWER

This section presents a summary of the physics measurements that were conducted with the reactor at power. Testing was conducted at the four major power plateaus of 15%, 40%, 76% and 100% of 2535 megawatts thermal core power, as determined from primary and secondary calorimetric measurements. Operation in the power range began on June 15, 1974.

Periodic measurements and calibrations were performed on the plant nuclear instrumentation during the escalation to full power. The four power range detector channels were calibrated based upon primary and secondary plant heat balance measurements. Testing of the incore nuclear instrumentation was performed to ensure that all detectors were functioning properly and that the detector outputs were processed correctly by the plant computer. Core axial imbalance determined from the incore instrumentation system was used to calibrate the out of core detector imbalance indication. Radiation surveys of the biological shield and reactor and auxiliary buildings were conducted to obtain base line data on accessible work areas while the reactor is operating at power.

The major physics measurements performed during power escalation consisted of determining the moderator and power Doppler coefficients of reactivity, determining the worth and associated power distributions effected by simulated dropped and ejected control rods, and obtaining detailed radial and axial core power distribution measurements for several core axial imbalances. Values of minimum DNBR and maximum linear heat rate were monitored throughout the test program to ensure that core thermal limits would not be exceeded.

(a) Biological Shield Survey

Radiation levels in all accessible locations of the plant adjacent to the biological shield were measured. The maximum radiation levels found in all accessible areas were below 100 mRem/hr. and the biological shield meets all design criteria.

(b) Nuclear Instrumentation Calibration at Power

The power range channels were calibrated as required during the startup program to indicate within two percent of the total core power as determined by a primary or secondary plant heat balance. These calibrations were required due to power level, boron and/or control rod configuration changes during testing. The acceptance criteria were met in all instances.

(c) Incore Detector Testing

Tests conducted on the incore detector system demonstrated that all detectors were functioning as expected, that symmetrical detector readings agreed within acceptable limits and that the plant computer applied the correct background, length and depletion correction factors.

(d) Power Imbalance Detector Correlation

The results of the Axial Power Shaping Rod (APSR) scans performed at 40% and 76% FP show that an acceptable incore versus out-of-core offset relationship is obtained by using a gain factor of 4.033 in the power range scaled difference amplifiers. The measured values of minimum DNBR and maximum linear heat rate for various axial core imbalances indicate that the Reactor Protection Trip Setpoints provide adequate protection to the core. Imbalance calculations using the backup recorder provided a reliable alternate to computer calculated values.

(e) Rod Reactivity Worth Measurements

Differential control rod reactivity worth measurements were performed in conjunction with the reactivity coefficients and pseudo ejected rod tests. The measured rod worths agreed with the design values well within the acceptance criteria limits of $\pm 20\%$.

(f) Reactivity Coefficients at Power

The measured results at 40%, 76% and 100% FP indicate that the moderator coefficient of reactivity will be negative during operation above 95% FP. The power doppler coefficient measurements indicate that the least negative value is $-0.00710\%\Delta k/k/\%$ FP at full power conditions.

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(g) Dropped Control Rod Test

The dropped control rod test performed at 40% and 76% FP met all required acceptance criteria and the following conclusions were drawn as a result of the measurements:

(1) The core power distributions and thermal conditions that developed from a control rod dropped into the core during operation at power showed adequate margins to minimum DNBR and maximum linear heat rate limits. The measured worth of the dropped rod was 0.094%$\Delta k/k$.

(2) Quadrant power tilt calculations using the backup recorder were accurate in comparison with the computer calculated values.

(3) The rod drive control system properly responded to an asymmetric rod condition.

(h) Pseudo Ejected Control Rod Test

The measured worth of the pseudo ejected control rod by the rod swap technique was 0.278%$\Delta k/k$, which is well below the Technical Specification limit of 0.65%$\Delta k/k$. The measured values of maximum linear heat rate and minimum DNBR were 13.12 kw/ft and 4.85, respectively, with the ejected rod at 100% withdrawn. The maximum measured radial power peak was 2.38 in the fuel assembly containing the ejected rod. Substantial margins were observed to core thermal limits in a pseudo ejected rod accident.

(i) Core Power Distributions

Core power distribution measurements were conducted at 15%, 40%, 76% and 100% full power under steady state equilibrium xenon conditions for specified control rod configurations. Comparison of the measured distributions with the PDQ-07[1] results shows good agreement. For the three cases studied at 40%, 76% and 100% full power, the three largest measured and calculated radial peaks were chosen. In each case, the measured values were within 8% of the calculated results.

The results of the minimum DNBR and maximum LHR analyses are given in Table 5.9-6. The margins to the minimum DNBR limit of 1.55 and the maximum LHR value of 17.1 kw/ft were 109% and 42%, respectively, after extrapolation to 112% FP. All quadrant power tilts and axial core imbalances measured during the power distribution tests were within the Technical Specification and normal operational limits.

(j) Nuclear Steam Supply System Heat Balance

Good agreement was found between hand and computer calculated heat balances during power escalation. Preliminary calculations of total reactor coolant flow based upon heat balance results indicate a flow rate of 108.6% of design at 100% FP.

(k) Reactivity Depletion Versus Burnup

The measured critical boron concentration at 22.0 EFPD and 100% FP conditions was within 30 ppm of the predicted result and well within the acceptance criteria limit of 86 ppm.

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(1) PDQ-07 is the analytical model used by Babcock and Wilcox for core design studies.

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(1) Neutron Noise Measurements

Neutron noise data was recorded on the TMI Unit I Core during the startup test program to serve as baseline data for future periodic measurements. Initial analysis of the data indicates no major differences from the expected neutron noise signatures.

1.2.6 NUCLEAR STEAM SYSTEM PERFORMANCE

A summary of the testing performed during power operation to monitor the performance of the nuclear steam system is presented below. The test results presented include reactor coolant system steady state and transient operation, reactor coolant pump performance, radioactive waste management and primary and secondary system water chemistry.

(a) Reactor Coolant System Performance

Steady state operation of the reactor coolant system and the steam generators was monitored at various power levels during the escalation to 100% FP. The average values for reactor coolant inlet, outlet and average temperature; steam generator pressure, temperature and level; and feedwater flow and temperature followed the expected response with power. The response of the reactor coolant system to major unit transients has been satisfactory. One area that is under study is the low pressurizer level reached during a reactor trip. The reactor coolant pumps have performed well and produce flows in excess of their design values. Reactor coolant system leakage was maintained within the limits specified in the Technical Specifications.

(b) Auxiliary System Performance

Radioactive wastes generated during power operation consist of liquid, gaseous and solid wastes. The wastes generated during the startup program were adequately processed, stored and/or disposed using plant and off-site facilities in accordance with the plant Technical Specifications. Primary and secondary system water chemistry have been maintained within the limits allowable for operation at power. Radiochemistry analyisis of reactor coolant activity indicated that no fission product releases occurred during the startup test program.

1.2.7 BALANCE OF PLANT TESTING

This section presents a summary of the results of balance of plant testing, adjustments and operation at power. Balance of plant systems consist mainly of the turbine generator, main steam, turbine bypass, atmospheric dump, condensate, feedwater, moisture separator, steam extraction and feedwater heating, heater drain, emergency feedwater, and cooling water systems. The cooling water systems consist of the circulating water, natural draft cooling tower, intermediate cooling water, nuclear service closed cooling water, nuclear service river water, secondary service closed cooling water, secondary service river water and mechanical draft cooling tower systems.

A summary of the testing performed on the above systems is given below and includes the test results of the Turbine Generator Operational Test, the Turbine Bypass System Test and Main Steam Safety Valve Operation, Feedwater System Operation and Testing, Emergency Feed System Operation and Testing and Power Escalation Checkpoints.

(a) Turbine Generator Operational Testing

Turbine generator (TG) performance was very satisfactory throughout the startup test program. Approximately 7 days of testing time were lost due to unscheduled turbine trips and turbine related problems. Water ingestion into the turbine through the 4B heater extraction line due to the isolation valve failure to close on high shell-side level caused by ruptured tubes was the only unanticipated startup problem which could have led to major damage and delays; however, subsequent turbine operation indicates that the turbine suffered no damage. #3 bearing vibration is approximately 0.5 mils higher than acceptable for long term operation; balancing operations will be performed at the first convenient outage. TG output at 2535 MWt is 861 MWe, when conservatively corrected to design vacuum conditions and compares well with a guaranteed vlaue of 837 MWe. Steam conditions are 10,621,000 #/hr at 592°F compared with design of 11,158,286 #/hr at 559°F. Due to the increased amount of superheat over design, the turbine operates at less than "valves wide open" conditions. Gross heat rate is 9993 Btu/KWhr compared with design of 10,002 Btu/KWhr.

(b) Turbine Bypass System and Main Steam Safety Valve Operation

Acceptable response of the turbine bypass valves in maintaining turbine header pressure setpoint and response to small changes in setpoint at reactor powers ≥15% was attained after the difference between steam generator pressures was included in the control system and a wiring reversal error was corrected. Peak to peak oscillations are ±6 psi.

The turbine bypass valves, along with the main steam safety valves, function adequately to limit main steam pressure during turbine trips to ≤1100 psia. The longest valve opening time was 2.1 seconds; peak steam pressure following the 100% generator/turbine trip was 1082 psia.

Operation of the atmospheric dump valves was not required to limit main steam pressure to ≤1100 psia during the loss of offsite power test.

Final settings of the main steam safety valves appear adequate for continued plant operation; however, safety valve operation is one of several areas under study in an attempt to optimize plant response to major transients.

(c) Feedwater System Operation and Testing

The condensate, feedwater, moisture separator, heater drain, feedwater heating, and steam extraction systems function acceptably to support steady state and transient operation at 100% power. Oscillations and transient response associated with these systems are acceptable; however, investigations are continuing in several areas in an effort to further optimize plant response. These areas are:

(1) Heater drain pump discharge valve control.

(2) Heater drain pump recirculation valve control.

(3) Ability of the feedwater pumps to supply feedwater to the steam generators when turbine header pressure increases rapidly.

(4) Ability of the feedwater control valves to respond to changes in valve differential pressure.

(d) Emergency Feedwater System Operation and Testing

With the amount of decay heat present during performance of the loss of offsite power test, the turbine driven emergency feedwater pump provided more than adequate flow to control RCS temperature and pressure. The EFW valves had to be throttled to keep from exceeding RCS cooldown limits as steam generator levels began increasing to 95% on their operating range level indication. A setpoint of 50% instead of 95% will be used to adequately remove decay heat without exceeding cooldown rate limitations.

(e) Power Escalation Checkpoints

The secondary service closed cooling water adequately cooled its heat loads; SSCCW heat exchanger discharge temperatures were well below their design limit of 95°F at 100% power.

The mechanical draft cooling tower effluent temperature and differential temperature between influent and effluent had to be controlled manually because the automatic controller was inoperative. Acceptable operation could be obtained with continuous surveillance; however, until operators gained familiarity with tower operation, differential temperature limits were violated several times. Final testing of the cooling towers will be conducted at a later date.

The natural draft cooling towers were performance tested under Summer conditions. Capacity was determined as 104.1% of design. April and December performance tests will be conducted at a later date.

Powdex effluent chemistry analysis demonstrates acceptable capability to clean up the condensate system during 100% power operation.

1.2.8 UNIT PERFORMANCE

A summary is presented in this section of the tests performed during and after escalation to 100% FP which measure the overall performance of the unit under normal operating, transient and emergency conditions. A summary of unit response to planned and unplanned major load changes is presented in the section on Unit Transient Response. The Loss of Offsite Power and Shutdown From Outside the Control Room tests demonstrated the ability to safely control the unit under emergency conditions. The Unit Acceptance Test verified that the nuclear steam supply system can operate in accordance with the warranted design specifications. In all instances, safe operation of the unit was demonstrated and the applicable Technical Specifications requirements were met.

(a) Unit Transient Response

Transient testing of the unit was conducted at specified ramp rates in the turbine following, reactor/steam generator following and fully integrated modes of control at 40%, 76% and 100% of full power. The Integrated Control System (ICS) successfully maneuvered the plant in all three modes of control during the 40% and 76% FP tests. In the 100%-50%-100% transient, the 10%/min design ramp rate was accomplished within acceptable limits in the fully integrated ICS mode. The transient was completed at 8%/min in the turbine following mode and at 6%/min on the decrease and 4%/min on the increase in the steam generator/reactor following mode. The ability of the ICS to control the plant during a loss of reactor coolant pump, a loss of a main feedwater pump and a dropped control rod transient was satisfactory.

The reactor was successfully runback to 15% FP during the turbine trips from 76% FP. The reactor tripped on high RC pressure after a full load rejection at 100% FP.

(b) Loss of Offsite Power

The station emergency blackout procedure was verified for use during a blackout and all required automatic action occurred as expected. RCS temperature and pressure remained well within their respective limits and no increase in fission product activity occurred as a result of this test.

(c) Shutdown From Outside The Control Room

The reactor plant can be maintained in a hot shutdown condition from locations outside of the main control room by a normal shift compliment. The alternate control center contains sufficient instrumentation and communications to permit satisfactory monitoring and direction of shutdown operations.

(d) Unit Acceptance Test

The Three Mile Island Unit I nuclear steam supply system produces 2552.615 MWt gross energy output compared with the warranty value of 2449 MWt. Main steam temperature is 591.6°F compared with the warranty of 569°F. These results indicate a substantial margin of NSS performance above warranty specifications.

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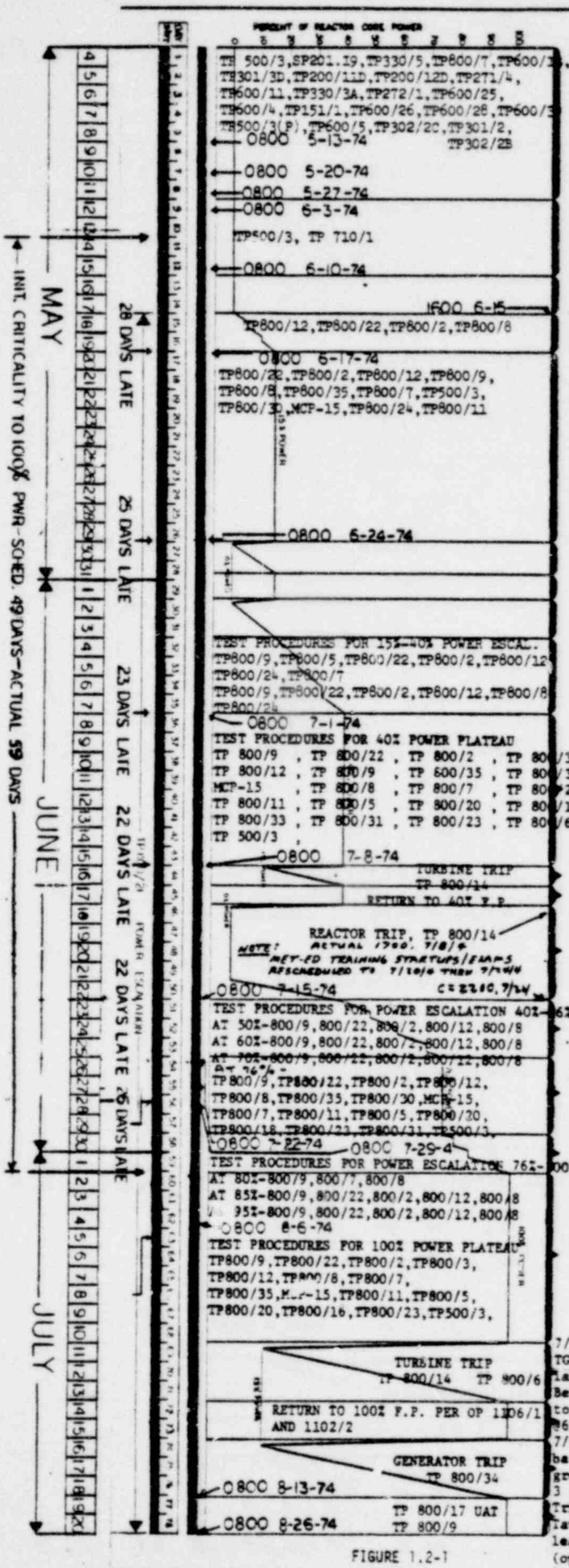


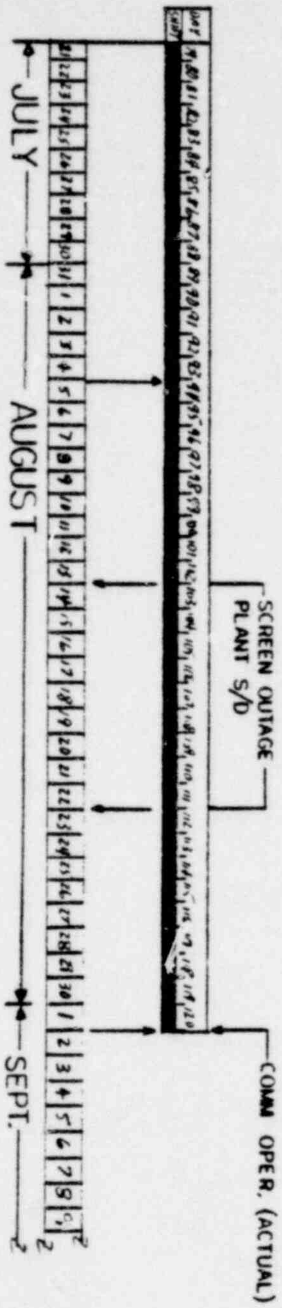
FIGURE 1.2-1

5/4 - Compl. RCS Fill
5/5 - Vented RCS
5/6 - RCS @45#-Support Sys. Vlv. Align.
5/7 - Bubble in PZR-Press to 310#
5/8 - Ran RCS 1A,1C for venting
5/9 - Compl. CRD venting.
5/10- CRD Testing (Start)
5/12-CRD Testing-330/3A compl.
5/13-CRD No Flow Cold Test -RCP-1D Oil Cooler Leak
5/14-RCP-1C Oil Lk.-Upp. Insp. Plt.-Rod#29,GR.2-Stuck on Bot.
5/15-Re-vented CRD's
5/16-RCS Temp. > 200°F
5/17-Continuing H/u w 2-RCP's; RCP problems
5/18-Ran 1A RCP-cont. H/u
5/19-Plant Testing @ >500°F-RCS
5/20-CRD Hot Testing-No Flow
5/21-Ran TG interface Test-RCP-1D lube oil leak repairs
5/22-Incr. Res. Temp. to 532°F. ran RCP Flow/Coastdown (part.) A-RCP 005 Mtr. insp. port covers
5/23-Ran RCP B,C,&D comb.-ran EF-P2A after repair-MS rel. vlv. setting
5/24-RCP Flow/Coastdown in progress-RC-R -1B flg. leak(PZR side) ST. depress & cooldown for repr.
5/25-Repair attempt of RC-RV-1B failed-temp. @ 400°
5/26-Compl. RCS Cooldown & Depress. Removed RC-RV-1B-CRDM's & discon.
5/27-Repl. Gasket in RC-RV-1B-Retorq. PZR. Code Reliefs-Repl. O Ring in incore #27
5/28-EF-P1 (EFW Pump) malfunct.-repr. in progress-fill & vent RCS: incr. press. to 330 psig.
5/29-Compl. RCS & CRD Venting; incr. temp. to 240°F.
5/30-Temp. to 532°F-oil leaks on RCP's A,C,&D-cooled down to 390°F
5/31-Compl. Oil leak repairs on RCP's & leak on MU-P1B recirc. orifice
6/1 -CRD Testing compl/Bal. RCP's Flow/No Flow Test compl.
6/2 -RC-V24 leak being repaired; unbal. of ∅ currents on CRD
6/3 -RC-V2 leak-depress to ≅600psig & repair leak in MU recirc. ln. repaired.
6/4 -RC-V2 stuck cl.-fixed & incr. press. to 2285# for OP. Hydro per SP.-Bal RC-P-1C/CRD SP's compl.
6/5 -Blown seal in MU-P-1C secured for repair/set PZR bypass vlv. initial crit. occurred at 2236.
6/6 -Continued ZPPT sequence
6/7 -Shutdown 0600-1200-sec. plnt. leaks.
6/8 -ZPPT Section 9.7-9.11 compl.
6/9 -ZPPT Section 9.12-9.15 compl.
6/10-Completed ZPP Testing-Plant T/O to MEC for training startups
6/11-MEC training continuing-A MU pump started inadvertently due to trouble shooting of DC ground in ESAS cab.(part ES)-Pump ran for ≅1 min w/o suct. flow (Damaged rotating assy./Reactor shutdown & training secured.
6/12-RPS problems on NI5&7 wkg./ MU-P1C flow curve sat./MU-P1A wkg. repairs.
6/13-MU-P1C started and ran for ≅26 secs. w/o flow (suct. vlv. MU-V72C shut)-pump suffered internal damage.
6/14-Maint. work items to P.Z.T. continuing-MU-P1C/MS-V3D repair/NI5-CH.A/small stm leak on stm. chest FW-P1B. MU-P1A returned to service ≅2300.
6/15-Noise in "A" OTSG cleared-reac. crit. @0215-resumed MEC training startups & completed. MU-P1C repaired and in service. (All MU pumps avail.)
6/16-P.Z.T. in progress (TP 800/21)-Rx. pwr. to 15%-TG to 1800 RPM-TG core monitor inoperable-unable to energize Hold.New CM due 6/17-dropped #2 rod-group #7-blown fuse.

6/17-Mn.Turb. tripped ≈0340(Hi exhaust temp-uneven dumping) New CM inst.(Lost 2 shifts) Hi wtr. leakage on RCP B - PC158 -TG tripped twice at sp.
6/18-Hi TG #7 brg.vib.&high exh. hood temp.& pwr. reduced to 7%-TG to speed.-Low IA alarms on TC&AB caused turb. bypass valves to fail shut;FW-V16A/B failed shut - Loss of flow to "C" cond. booster pump. -turb &Rx. tripped on no FW to the SG reset IA-V1 to 75psig.
6/19Rx.crit. at 0402-TG on line @ 1529-TG tripped due to high lvl. in "C" moist.sep @1604-Gen. bkrs did not trip-wiring problem in xcl pnl. corrected. RCS leak rate≈7gpm via RC-V1 to RCDT
6/20-TG on line ≅60mw blown rupt. disc in RCDT = Rx.trip due to low RCS press. @1220
6/21-RCS shutdown
6/22-MS-V3D will not open
6/23-Rx. crit. at 0130-ran SOP on MS-V3's (except 3D)-SP-10-A-PT2 & SP-10-B-PT1 not wired correctly-Rx. at 15%-preps for shutdown from outside CR in progress.
6/24-Shutdown from O/S CR-TP800/36 Returned to I.C. Op.
6/25-Completed Loss of O/S Pwr. TP800/32-Ret. to I.C.-completed Mec.Train-Perf.CODE REL.VLV. INSP. W/VDR.-RC-V1 excessive leakage to Hot Shutdown for repair.
6/26-RC-V1 leakoff Mod.-Rx.ret.to I.C.
6/27-Rx.to 15%-TG on line for Turb. overspeed trip test.
6/28-Rx.to 22%-TG on line >80mwe-RC-V2 stuck shut-Rod#1,GR6 dropped due to blown fuse in CRD. cab-Rx to 35% pwr.
6/29-Unit Load Steady State Test compl. at 40%
6/30-Rx.@40% awaiting xenon equilb. High Iron in heaters,cleaning up w/Powdex.
7/1 -TG Op cks. compl. @40%/compl. ht. bal./st. react. coeff - 800/5 & 800/20/1.5 to 2. GPM lk. to RCDT. fr. pzr. rel. vlvs.
7/2 -40% cont. 800/5 & 20/FW-P1B lk. repaired/Bio. survey 800/3 compl./800/35 eff. monit.compl/
7/3-40% RC-V2 stuck shut/computer prob.-info on incore imbal. 800/18 lost ≈3 shifts/B MU pmp. returned to serv./FW testing compl. @40% - 800/7/
7/4 -40% MS-V3E leaks by/pwr. imbal. compl. - 800/18 repeat at 76% - modify/
7/5 -40% pseudo rod eject - 800/33 compl./5.5 GPM RCS leak traced to S.L.O. on RC-V3 and/or 31-backseated to correct
7/6-40% Estab. boron conc. for rod testing - 800/31
7/7 -Observed xenon oscillation - 24 hrs. B&W recomm. - Rx trip at 1700 [SCHED. TRIP]
7/8 -Rx.Crit.@0530/Wkg.Rx.Maint. Repairs RC-V2 oper.
7/9 -RC-V2 repair compl.-Rx.pwr. incr. to 40%-TG on line
7/10-40% Rx.pwr-TG on line Tuning ICS for ULTT/Resetting MS Rel. Vlvs.
7/11-Rx. to 50% for Ht. Bal. ULTT & NI Cal./resetting MS Rel.Vlvs.
7/12-C=ULTT @ 50%/Rx.to 60%,TG on line-unsched. turb. trip @ 0709 recovered to 62%-had 2nd Turb. Trip-Rx.trip followed.
7/13-Rx.to 15%-Loaded TG-Plant tripped on variable LP oscillation Bet/B/P vlvs.&Turb./Rx to 65%-TG loaded/C=ULTT @65%
7/14-Rx.to 76%-Rx.Runback to 55%-Loss of group top light for GR 3 returned to 76%-Rx. Trip @2352 while taking Tave data in Ics.clip leads shorted together (oper.error)/4th htr. iso.vlv.failed to shut-backed up to turb. and flooded mo. sep.

THREE MILE ISLAND NUCLEAR POWER STATION
UNIT NO 1
POWER ESCALATION TEST PROGRAM
PREPARED
REVIEWED
APPROVED
REV 1

1414 025



PET PROGRAM HISTORY (CON'T)

7/15 - GE 24 hr. restriction to Roll TG/Tube leak repair in 4th stage htr. in prog./MEC compl. S/U's (Training).

7/16 - Comm. TG Roll @0530/Rx. to 76% GE CIV Test Sat.

7/17 - EHC lk. @CV#3, forced S/D - compl. 30% TG Trip @1100/EHC lk. repaired - returned to 76%.

7/18 - Rx. @76% - Awaiting 3-D Equilb. zenon.

7/19 - EHC fluid lk. @0320 - tripped TG - Rx. to 15% for EHC lk. repair/wkg. MS hgrs./Rx. to 65% 5 hr fuel soak/to 76%.

7/20 - TG trip @1030 during TG over-speed trip test - TG controls malfunction/recovered @1130 - Rx. to 80% - TG on line/set M.S. Reliefs (RV-1 thru 6).

7/21 - Rx. @76%-/Plant S/D @2300 for MEC AEC exams and S/U's.

7/22 - Rx @10^{-8} amps (IR) - Coll. data for B&W on zenon transient/made insp. of Rx Bldg. - cleaned up oil under RC-P1C & 1D/working outage items.

7/23 thru 7/24 - Rx shutdown @0300 for AEC exams/ Outage work items in progress.

7/25 - Minor delay on S/U due to O_2 problem - "A" cond. drn. vlv. was left open during maint./tripped turbine due to GE testing from 1800 RPM/excessive vib. on #11 bearing (7.5 mils) - Turb. off line for repair/Rx to 70% pwr. @2130 - compl. 5-hr. soak period.

7/26 - Rx. to 76% @0300 TG on line - await. 3D Zenon equilib.

7/27 - Rx @76% - TG on line/compl. rod worth & reactivity coeff.

7/28 - Reduction in FW Flow due to low level in Hotwell - reduced pwr. to 40% - "C" cond. boost. pump mech. bound up - holding pwr. at 40-50% for repair CO-P2C.

7/29 - Rx. at 46% - Waiting CO-P2C repair/incr. to 76%-6th stage HD tk. vlv. cont. vlvs. caused ICS oscillations - runback occurred - Ret. to 76%.

7/30 - Rx. at 76% - Attained zenon Equilib. & "O" imbalance; also NI adjust./started pwr. imbal.

7/31 - Rx. at 55% due to asymetric rod runback (8-8) - blown fuse/ret. to 76% - computer failure/"A" condenser side tube leak - chemistry problems forced pwr.reduct. to 50%.

8/1 - Rx. at 50% - leaks in "A" Cond. Repaired (loss ≈24 Hrs.)

8/2 - Rx. to 80% - Estab. zenon equilib.

8/3 - Rx. to 85% - compl. req'd testing/ Rx. to 95% - compl. req'd testing/ Rx. at 100% @2010/Rx. trip @2113 due to flux/imbal./flow on CH. A. CH. C @ ≈103% pwr. trip should have been ≥106% by B&W revised flow summer cal.

8/4 - Rx. crit. at 0700/Rx. to 80% @1600 - Await. zenon equilib./ Reduced to ≈45% (2 hrs) to change oil in CO-P2C (wtr. in oil)/Rx. to 80% - hold for zenon equilib.

8/5-Rx. Pwr. to 100%

8/6-Reduced pwr. to ≈75%-oil leak at B Fw. pump resulted in sm. fire/returned to 100% pwr.

8/7-Decr. Rx. to 90%-change powdex usl./returned to 100%-Rx. runback to 55% due to loss of Gr.3 out limit light on pnl. ret. to 80% pwr.

8/8-Rx. at 98.7%

8/9-Rx. at 100% pwr.-for TP 800/5 & 20.

8/10-Rx. at 70%-at 43% pwr.-reduced to 25% returned to 40%.

8/11-Rx. at 98%-reduced to 60%-incr. to 81.5%

8/12-Rx. at 80%-transient testing in progress.

8/13-Rx. at 95%/TP 800/34-TG trip-Rx. tripped on hi RC press/plant cooldown/ Plnt. at hot S/D.

8/14-Plnt in cooldown 260°F 0345PSI/ commenced CIV screen rem. outage plnt S/D/RC press @80 psig/temp.145°F.

8/19-Sec. plnt. wtr. chem. cleanup in progress/making preps for S/U.

8/20-Plant S/U delayed by seal inj. filter leakage - Crouse wkg. to repair/CRD & RCS venting in progress/comm. H/U to 525°

8/22-Incr. temp. to 343°/press. to 1500psig

8/23-Res. temp. to 532°F/Press. to 2155psig Rx. crit @0850-incr. pwr. to 13%.

8/24-Rx. to 80% pwr.

8/25-Rx. to 90% pwr.

8/26-Rx. @100% pwr. @0345/commenced U.A.L. @1000, compl. @1700/Rx. shutdown to repair misc. leaks/depr. to 1600psig, temp. to 400°F.

8/27-Repaired leaks & comm. preps. for return to pwr.

8/28-1030-comm. Rx. S/U/Rx. crit. @1047/ 1629-T. trip on ov.-speed & exh. hood temp. hi/reset turb. & syn./to 30% pwr. 2200 - FW pmp. strn. leak-reduced pwr. to 20% pwr.

8/29-0106-incr. pwr. to 40%
0920-Rx. to 60%-500 MW
1200-incr. to 70% pwr.

8/30-1310 T. trip on #11 bear. vib./1312 Rx. trip on press/temp.

8/31-0135-comm. Rx. S/U
0720-Rx. @3%
1840-Syn. sm. turb. to 1800 RPM & incr. Rx. pwr. to 40%.

9/1-0013-Rx. @70%
0904-Rx. @80%

9/2-0001-TMI 1 commercial 0001/Rx. @100% pwr. @1100

NOTES:

1 -	START POST FUEL LOAD PRE-CRITICAL TESTING THRU COMMERCIAL OPERATION.	SCHED. 78 DAYS ACTUAL 120 DAYS
2 -	INIT. CRIT. TO 100% POWER OPER.	SCHED. 49 DAYS ACTUAL 59 DAYS
3 -	START 100% POWER TESTING THRU COMMERCIAL OPERATION.	SCHED. 19 DAYS ACTUAL 29 DAYS (9 OF THESE 29 WERE ATTRIBUTED TO THE TURB. PLANT SCREEN OUTAGE).

FIGURE 1.2-1 (Cont'd)

2.0 INITIAL FUEL LOADING

Fuel loading was initiated with the insertion of fuel assembly 1C10 into core location 14-N on April 20, 1974 and was completed with the insertion of fuel assembly 1C40 into core location 15-F on April 25, 1974. Figure 2.0-1 presents a detailed map of the final core configuration, listing each fuel, burnable poison, control rod and orifice rod assembly location. Table 2.0-1 provides a detailed sequence of events for initial fuel loading of TMI Unit I.

Neutron countrate was monitored during the core assembly on four seperate detector channels, with a minimum of two (2) channels operating whenever core geometry was changed. In addition to the permanently installed source range channels, NI 1 and 2, two (2) temporary incore BF_3 proportional counters were used. Independent plots of inverse neutron countrate versus the number of fuel assemblies loaded were maintained from the outputs of these detectors to ensure that the core remained subcritical at all times.

Initial fuel loading at TMI Unit I was a semi-dry operation with the reactor vessel water level maintained six inches to three feet below the vessel flange. The semi-dry loading improved visibility of the fuel assemblies during manipulation and provided accessability to the vessel flange area when repositioning the incore detectors. Radiation levels were not overly restrictive due to the lower water levels in the fuel transfer and spent fuel pool canals. The maximum radiation level measured was 25mr/hr (β-γ) at the fuel handling bridge during transfer of the fuel assemblies with the sources.

Several minor problems and delays were encountered during fuel loading. A description of the problems and their resolution is given below:

(a) Hydraulic pressure on the west transfer carriage upender was lost several times due to a loose coupling between the hydraulic pump and the motor. The coupling was retightened periodically to permit use of the upender.

(b) On the initial attempt to load fuel assembly 1B01, interference occurred with an adjacent assembly. The fuel handling bridge was re-indexed and 1B01 was inserted with no further problem.

(c) During insertion of fuel assembly 1B16, the tube down light on the main fuel handling bridge failed to actuate at 2100 lbm and the low-load cut-out was obtained at 600 lbm. Seating of the assembly was verified visually and the 2100 lbm low-load interlock was bypassed to ungrapple the assembly at the 600 lbm cut-out. The interlock was readjusted and no subsequent problems were encountered.

(d) While inserting fuel assembly 1C24, the 2100 lbm low load cut-out was obtained with the assembly three inches from the down position. The fuel handling bridge was reindexed and the fuel assembly was then inserted smoothly.

(e) Fuel assemblies 1B18 and 1B02 would not seat properly on the initial attempts to insert them. After the assemblies were re-grappled, both were loaded without any further problem.

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(f) The major delay during the fuel load sequence occurred when power was lost to the main fuel handling bridge. The power loss resulted from shorting two leads in the main power cable which became frayed due to insufficient support as the cable moved with the bridge. The total delay was 12 hours.

In spite of the minor delays, initial fuel loading at TMI Unit I went smoothly and was conducted in a safe manner. Figure 2.0-2 depicts number of fuel assemblies loaded versus time.

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FUEL ASSEMBLY LOADING SEQUENCE

CRA - Control Rod Assembly
ORA - Orifice Rod Assembly
BPRA - Burnable Poison Rod Assembly
APSRA - Axial Power Shaping Rod Assembly
DET( )- Auxiliary Neutron Detector

STEP NO.	ASSEMBLY TYPE	ID#	FEATURE	ID #	CORE LOCATION	ACTION
1	Support		DET A		14-H	Insert
2	Support		DET B		10-P	Insert
3	Fuel	1C10	ORA (Neutron Source)	026	14-N	Insert
4	Fuel	1C22	ORA (Neutron Source)	005	2-D	Insert
5	Fuel	1C37			13-O	Insert
6	Fuel	1C28	BPRA	B28	13-N	Insert
7	Fuel	1C26	BPRA	B26	12-O	Insert
8	Fuel	1C53			14-M	Insert
9	Fuel	1A14	CRA	C10	12-N	Insert
10	Fuel	1A02	CRA	C09	13-M	Insert
11	Fuel	1B55	BPRA	B68	12-M	Insert
12	Fuel	1B27	BPRA	B71	13-L	Insert
13	Fuel	1B57	BPRA	B55	11-N	Insert
14	Fuel	1A49	APSRA	A03	12-L	Insert
15	Fuel	1A17	CRA	C29	11-M	Insert
16	Fuel	1B23	BPRA	B56	11-L	Insert
17	Fuel	1B24	BPRA	B21	12-K	Insert
18	Fuel	1B26	BPRA	B54	10-M	Insert
19	Fuel	1A01	CRA	C43	10-L	Insert
20	Fuel	1A30	CRA	C42	11-K	Insert
21	Fuel	1B06	BPRA	B04	10-K	Insert
22	Fuel	1B59	BPRA	B03	9-L	Insert
23	Fuel	1B17	BPRA	B46	11-H	Insert

TABLE 2.0-1

STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
24	Fuel	1A21	APSRA	A04	10-N	Insert
25	Fuel	1A36	CRA	C44	9-M	Insert
26	Fuel	1B3B	BPRA	B22	9-N	Insert
27	Fuel	1B08	BPRA	B23	8-M	Insert
28-1	Support		DET B		10-P	Remove
28-2	Support		DET B		7-M	Insert
29	Fuel	1A04	CRA	C11	11-O	Insert
30	Fuel	1B33	BPRA	B27	10-O	Insert
31	Fuel	1C21	CRA	018	12-P	Insert
32	Fuel	1C42			11-P	Insert
33	Fuel	1A46	CRA	C56	9-K	Insert
34	Fuel	1A29	CRA	C57	8-L	Insert
35	Fuel	1A15	CRA	C55	10-H	Insert
36	Fuel	1B11	BPRA	B07	8-K	Insert
37	Fuel	1B01	BPRA	B05	9-H	Insert
38	Fuel	1B35	CRA	C61	8-H	Insert
39	Fuel	1B20	BPRA	B07	7-L	Insert
40	Fuel	1B60	BPRA	B18	10-G	Insert
41	Fuel	1A26	CRA	C58	7-K	Insert
42	Fuel	1A06	CRA	C54	9-G	Insert
43	Fuel	1B54	BPRA	B08	7-H	Insert
44	Fuel	1B12	BPRA	B09	8-G	Insert
45	Fuel	1A10	CRA	C60	7-G	Insert
46	Fuel	1B52	BPRA	B10	6-K	Insert
47	Fuel	1B19	BPRA	B16	9-F	Insert
48	Fuel	1A56	CRA	C59	6-H	Insert
49	Fuel	1A27	CRA	C53	8-F	Insert

TABLE 2.0-1 (cont'd)

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STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
50-1	Support		DET A		14-H	Remove
50-2	Support		DET A		9-E	Insert
51	Fuel	1B61	BPRA	B13	6-G	Insert
52	Fuel	1B45	BPRA	B11	7-F	Insert
53	Fuel	1A44	CRA	C51	6-F	Insert
54	Fuel	1B49	BPRA	B37	5-H	Insert
55	Fuel	1B04	BPRA	B30	3-E	Insert
56	Fuel	1A39	CRA	C50	5-G	Insert
57	Fuel	1A51	CRA	C52	7-E	Insert
58	Fuel	1B28	BPRA	B63	5-F	Insert
59	Fuel	1B03	BPRA	B61	6-E	Insert
60	Fuel	1A54	CRA	C35	5-E	Insert
61	Fuel	1B50	BPRA	B41	4-G	Insert
62	Fuel	1B40	BPRA	B29	7-D	Insert
63	Fuel	1A11	APSRA	A07	4-F	Insert
64	Fuel	1A22	APSRA	A08	6-D	Insert
65	Fuel	1B37	BPRA	B64	4-E	Insert
66	Fuel	1B07	BPRA	B62	5-D	Insert
67	Fuel	1A35	CRA	C22	4-D	Insert
68	Fuel	1B31	BPRA	B43	3-F	Insert
69	Fuel	1B43	BPRA	B33	6-C	Insert
70	Fuel	1A13	CRA	C21	3-E	Insert
71	Fuel	1C33	BPRA	B44	3-D	Insert
72	Fuel	1C36			2-E	Insert
73	Fuel	1A42	CRA	C23	5-C	Insert
74	Fuel	1C31	BPRA	B40	4-C	Insert
75	Fuel	1C46			3-C	Insert
76	Fuel	1C41	CRA	C38	4-B	Insert

TABLE 2.0-1 (cont'd)

. . . .

STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
77	Fuel	1C23			5-B	Insert
78	Fuel	1A07	CRA	C47	6-L	Insert
79	Fuel	1A28	CRA	C48	5-K	Insert
80	Fuel	1B09	BPRA	B59	5-L	Insert
81	Fuel	1A19	CRA	C49	4-H	Insert
82	Fuel	1B13	BPRA	B36	4-K	Insert
83	Fuel	1A09	APSRA	A06	4-L	Insert
84	Fuel	1A37	CRA	C34	3-G	Insert
85	Fuel	1B53	BPRA	B52	3-H	Insert
86	Fuel	1A47	CRA	C33	3-K	Insert
87	Fuel	1B25	BPRA	B42	3-L	Insert
88	Fuel	1C01	CRA	C20	2-F	Insert
89	Fuel	1B16	BPRA	B14	2-G	Insert
90	Fuel	1C54	CRA	C19	2-H	Insert
91	Fuel	1B14	BPRA	B15	2-K	Insert
92	Fuel	1C50	CRA	C18	2-L	Insert
93	Fuel	1C49			1-F	Insert
94	Fuel	1C17			1-G	Insert
95	Fuel	1C29			1-H	Insert
96	Fuel	1C16			1-K	Insert
97-1	Support		DET B		7-M	Remove
97-2	Support		DET B		1-L	Insert
98	Fuel	1C56	CRA	C12	10-P	Insert
99	Fuel	1C24			10-R	Insert
100	Fuel	1A41	CRA	C30	9-O	Insert
101	Fuel	1B47	BPRA	B02	9-P	Insert
102	Fuel	1C58			9-R	Insert

TABLE 2.0-1 (cont'd)

STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
103	Fuel	1A52	CRA	C45	8-N	Insert
104	Fuel	1B41	BPRA	B24	8-O	Insert
105	Fuel	1C03	CRA	C13	8-P	Insert
106	Fuel	1C45			8-R	Insert
107	Fuel	1A45	CRA	C46	7-M	Insert
108	Fuel	1B51	BPRA	B32	7-N	Insert
109	Fuel	1A38	CRA	C31	7-O	Insert
110	Fuel	1B18	BPRA	B06	7-P	Insert
111	Fuel	1C06			7-R	Insert
112	Fuel	1B02	BPRA	B57	6-M	Insert
113	Fuel	1A12	APSRA	A05	6-N	Insert
114	Fuel	1B36	BPRA	B38	6-O	Insert
115	Fuel	1C60	CRA	C14	6-P	Insert
116	Fuel	1C20			6-R	Insert
117	Fuel	1A55	CRA	C32	5-M	Insert
118	Fuel	1B22	BPRA	B58	5-N	Insert
119	Fuel	1A32	CRA	C15	5-O	Insert
120	Fuel	1C25			5-P	Insert
121	Fuel	1B05	BPRA	B60	4-M	Insert
122	Fuel	1A18	CRA	C16	4-N	Insert
123	Fuel	1C47	BPRA	B34	4-O	Insert
124	Fuel	1A25	CRA	C17	3-M	Insert
125	Fuel	1C34	BRRA	B35	3-N	Insert
126	Fuel	1C35			3-O	Insert
127	Fuel	1C52			2-M	Insert
128	Fuel	1C51			2-N	Insert
129	Fuel	1A43	CRA	C39	10-F	Insert
130	Fuel	1A48	CRA	C40	11-G	Insert

TABLE 2.0-1 (cont'd)

STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
131	Fuel	1B10	BPRA	B65	11-F	Insert
132	Fuel	1A23	CRA	C41	12-H	Insert
133	Fuel	1B21	BPRA	B47	12-G	Insert
134	Fuel	1A50	APSRA	A02	12-F	Insert
135	Fuel	1A31	CRA	C28	13-K	Insert
136	Fuel	1B44	BPRA	B49	13-H	Insert
137	Fuel	1A05	CRA	C27	13-G	Insert
138	Fuel	1B30	BPRA	B50	13-F	Insert
139	Fuel	1C57	CRA	C08	14-L	Insert
140	Fuel	1B34	BPRA	B20	14-K	Insert
141	Fuel	1C13	CRA	C07	14-H	Insert
142	Fuel	1B15	BPRA	B19	14-G	Insert
143	Fuel	1C05	CRA	C06	14-F	Insert
144	Fuel	1C15			15-L	Insert
145	Fuel	1C59			15-K	Insert
146	Fuel	1C39			15-H	Insert
147	Fuel	1C08			15-G	Insert
148-1	Support		DET A		9-E	Remove
148-2	Support		DET A		15-F	Insert
149	Fuel	1C04	CRA	C24	6-B	Insert
150	Fuel	1C18			6-A	Insert
151	Fuel	1A20	CRA	C36	7-C	Insert
152	Fuel	1B56	BPRA	B12	7-B	Insert
153	Fuel	1C07			7-A	Insert
154	Fuel	1A24	CRA	C37	8-D	Insert
155	Fuel	1B42	BPRA	B31	8-C	Insert
156	Fuel	1C02	CRA	C01	8-B	Insert
157	Fuel	1C30			8-A	Insert

TABLE 2.0-1 (cont'd)

STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
158	Fuel	1A53	CRA	C38	9-E	Insert
159	Fuel	1B32	BPRA	B39	9-D	Insert
160	Fuel	1A40	CRA	C25	9-C	Insert
161	Fuel	1B46	BPRA	B17	9-B	Insert
162	Fuel	1C14			9-A	Insert
163	Fuel	1B48	BPRA	B53	10-E	Insert
164	Fuel	1A33	APSRA	A01	10-D	Insert
165	Fuel	1B29	BPRA	B45	10-C	Insert
166	Fuel	1C12	CRA	C02	10-B	Insert
167	Fuel	1C48			10-A	Insert
168	Fuel	1A16	CRA	C26	11-E	Insert
169	Fuel	1B58	BPRA	B66	11-D	Insert
170	Fuel	1A08	CRA	C03	11-C	Insert
171	Fuel	1C19			11-B	Insert
172	Fuel	1B39	BPRA	B67	12-E	Insert
173	Fuel	1A34	CRA	C04	12-D	Insert
174	Fuel	1C27	BPRA	B48	12-C	Insert
175	Fuel	1A03	CRA	C05	13-E	Insert
176	Fuel	1C32	BPRA	B51	13-D	Insert
177	Fuel	1C44			13-C	Insert
178	Fuel	1C55			14-E	Insert
179	Fuel	1C38			14-D	Insert
180-1	Fuel	1C10	ORA	026	14-N	Remove
180-2	Fuel	1C10	ORA	026	4-P	Insert
181	Fuel	1C11			14-N	Insert
182-1	Fuel	1C22	ORA	005	2-D	Remove
182-2	Fuel	1C22	ORA	005	12-B	Insert

TABLE 2.0-1 (cont'd)

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STEP NO.	ASSEMBLY TYPE	ID #	FEATURE	ID #	CORE LOCATION	ACTION
183	Fuel	1C43			2-D	Insert
184	Support		DET B		1-L	Remove
185	Fuel	1C09			1-L	Insert
186	Support		DET A		15-F	Remove
187	Fuel	1C40			15-F	Insert

1414 036

TABLE 2.0-1 (cont'd)

FINAL FUEL LOADING DISTRIBUTION FOR CORE 1, CYCLE 1

X

FUEL TRANSFER CANAL

W — Y

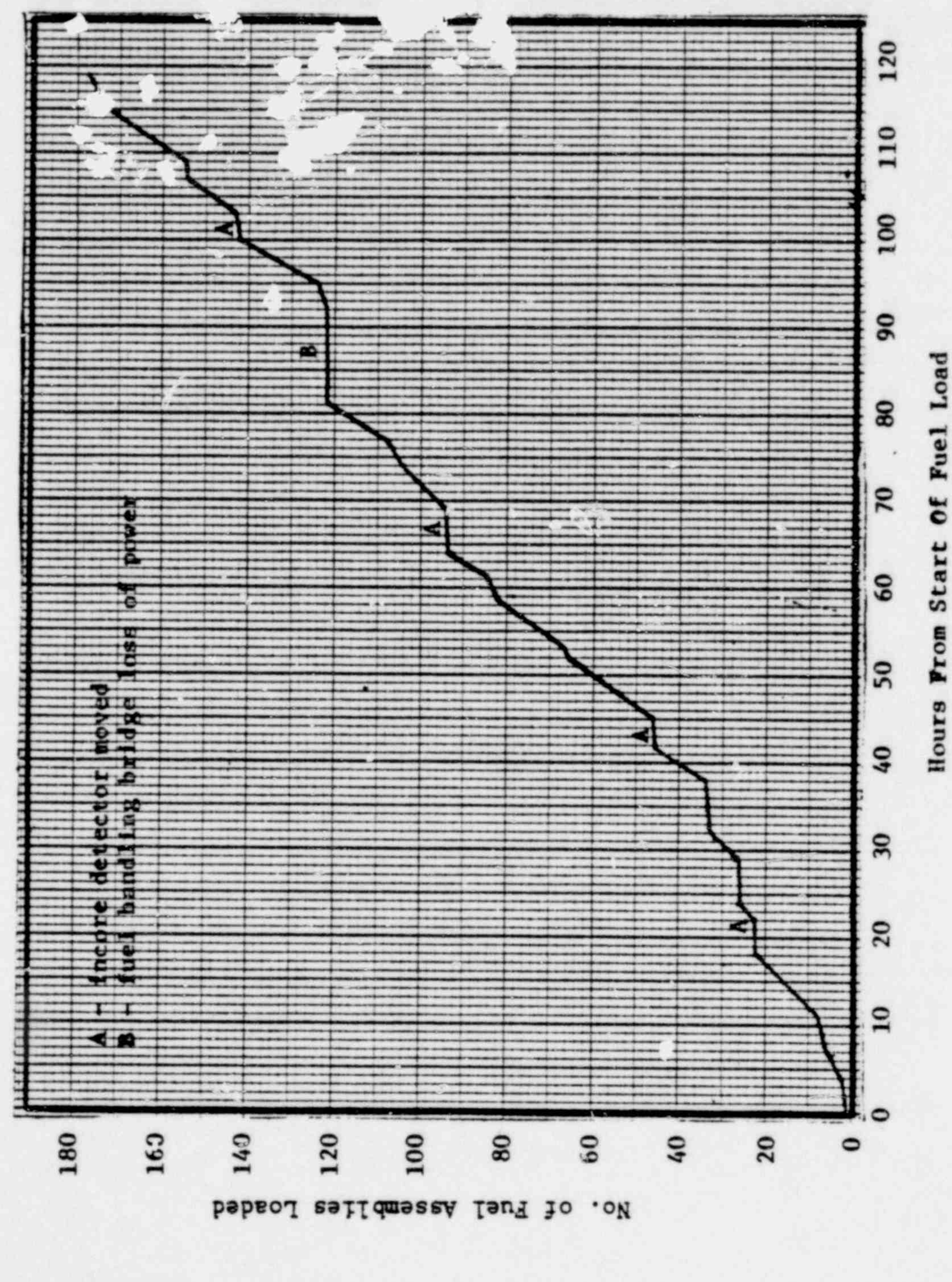
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						1C18	1C07	1C30	1C14	1C48					
B				1C41 038	1C23	1C04 C24	1B56 B12	1C02 C01	1B46 B17	1C12 C02	1C19	1C22 (*) 005			
C			1C46	1C31 B40	1A42 C23	1B43 B33	1A20 C36	1B42 B31	1A40 C25	1B29 B45	1A08 C03	1C27 B48	1C44		
D		1C43	1C33 B44	1A35 C22	1B07 B62	1A22 A08	1B40 B29	1A24 C37	1B32 B39	1A33 A01	1B58 B66	1A34 C04	1C32 B51	1C38	
E		1C36	1A13 C21	1B37 B64	1A54 C35	1B03 B61	1A51 C52	1B04 B30	1A53 C38	1B48 B53	1A16 C26	1B39 B67	1A03 C05	1C55	
F	1C49	1C01 C20	1B31 B43	1A11 A07	1B28 B63	1A44 C51	1B45 B11	1A27 C53	1B19 B16	1A43 C39	1B10 B65	1A50 A02	1B30 B50	1C05 C06	1C40
G	1C17	1B16 B14	1A37 C34	1B50 B41	1A39 C50	1B61 B13	1A10 C60	1B12 B09	1A06 C54	1B60 B18	1A48 C40	1B21 B47	1A05 C27	1B15 B19	1C08
H	1C29	1C54 C19	1B53 B52	1A19 C49	1B49 B37	1A56 C59	1B54 B08	1B35 C61	1B01 B05	1A15 C55	1B17 B46	1A23 C41	1B44 B49	1C13 C07	1C39
K	1C16	1B14 B15	1A47 C33	1B13 B36	1A28 C48	1B52 B10	1A26 C58	1B11 B01	1A46 C56	1B06 B04	1A30 C42	1B24 B21	1A31 C28	1B34 B20	1C59
L	1C09	1C50 C18	1B25 B42	1A09 A06	1B09 B59	1A07 C47	1B20 B07	1A29 C57	1B59 B03	1A01 C43	1B23 B56	1A49 A03	1B27 B71	1C57 C08	1C15
M		1C52	1A25 C17	1B05 B60	1A55 C32	1B02 B57	1A45 C46	1B08 B23	1A36 C44	1B26 B54	1A17 C29	1B55 B68	1A02 C09	1C53	
N		1C51	1C34 B35	1A18 C16	1B22 B58	1A12 A05	1B51 B32	1A52 C45	1B38 B22	1A21 A04	1B57 B55	1A14 C10	1C28 B28	1C11	
O			1C35	1C47 B34	1A32 C15	1B36 B38	1A38 C31	1B41 B24	1A41 C30	1B33 B27	1A04 C11	1C26 B26	1C37		
P				1C10 (*) 026	1C25	1C60 C14	1B18 B06	1C03 C13	1B47 B02	1C56 C12	1C42	1C21 018			
R						1C20	1C06	1C45	1C58	1C24					

Z

1A01 through 1A56 - 2.06 wt. % fuel assemblies
1B01 through 1B61 - 2.72 wt. % fuel assemblies
1C01 through 1C60 - 3.05 wt. % fuel assemblies
C01 through C61 - Control Rod Assemblies
A01 through A08 - Axial Power Shaping Rod Assemblies
038, 005, 026, 018 - Orifice Rod Assemblies
B01-B24, B26-B68, B71 - Burnable Poison Rod Assemblies
(*) - Assembly with Neutron Source

FIGURE 2.0-1

TMI UNIT I FUEL LOADING TIME



1414 038

FIGURE 2.0-2

3.0 POST FUEL LOAD PRECRITICAL TEST PROGRAM

A Post Fuel Load Precritical Test Program was conducted following initial fuel loading. This section of the report presents the scope and results of that testing.

The Control Rod Drive Drop Time Test was conducted at reactor coolant system conditions of 150°F, 450 psi and 532°F, 2155 psi with and without reactor coolant flow. The purpose of the test is to measure the total trip insertion time from trip initiation to three-fourths insertion for each control rod assembly. Reactor Coolant Pump Flow and Flow Coastdown measurements were conducted at system conditions of 532°F, 2155 psi to determine core flow characteristics. Pressurizer testing was also conducted at hot conditions to determine the pressurizer spray valve and bypass flow settings. Reactor coolant system leakage measurements were performed to verify that RCS leak rate was within acceptable limits. In all cases, applicable test criteria and Technical Specification requirements were met.

1414 039

3.1 REACTOR COOLANT PUMP FLOW TEST

3.1.1 PURPOSE

The Reactor Coolant Pump Flow Test was performed with the core installed to determine the functional capabilities of the Reactor Coolant System and Reactor Coolant Pumps and to determine the reactor coolant flow characterisitcs for various pump operating combinations.

3.1.2 TEST METHOD

Reactor coolant loop flows were determined by means of loop flowmeter ΔP cells installed in the reactor coolant system. The output of the ΔP cells were converted to temperature compensated flow indication according to Equation 3.1-1.

$$\text{Flow} = C_f \left[\Delta P \frac{V_c}{V_s} \right]^{1/2} \qquad \text{(Equation 3.1-1)}$$

Where: C_f = Flow coefficient = 397100

ΔP = Indicated flowmeter differential pressure

V_c = Specific volume at reference conditions of 68°F, 14.7 psi

V_s = Specific volume at system conditions

For each reactor coolant pump combination, steady state temperature, pressure and flow was maintained and data was recorded by the plant computer, brush recorders and reactimeter. Measured flow rates were averaged over a specified time and the results were compared with acceptance criteria.

3.1.3 TEST RESULTS

Reactor coolant flow was measured at 532°F, 2155 psi for twelve (12) reactor coolant pump operating combinations. Table 3.1-1 lists the measured flow rates along with the minimum and maximum allowable flows. As can be seen from table 3.1-1, all measured flow rates were within the acceptance criteria.

3.1.4 CONCLUSIONS

Reactor coolant flow measurements were conducted at 532°F, 2155 psi with the core installed and all measured flow rates were within the range of acceptable values.

1414 040

REACTOR COOLANT FLOW RATES AT 532^{o}F, 2155 PSI WITH REACTOR CORE INSTALLED

Case	Pump Combination (Pumps Running)	Minimum Acceptable Flow Rate ($X10^6$ lbm/hr)	Maximum Acceptable Flow Rate ($X10^6$ lbm/hr)	Measured Flow Rate ($X10^6$ lbm/hr
1	A	*	*	39.8
2	B	*	*	40.48
3	C	*	*	40.85
4	D	*	*	40.72
5	A, B, C, D	138.5	154.5	146.0
6	A, B, D	103.2	154.5	110.13
7	A, D	67.8	154.5	73.75
8	B, C, D	103.2	154.5	109.7
9	A, C	67.8	154.5	74.25
10	A, B	62.4	154.5	80.35
11	C, D	62.4	154.5	81.17
12	B, D	*	*	73.98

* - Indicates that no acceptance criteria was applied to the pump combination.

1414 041

TABLE 3.1-1

3.2 REACTOR COOLANT PUMP FLOW COASTDOWN TEST

3.2.1 PURPOSE

The Reactor Coolant Pump Flow Coastdown Test was performed to determine reactor coolant flow characteristics for specific reactor coolant pump trip combinations. Testing was conducted at system conditions of 532°F, 2155 psi with the core installed.

3.2.2 TEST METHOD

Eight (8) different reactor coolant pump combinations were selected for the measurement of flow coastdown characteristics. The eight combinations and a description of each is summarized in Table 3.2-1. For each pump combination, steady state conditions were established and data was recorded by the plant computer and test recorders. All or a portion of the coolant pumps were then tripped and data was recorded through the resulting flow transient. Reactor coolant flow indication was obtained from the loop flowmeter instrumentation as described in section 3.1. The hierarchy of single reactor coolant pump flows was determined during the Reactor Coolant Pump Flow Test, which was performed in conjunction with the coastdown test.

3.2.3 TEST RESULTS

The results of this test at 532°F, 2155 psi with the core installed are summarized in Figures 3.2-1 through 3.2.-4. Measured reactor coolant flow versus time is plotted along with the acceptance criteria limits for each reactor coolant pump combination. The flow values plotted were obtained by dividing the indicated flow at a specific time, t, after the trip by the measured minimum initial core flow with the pump combination running. All minimum core flow criteria were met. In addition to the minimum flow criteria, a further requirement was imposed upon cases 4, 5 and 6, that the reactor coolant flow decrease by a certain percentage within a specified time after the pumps were tripped. This criteria was also met in each case.

3.2.4 CONCLUSIONS

The reactor coolant flow coastdown characteristics measured at system conditions of 532°F, 2155 psi with the core installed met all applicable acceptance criteria.

1414 042

TABLE 3.2-1

REACTOR COOLANT PUMP FLOW COASTDOWN COMBINATIONS

Case	Pump Initially Running	Pumps Tripped
1	A, B, C, D	A, B, C, D
2	A, B, D - three lowest flow pumps	A, B, D
3	A, D - lowest flow pumps in each loop	A, D
4	A, B, C, D	C - highest flow pump
5	A, B, C, D	C, D - pumps in higher flow loop
6	A, B, C, D	B, C - higher flow pump each loop
7	B, C, D	B - pump in loop with idle pump
8	B, C, D	C - higher flow pump in loop with two pumps operating

1414 043

MEASURED REACTOR COOLANT FLOW COASTDOWN AT 532°F

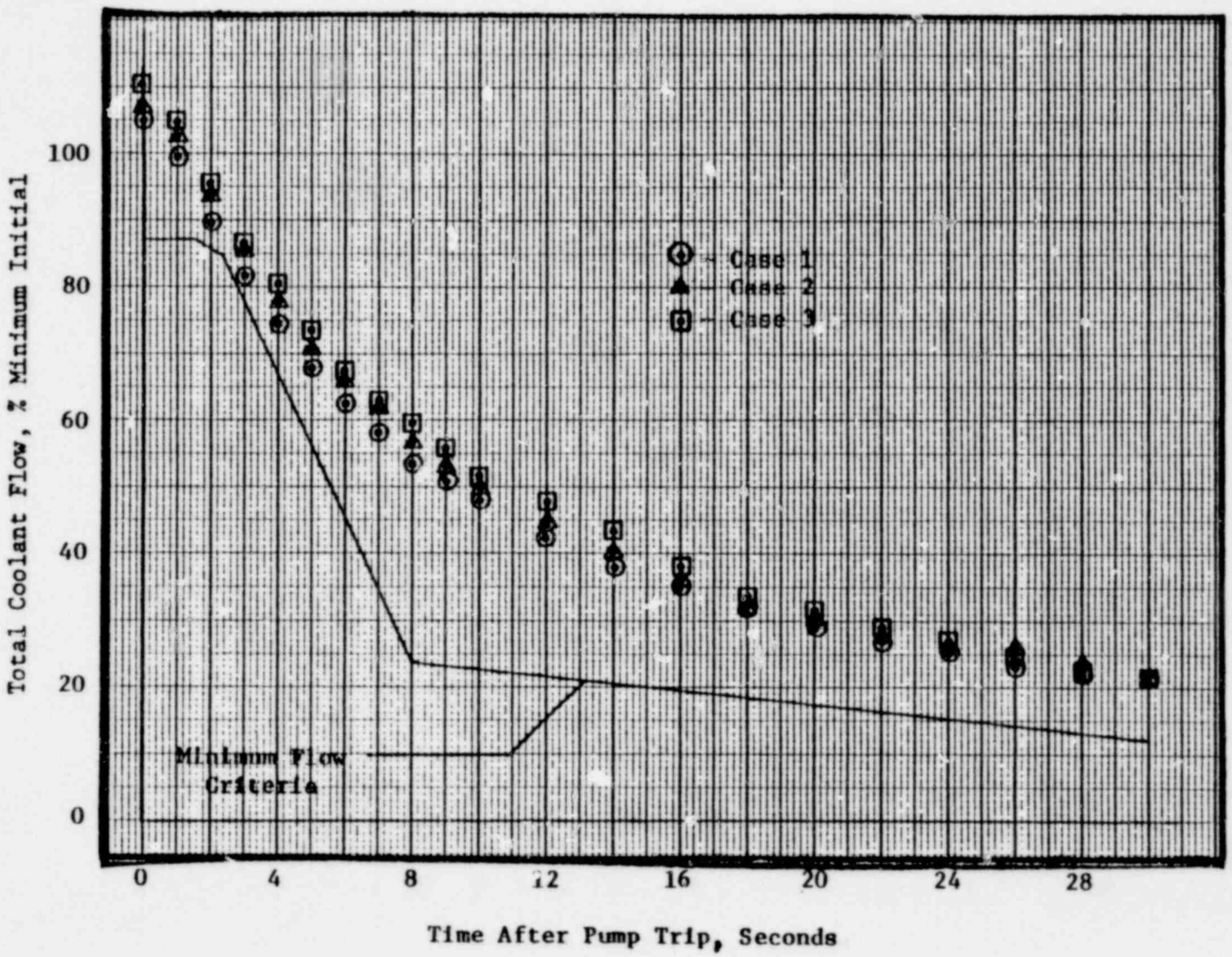


FIGURE 3.2-1

1414 044

MEASURED REACTOR COOLANT FLOW COASTDOWN AT 532°F

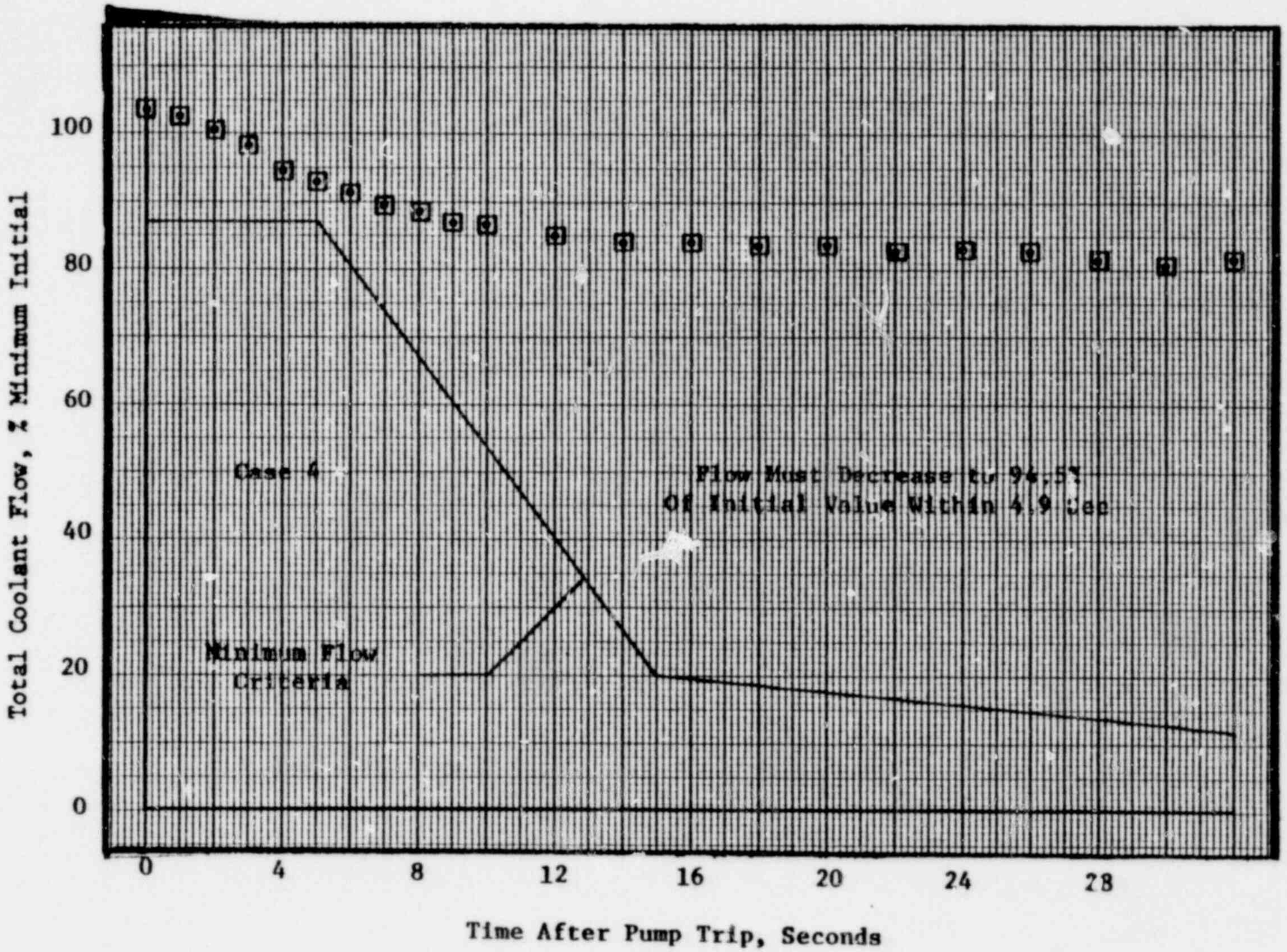


FIGURE 3.2-2

1414 045

MEASURED REACTOR COOLANT FLOW COASTDOWN AT 532°F

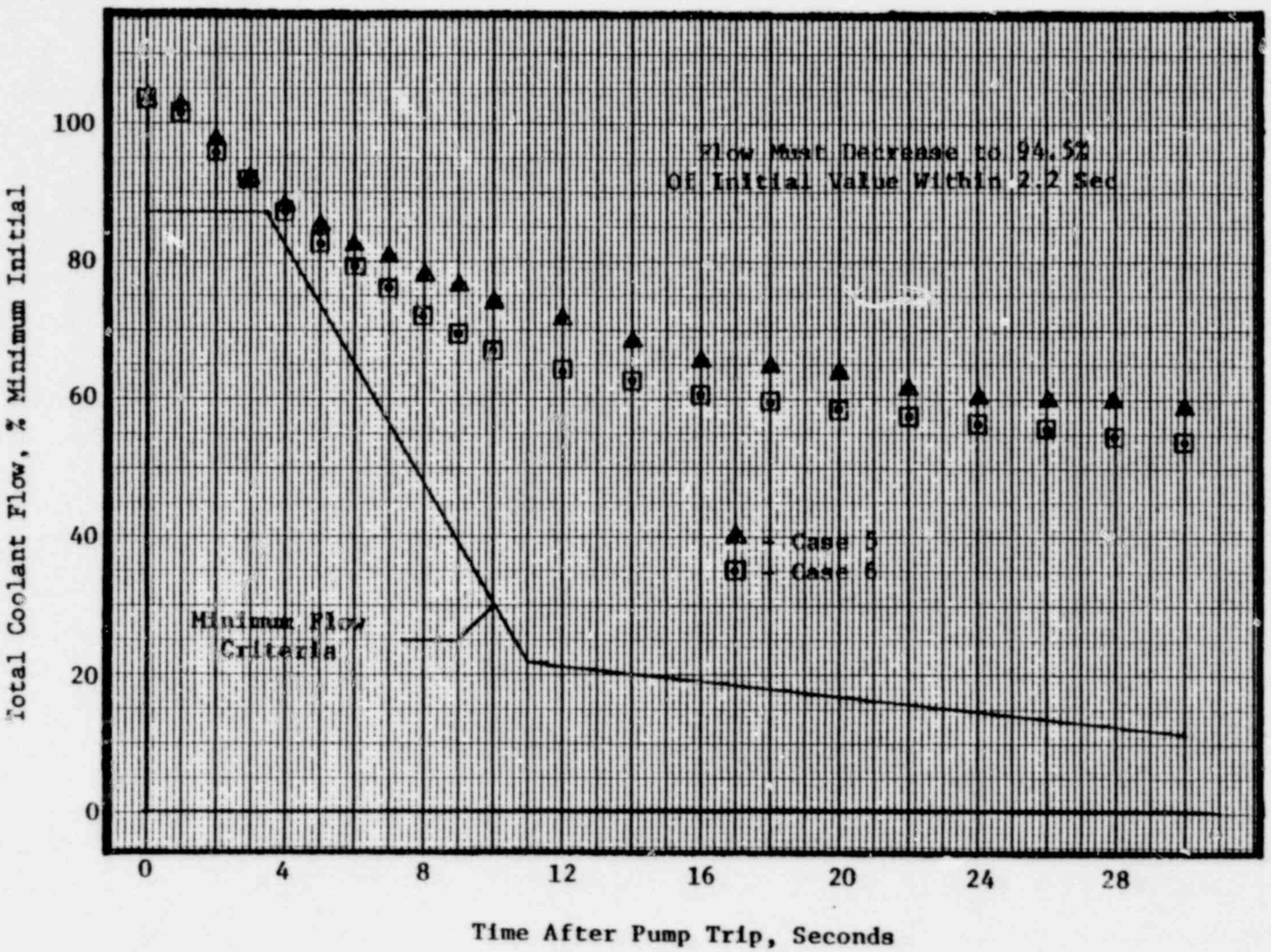


FIGURE 3.2-3

MEASURED REACTOR COOLANT FLOW COASTDOWN AT 532°F

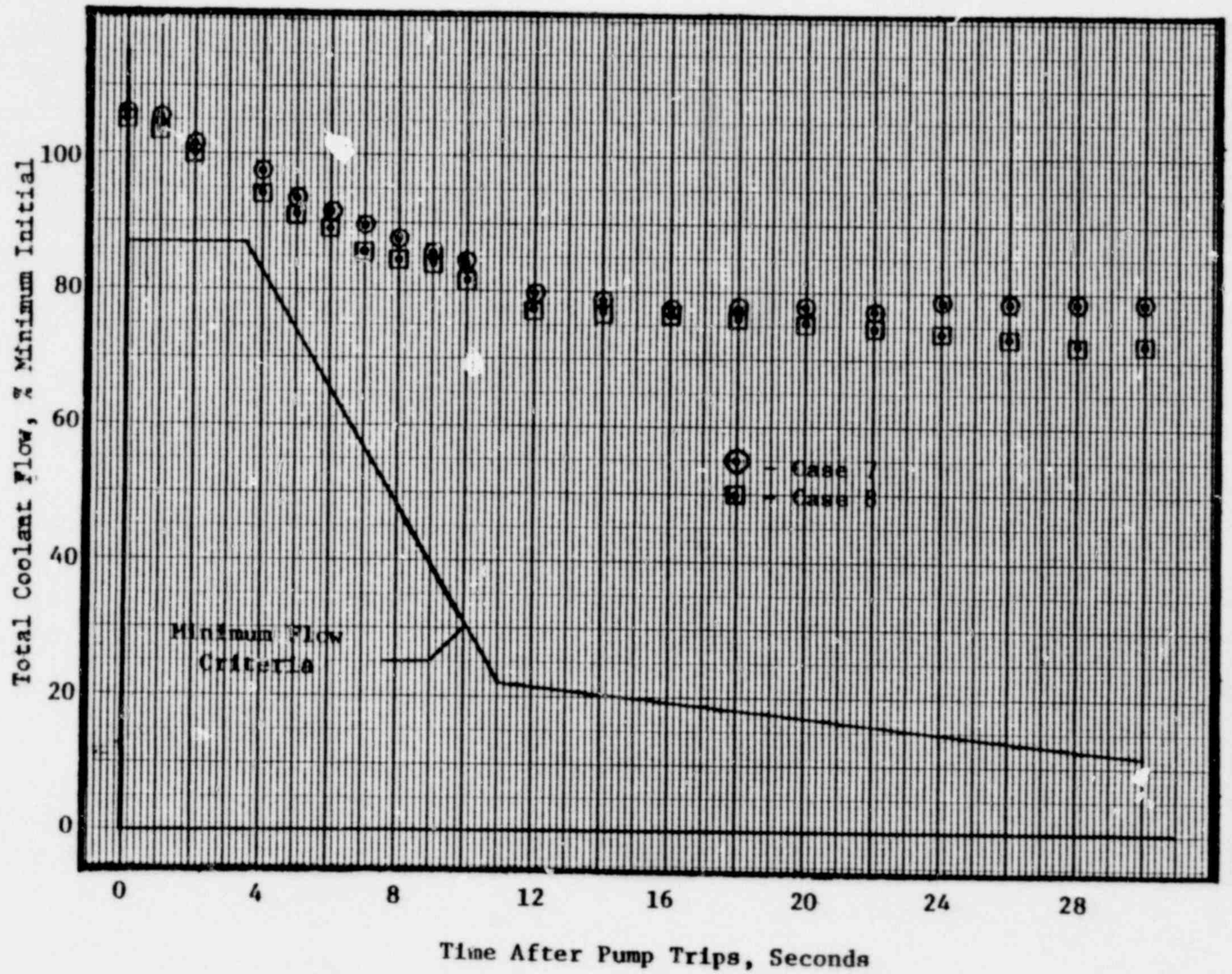


FIGURE 3.2-4

1414 047

3.3 CONTROL ROD DRIVE DROP TIME TEST

3.3.1 PURPOSE

Technical Specifications 4.7 places limits on the control rod trip insertion times for reactor coolant system full flow and no flow conditions. The Control Rod Drive Drop Time Test measures the data to fulfill the Technical Specification limit and to establish data for future periodic testing.

3.3.2 TEST METHOD

The Control Rod Drive Drop Time Test was performed using strip chart recorders to time the rod drops. Each control rod group was pulled to 100% withdrawn and then dropped into the core using the manual trip pushbutton. A zero time signal was furnished to the test recorders for each control rod assembly from a contact on the manual trip switch. A second signal to indicate three-fourths insertion was furnished to the recorders by a reed switch located on the position indicator tube of each control rod drive. The test was conducted at nominal reactor coolant system conditions of 150°F, 450 psi and 532°F, 2155 psi under flow and no flow conditions. Control rod groups 1 through 7 were each withdrawn to 100% and tripped at each of the four (4) test conditions. After drop time measurements on all the groups were completed, the rods with the fastest and slowest trip insertion times were tripped ten additional times to demonstrate repeatability of the measurement. Measurements were performed on the group 8 control rods to verify that they do not drop into the core when power to the control rod drive trip breaker undervoltage coils is interrupted.

3.3.3 TEST RESULTS

The measured results for the first test condition of 150°F, 450 psi with no reactor coolant flow show that rod H-10 was the fastest at 1.104 sec. and rod M-7 was slowest at 1.16 sec. Ten additional drops on rods H-10 and M-7 produced drop times within 16 ms and 20 ms, respectively. The group 8 rods were withdrawn to 25% and no rod movement was observed when the control rod drives were tripped, as required.

The measured results for the second test condition of 150°F, 450 psi with one reactor coolant pump operating show that rod F-10 was fastest at 1.128 seconds and rod M-9 was slowest at 1.176 seconds. Ten additional drops on rods F-10 and M-9 produced drop times within 16 ms and 24 ms, respectively.

The measured results for the third test condition of 532°F, 2155 psi with no reactor coolant pumps operating show that rod H-10 was fastest at 1.072 seconds and rod O-5 was slowest at 1.115 seconds. Ten additional drops on rods H-10 and O-5 produced drop times within 25 ms and 20 ms, respectively.

The measured results for the fourth test condition of 532°F, 2155 psi with 100% reactor coolant flow conditions show that rod H-10 was the fastest at 1.225 seconds and rod M-5 was the slowest at 1.363 seconds. Ten additional drops on rods H-10 and M-5 produced drop times within 25 ms and 19 ms, respectively.

1414 048

3.3.4 CONCLUSIONS

Control Rod Drop Time measurements conducted at 150°F and 532°F show that the control rod assembly trip insertion time from 100% withdrawn to three-fourths insertion will not exceed 1.40 seconds under reactor coolant no flow conditions and 1.66 seconds under reactor coolant flow conditions. The requirements of Technical Specification 4.7.1 were met in all cases.

1414 049

3.4 PRESSURIZER TEST

3.4.1 PURPOSE

Pressurizer Operational testing was conducted prior to initial criticality to set the pressurizer spray and bypass flows at the prescribed setpoints.

3.4.2 TEST METHOD

The technique used to set the pressurizer spray and bypass flows was based upon balancing the heat input to and the heat losses from the pressurizer. Initial steady state pressure and temperature conditions were established in the pressurizer without spray or bypass flow. The power input from the pressurizer heaters necessary to maintain steady state conditions was recorded. The additional heat input required to balance spray and bypass flow was then calculated using Equation 3.4-1.

$$F = \frac{K\ (\Delta Q)}{h_{fp} - h_{f\ RCS}} \qquad \text{(Equation 3.4-1)}$$

Where: F - is the spray or bypass flow

K - is a constant = 9.03

ΔQ - is the difference between the heater input with flow and the heater input without flow

h_{fp} - is the enthalpy of saturated water at the pressurizer temperature

$h_{f\ RCS}$ - is the enthalpy of saturated water at the RCS temperature

Heat input to the pressurizer from the heaters was then increased by the amount calculated. The bypass and spray valve flows were increased to balance the additional heat input and maintain the pressurizer temperature and pressure at their initial values.

3.4.3 TEST RESULTS

The measured results from setting the pressurizer spray and spray valve bypass flow are listed in Table 3.4-1. The bypass and spray flows were set at 0.99 gpm and 190.5 gpm, respectively. The measured pressurizer heat loss was in excess of 100KW at system conditions of 532°F and 2155 psi.

3.4.4 CONCLUSIONS

The pressurizer spray flow was set within the acceptance criteria limit of 190.0 +19/-6 gpm. The pressurizer spray bypass flow was set within the acceptance criteria limit of 1.0 +0.5/-0.25 gpm.

1414 050

TABLE 3.4-1

MEASURED RESULTS FOR DETERMINATION OF PRESSURIZER SPRAY AND BYPASS FLOW AT 532°F

A. Pressurizer Spray Bypass Flow

Test Conditions	RCS Pressure	RCS Temperature	Pressurizer Temperature	Heater Power	Spray Bypass Flow
initial	2157psig	532.0°F	644.1°F	106.9KW	0.0gpm
final	2159psig	531.0°F	644.2°F	124.5KW	0.99gpm

B. Pressurizer Spray Flow (with bypass flow)

Test Conditions	RCS Pressure	RCS Temperature	Pressurizer Temperature	Heater Power	Spray Flow
initial	1406psig	530.7°F	586.8°F	132.69KW	0.0gpm
final	1404psig	533.6°F	587.1°F	1609.35KW	190.5gpm

1414 051

3.5 REACTOR COOLANT SYSTEM LEAKAGE

3.5.1 PURPOSE

The purposes of the Reactor Coolant System (RCS) hot leakage test were as follows:

1) Determine RCS leakage by calculating change in RCS inventory over a period of time.

2) Determine the accuracy of the method used to determine RCS leakage by imposing a "known" leak rate.

3) Examine systems containing reactor coolant to identify leakage.

4) Establish a value for"normal exaporative losses" as used by Technical Specification 3.1.6.2.

5) Verify the Surveillance Procedure for RCS leakage determination.

3.5.2 TEST METHOD

The RCS hot leakage and surveillance procedure verification test was performed during the hot functional and post fuel load pre-critical test programs and its results served as a basis for conducting the Surveillance Procedure for RCS leakage determination during the power escalation test program.

With the primary plant at 532°F and 2155 psig, pressurizer level, makeup tank level, reactor coolant drain tank (RCDT) level and RCS temperature were monitored as a function of time. Changes in RCS inventory were computed over a four hour period. These computations resulted in a measured leak rate of 0.821 gpm during the four hours, when corrected for an RCP #3 seal purge flow addition of 0.07 gpm.

A known leak rate of .68 gpm was then established to determine the sensitivity of the above computations in yielding accurate values for leakage. Again, changes in RCS inventory were computed over a four hour period. The computations resulted in a leak rate of 0.659 gpm during the four hours, when corrected for RCP #3 seal purge flow addition and the known leak rate of .68 gpm.

Independent of the leakage monitoring operations above, a survey of all primary system boundary piping, valves, fittings, instrument connections, and flanges was made in an attempt to measure and identify every drop of leakage that was not evaporating to the containment, auxiliary building, or RCDT atmosphere. This survey resulted in a measured leakage of 0.231 gpm. The difference between the average value of computed leakage for the two four-hour runs minus the survey measured (identified) leakage is the established value of "normal evaporative losses" used in the Surveillance Procedure for leakage determination during normal plant operation.

RCS leakage was monitored every day during the power escalation program when the reactor was critical, as required by Technical Specifications. The Surveillance Procedure for leakage determination was used for this purpose.

3.5.3 TEST RESULTS

The average value of computed leakage for the two four-hour runs was 0.740 gpm. No value of leakage computed for any single hour out of the eight differed from the 0.740 gpm figure by more than ± 1 gpm, thereby supporting the contention that RCS level and temperature instrumentation is sensitive enough to detect a 1 gpm leak within 1 hour.

The total measured identified leakage was 0.231 gpm. This results in a value of 0.51 gpm for the "normal evaporative losses".

3.5.4 CONCLUSIONS

Reactor Coolant System hot leakage measurements were conducted prior to initial criticality. The measured results verify that the reactor coolant leakage does not exceed the Technical Specification requirements and that the normal control instrumentation is sensitive enough to perform leak rate measurements.

1414 053

4.0 CORE PERFORMANCE - MEASUREMENTS AT ZERO POWER

Three Mile Island Unit One, Core 1 consists of 177 fuel assemblies, each containing 208 fuel rods, 16 control rod guide tubes and one incore instrument guide tube. The arrangement of these assemblies is shown in Figure 4.0-1. The inner 117 assemblies, which are arranged in a checkerboard pattern, are of two different enrichments - 2.06 and 2.72 wt. % uranium - 235. An outer ring of 60 assemblies enriched to 3.05 wt. % uranium - 235 completes the core. Lumped burnable poison is distributed throughout the core. A detailed loading map of Core 1, with each fuel assembly, control rod, orifice rod and lumped burnable poison assembly is given in Figure 2.0-1 of section 2.0.

The reactivity of the core is controlled by 61 full-length Ag-In-Cd control rods and soluble boron in the Reactor Coolant System (RCS). Eight (8) partial length control rods are provided for additonal control of axial power distributions. The locations of the 69 control rods are also shown in Figure 4.0-1. The important design data and calculated performance characteristics of Core 1 are tabulated in Table 4.0-1.

Core performance measurements were conducted during the Zero Power Test Program which began on June 5, 1974 and ended on June 10, 1974. This section presents the results and an evaluation of the zero power tests, which included initial criticality, nuclear instrumentation overlap, verification of reactivity calculations, all rods out critical boron determination, temperature coefficient measurements, shutdown margin determination and soluble poison and control rod reactivity worth measurements. A comparison of measured and predicted results is given based on on-site analysis. In all cases, the applicable test and Technical Specification acceptance criteria were met.

1414 054

Table 4.0-1. Core 1 Design Data and Performance Characteristics

Reactor	
Design heat output, MWt*	2535
Vessel coolant inlet temp, F	554
Vessel coolant outlet temp, F	603.8
Core coolant outlet temp, F	606.2
Core coolant operating pressure, psig	2185
Core coolant T_{avg}, F	579.3
Core and Fuel Assemblies	
Total number fuel assy in core	177
Number fuel rods per fuel assy	208
Number control rod guide tubes per assy	16
Number incore instr positions per fuel assy	1
Fuel rod outside diameter, in.	0.430
Cladding thickness (min) in.	0.026
Fuel rod pitch, in.	0.568
Fuel assembly pitch spacing, in.	8.587
Cladding material	Zircaloy-4 (cold worked)
Fuel	
Material	UO_2
Form	Dished-end, cylindrical pettets
Pellet diameter, in.[a]	0.364
Active length, in.[a]	141.2
Density (Unit 1, Core 1), % theor [a]	92.5
Heat Transfer and Fluid Flow at Rated Power [a]	
Total heat transfer surface in core, ft^2	48,766
Average heat flux, Btu/h-ft^2	174,870
Maximum heat flux (at min DNBR), Btu/h-ft^2	469,873
Average power density in core, kW/ℓ	82.31
Average thermal output, kW/ft of fuel rod	5.69
Maximum thermal output, kW/ft of fuel rod	18.2
Maximum cladding surface temp, F	650
Average fuel temp of hottest pin, F	3,237
Maximum fuel central temp at hot spot, F	4,953
Total reactor coolant flow, 10^6 lb/h	131.32
Core flow area (eff for heat transfer), ft^2	49.19
Core coolant average velocity, fps	15.73
Coolant outlet temp at hot channel, F	647.1

*Note: The core will be operated with a 100% FP value of 2535 MWt (Technical Specification limit) even though the rated power of this core is 2568 MWt.

Power Distribution

Maximum/average power ratio, radial x local ($F\Delta_h$ nuclear)	1.78
Maximum/average power ratio, axial (F_z nuclear)	1.70
Overall power ratio (F_q nuclear)	3.03
Power generated in fuel and cladding, %	97.3

Hot Channel Factors

Power peaking factor (FQ)	1.011
Flow area reduction factor (F_A)	
Interior bundle cells	0.98
Peripheral bundle cells	0.97
Local heat flux factor (F_Q")	1.014
Hot spot maximum/average heat flux ratio (Fq nuc and mech)	3.12

DNB Data

Design overpower, % rated power	112
DNB ratio at design overpower (W-3)	1.55
DNB ratio at design power (W-3)	2.0
Limiting DNB ratio at design overpower (W-3)	1.3

Fuel Assembly Volume Fractions

Fuel	0.303
Moderator	0.580
Zircaloy	0.102
Stainless steel	0.003
Void	0.012
	1.000

Total UO_2 (BOL, First Core)

Metric tons	93.1

Core Dimensions, in.

Equivalent diameter	128.9
Active height (with/without densification)	141.1/144.0

Unit Cell H_2O to U Atomic Ratio (Fuel Assembly)

Cold	2.88
Hot	2.06

Full-Power Lifetime, days

First cycle	460

1414 056

TABLE 4.0-1 (Cont'd)

Fuel Irradiation, MWd/mtU

First cycle average	14,400

Fuel Loading, wt % ^{235}U

Core average first cycle	2.62

Control Data

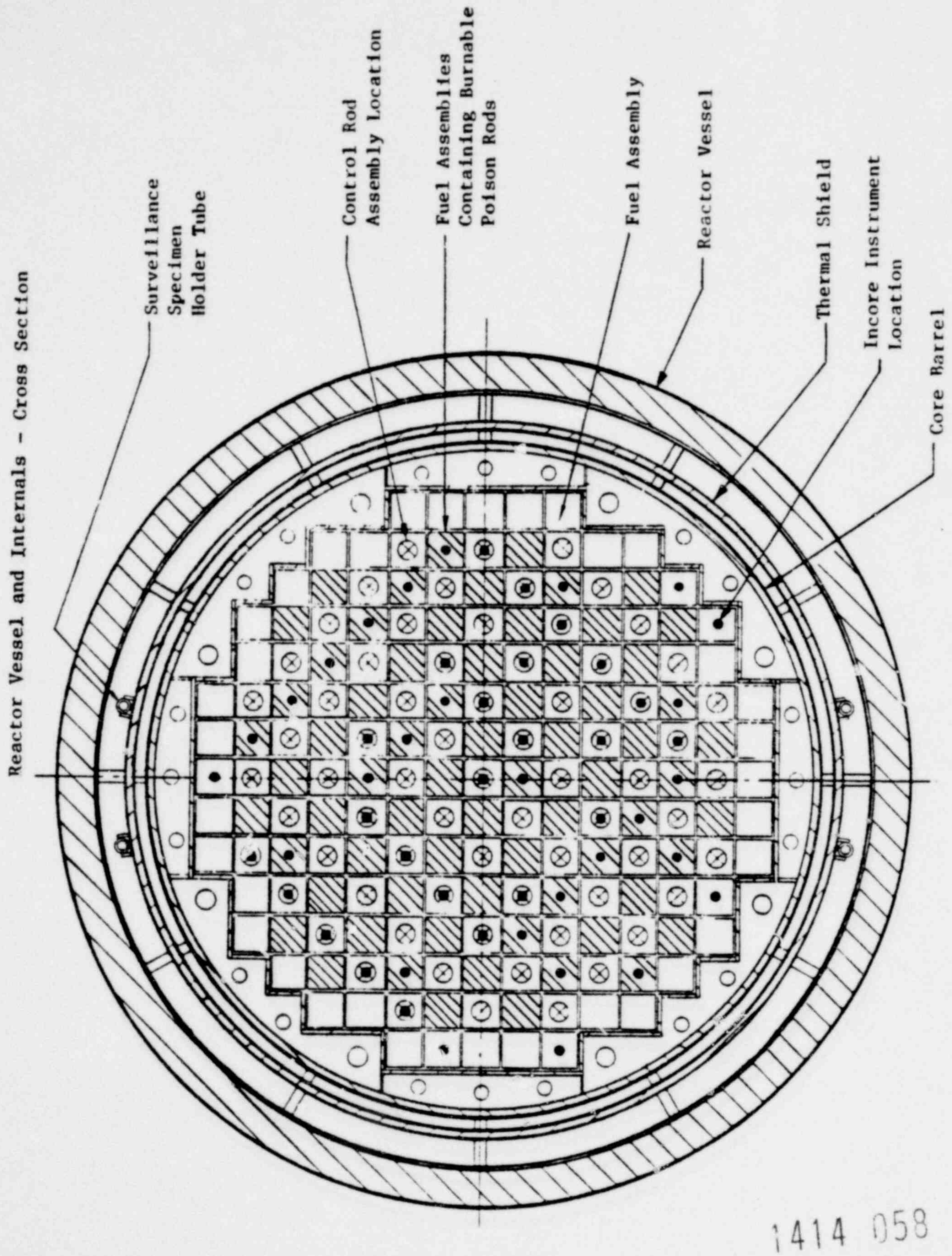
Control rod material	Ag-In-Cd
Number full-length rods	61
Number APSRs	8
Control rod cladding material	SS-304

- - - -

(a) Following densification

1414 057

TABLE 4.0-1 (Cont'd)



1414 058

FIGURE 4.0-1

4.1 INITIAL CRITICALITY

Initial criticality was achieved on June 5, 1974 at reactor conditions of 532°F and 2155 psig. Control rod groups 1 through 4 were previously withdrawn during the heatup to 532°F. The initial reactor coolant system (RCS) boron concentration was 2086 ppm. The approach to critical began by withdrawing control rod groups 5, 6 and 8 to 100% and positioning group 7 at 75% withdrawn. Criticality was subsequently achieved by deborating the reactor coolant system to a boron concentration of 1545 ppm. The procedure used in the approach to critical is outlined below in three basic steps.

Step 1 Control Rod Withdrawal

Group 8 100% withdrawn
Group 5 100% withdrawn
Group 6 100% withdrawn
Group 7 75% withdrawn

Step 2 Deborate using a feed and bleed flow rate of 50 gpm until criticality is almost achieved, as indicated by any inverse count rate plot reading approximately 0.05.

Step 3 Stop deboration and increase letdown flow to maximum (140 gpm) to enhance mixing between the makeup tank and the reactor coolant system. Achieve initial criticality and position control rod group 7 to control neutron flux as the reactor coolant system boron concentration reaches equilibrium.

Throughout the approach to criticality, plots of inverse multiplication were maintained by two independent persons. Two plots of inverse count rate (ICR) versus control rod position were maintained during control rod withdrawal. Two plots of ICR versus RCS boron concentration and two plots of ICR versus gallons of demineralized water added were maintained during the dilution sequence. At the end of each reactivity addition (boron dilution or control rod withdrawal), count rates were obtained from each startup range neutron detector channel. The ratio of the initial average count rate to the count rate at the end of each reactivity addition is the value plotted.

During control rod withdrawal (Step 1) ICR plots versus control rod group position were maintained from the outputs of source range channels NI 1 and 2. The withdrawal interval for each control rod group was limited to no more than half the remaining predicted distance to criticality as determined from the ICR plots.

Deboration of the reactor coolant system was accomplished in two steps as indicated above. First, deboration from 2086 ppm was commenced using a feed and bleed flow rate of 50 gpm (Step 2). RC boron samples were taken every 30 minutes and samples from the makeup tank and the pressurizer were taken hourly. Two ICR plots were maintained vs. every 1000 gallons of demineralized water added, and two plots were maintained versus RC letdown concentration every 30 minutes. Deboration at a letdown rate of 50 gpm was continued until one of the ICR plots indicated 0.05. At this time, demineralized water additions were stopped and the letdown flow rate was increased to 140 gpm to expedite mixing in the RCS (Step 3). RCS boron concentration at this time was approximately 1650 ppm. After initial criticality was achieved, control rod group 7 was inserted to control neutron flux during the subsequent mixing.

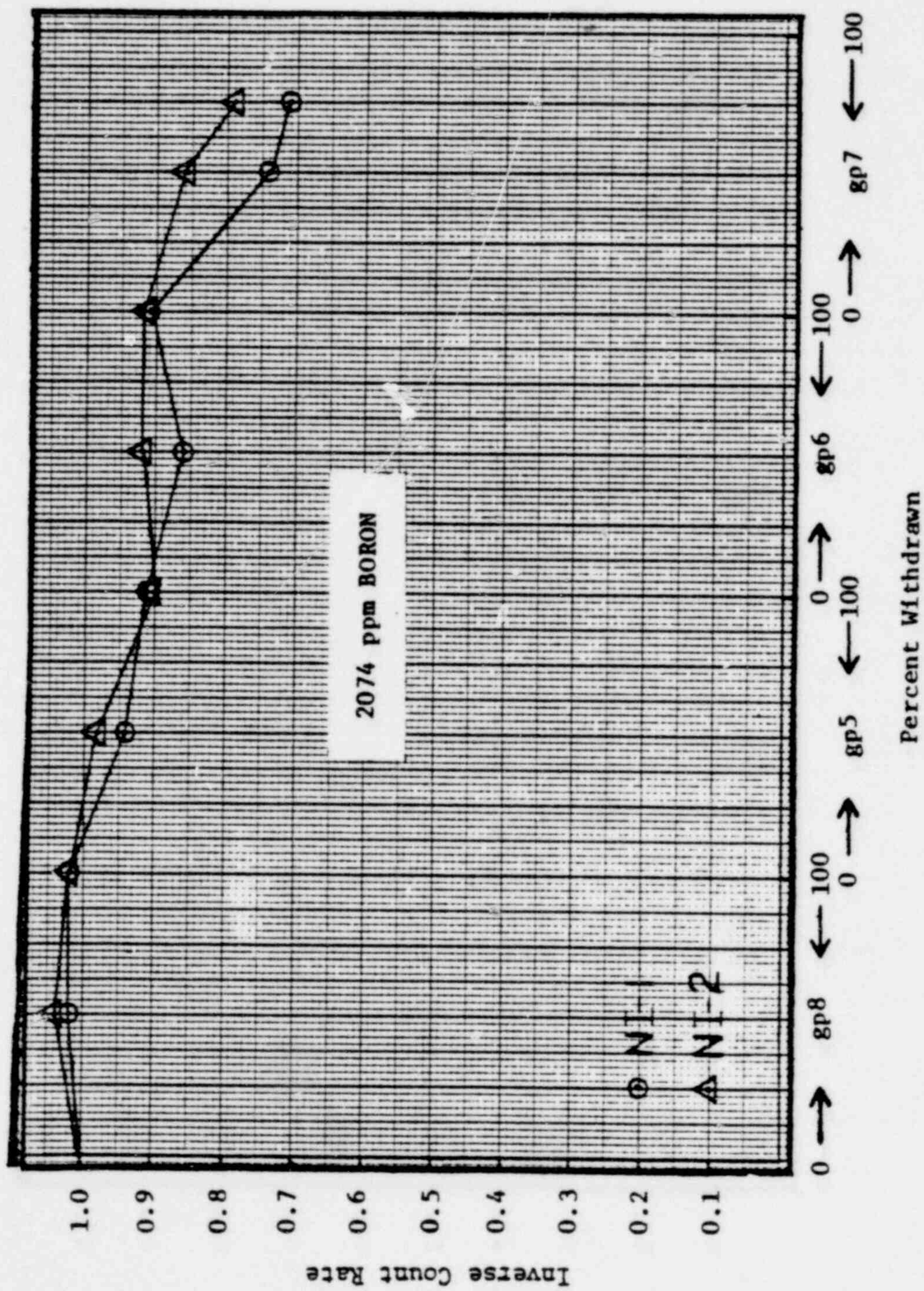
"Just critical" conditions were stabilized and maintained at an equilibrium boron concentration of 1545 ppm with group 7 at 26.5% withdrawn. The measured critical boron concentration with group 7 at 75% withdrawn was 1609 ppm which compares well with the predicted value of 1625 ppm. The inverse count rate plots maintained during the approach are presented in Figures 4.1-1 through 4.1-5. As can be seen from the plots, the response of the source range channels during reactivity additions was very good. Figure 4.1-1 is the plot of ICR versus control rod group withdrawal for data taken from NI channels 1 and 2. Figures 4.1-2 and 4.1-3 are the ICR plots versus RCS boron concentration and Figures 4.1-4 and 4.1-5 are the ICR plots versus gallons of demineralized water added to the RCS, for source range channels NI-1 and NI-2, respectively.

In summary, initial criticality was achieved in an orderly manner. There was good agreement between the measured and the predicted critical boron concentration, with the difference between the two directly attributable to the change in group 7 position from 75% to 26.5% withdrawn.

1414 060

INVERSE COUNT RATE VERSUS ROD WITHDRAWAL

NI - 1 and NI - 2



1414 061

FIGURE 4.1-1

INVERSE COUNT RATE VERSUS BORON CONCENTRATION

NI - 1

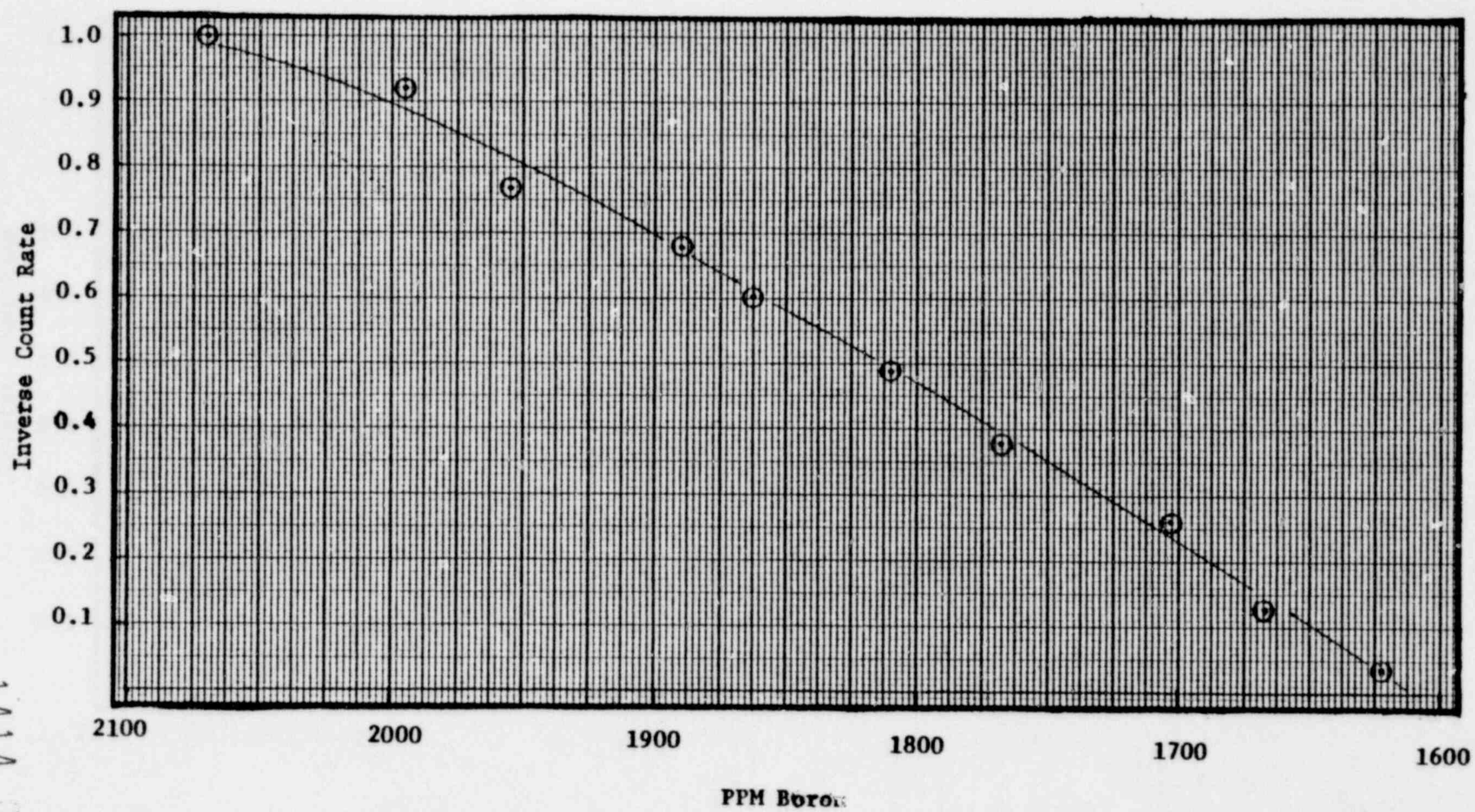


FIGURE 4.1-2

1414 062

INVERSE COUNT RATE VERSUS BORON CONCENTRATION

NI - 2

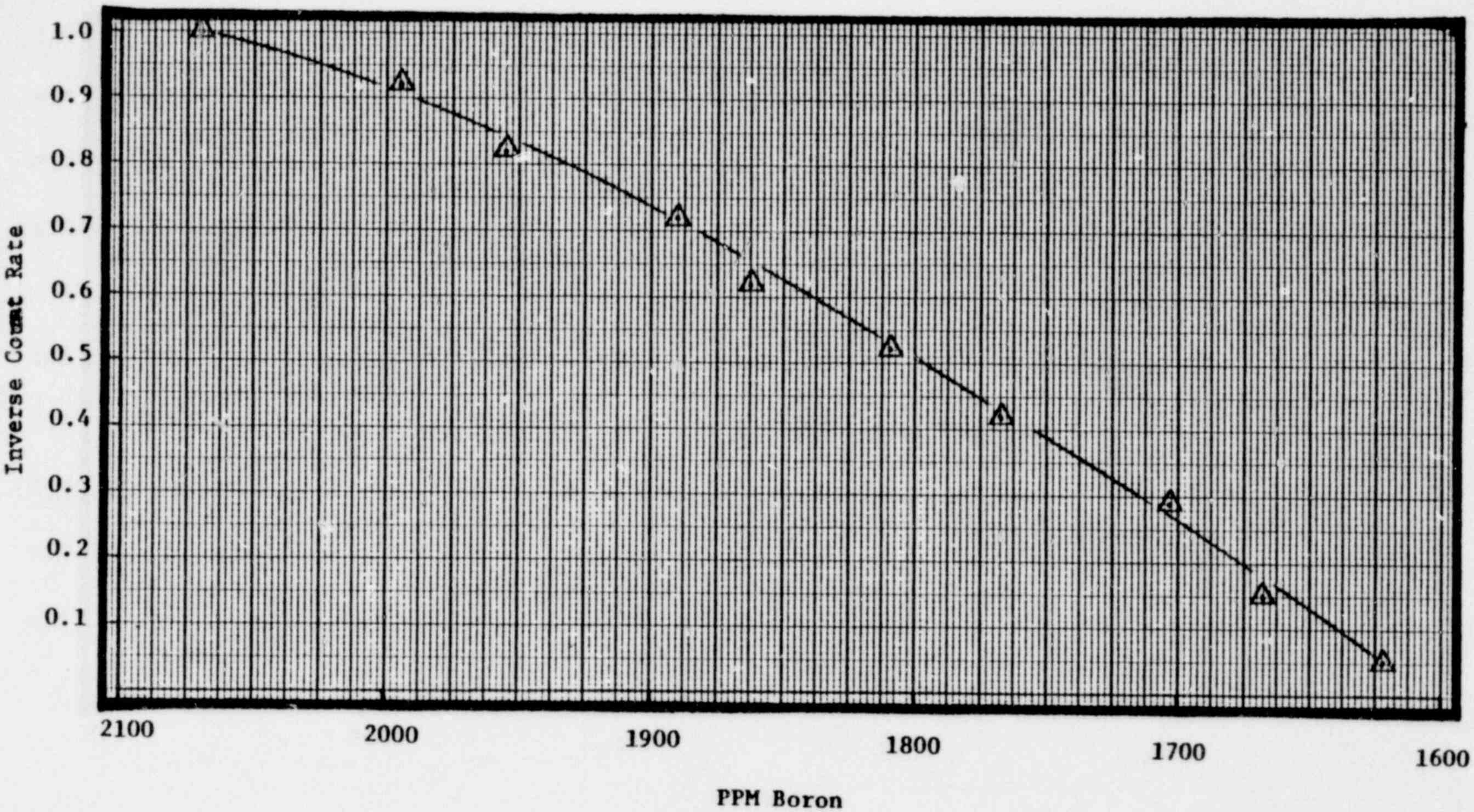


FIGURE 4.1-3

1414 063

INVERSE COUNT RATE VERSUS DEBORATION

NI - 1

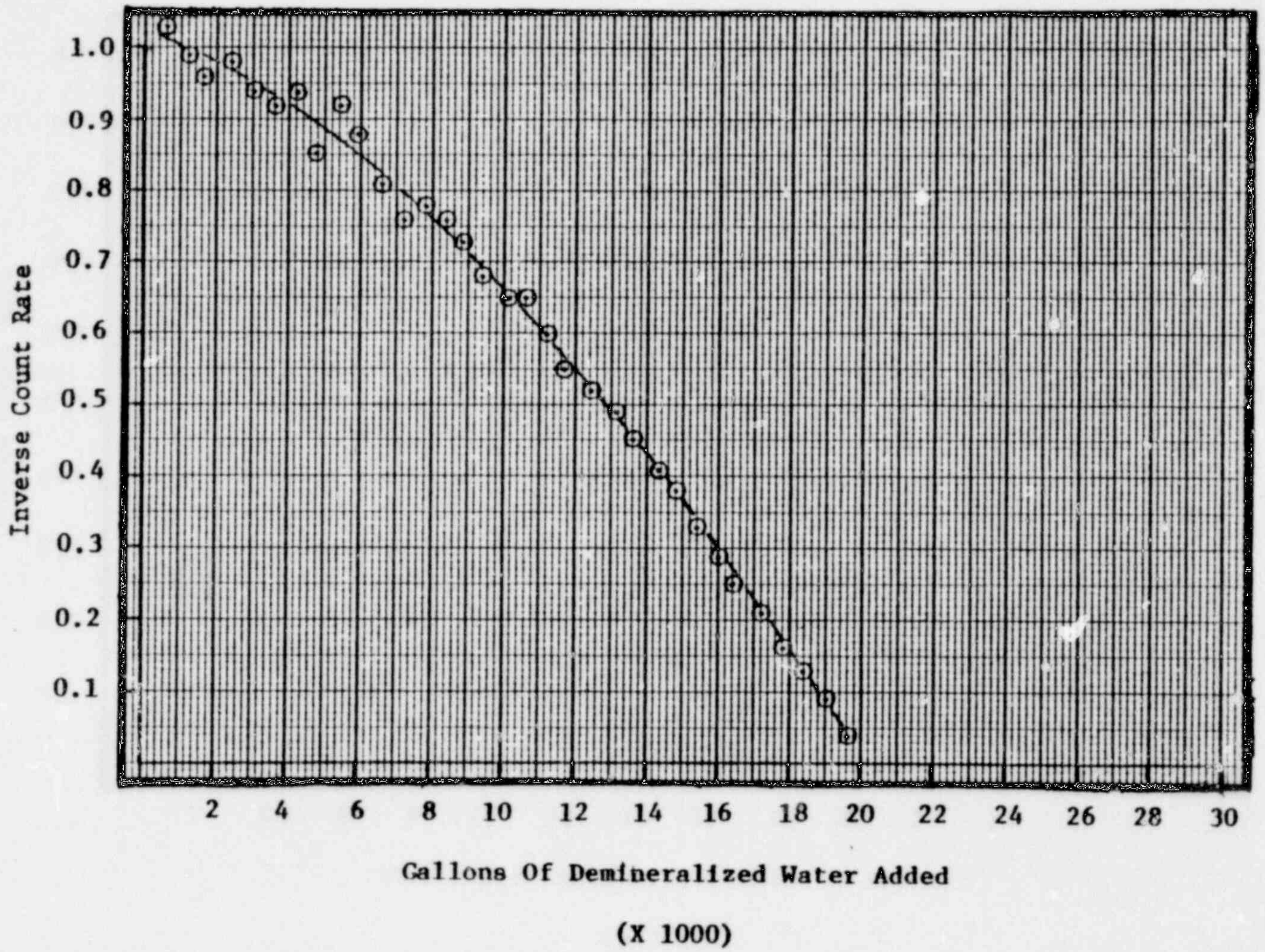


FIGURE 4.1-4

1414 064

INVERSE COUNT RATE VERSUS DEBORATION

NI - 2

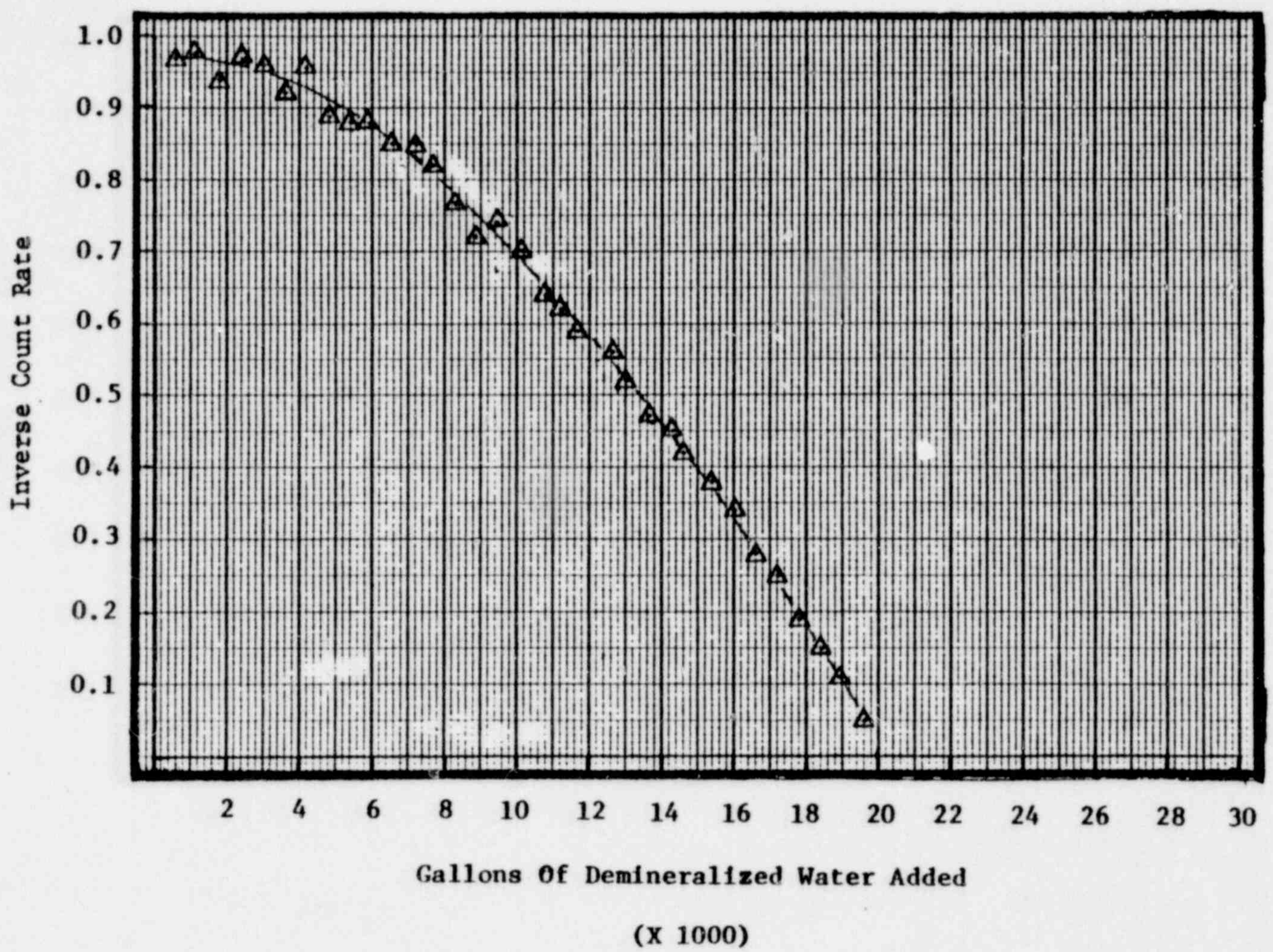


FIGURE 4.1-5

1414 065

4.2 NUCLEAR INSTRUMENTATION OVERLAP

4.2.1 PURPOSE

Technical Specification 3.5.1 states that prior to operation in the intermediate nuclear instrumentation (NI) range, at least one decade of overlap between the source range NIs and the intermediate range must be observed. This means that before the source range count rate equals 10^5 cps the intermediate range NI must be on scale. In addition, the following number of NI channels must be in operation for the test program to continue beyond initial criticality.

	Channels Available	Minimum Operating
Source Range NI	2	2
Intermediate Range, NI	2*	2

* One channel was input to the reactimeter but was operable.

4.2.2 TEST METHOD

To satisfy the above overlap requirements after initial criticality was achieved, core power was increased until the intermediate range channels came on scale. Detector signal response was then recorded for both the source range and intermediate range channels. This was repeated for two more decades until the source range channels approached 10^6 cps.

4.2.3 TEST RESULTS

The results of the initial NI overlap data at 532°F and 2155 psig are plotted on Figure 4.2-1. A minimum of two (2) decades overlap is observed between the source and intermediate ranges. The data is normalized to an estimated core power level of 10MW(t) at an intermediate range signal of 1.0×10^{-7} ampere. This estimate was made by observing a 0.19°F temperature increase across the core at this signal level at 532°F. This core differential temperature has been shown to be equivalent to a 10MW(t) heat addition to the reactor coolant system. Also plotted on Figure 4.2.-1 is the intermediate range detector response at power. The step change in intermediate range signal between 532°F and 579°F is due to a higher detector signal output for a given core power level caused by a reduction in reactor coolant hydrogen density and the reduction in the amount of boron per unit volume of reactor coolant.

4.2.4 CONCLUSIONS

Examination of Figure 4.2-1 shows that linearity, overlap and absolute output of the intermediate and source range detectors are within specifications and performing satisfactorily. There is at least two decades overlap between the source and intermediate ranges.

NUCLEAR INSTRUMENTATION OVERLAP

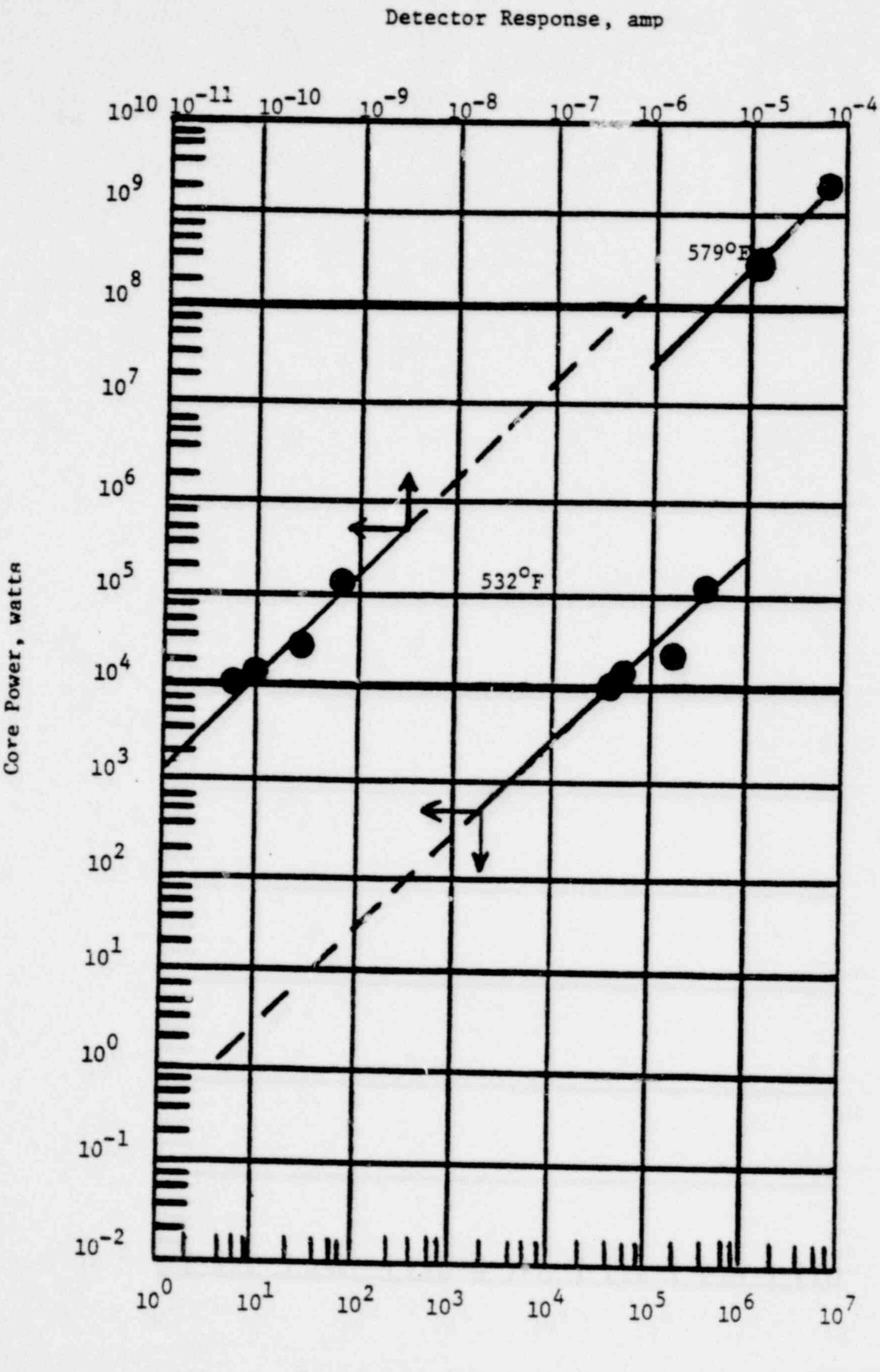


FIGURE 4.2-1

1414 067

4.3 REACTIVITY CALCULATIONS

4.3.1 PURPOSE

Reactivity calculations during the TMI Unit I test program were performed using the Reactimeter. After initial criticality and prior to the first physics measurement, an on-line functional check of the reactimeter was performed to verify its readiness for use in the test program.

4.3.2 TEST METHOD

Reactimeter is the name given to the Babcock and Wilcox reactivity-meter which solves the one-dimensional, inverse kinetics equation with six delayed neutron groups for core net reactivity based upon periodic samples of neutron flux. In addition to reactivity and neutron flux, the Reactimeter can also record 23 other analog and digital signals from the plant. The computational and data recording capability of the Reactimeter were used extensively throughout the test program.

After initial criticality and nuclear instrumentation overlap were established, intermediate range channel NI-4 was input to the reactimeter and the reactivity calculations were started. After steady state conditions with a constant neutron flux were established, a small amount of positive reactivity was inserted in the core by withdrawing control rod group 7. Stop watches were used to measure the doubling time of the neutron flux and the reactivity inserted was determined from period-reactivity curves. The measurement was repeated for several values of reactivity inserted by rod group 7, from $\pm 0.02\% \Delta k/k$ to $\pm 0.075\% \Delta k/k$. The reactivities determined from doubling time measurements were compared with the reactivity calculated by the reactimeter.

4.3.3 TEST RESULTS

The results of the reactimeter verification measurements are summarized in Table 4.3-1. In each case, the reactivity calculated by the reactimeter was well within the acceptance criteria limit of $\pm 2\%$ of the reactivity determined from doubling times.

4.3.4 CONCLUSIONS

An on-line functional check of the reactimeter was performed after initial criticality. The measured data shows that the core reactivity measured by the reactimeter was in good agreement with the values obtained from neutron flux doubling times.

1414 068

COMPARISON OF REACTIMETER AND DOUBLING TIME (DT) REACTIVITY MEASUREMENTS

Case No.	Measured[1] DT (Sec)	Measured[1] Reactivity (%Δk/k)	Calculated Reactivity (%Δk/k)	Percent Difference (%)
1	305	+0.0177	+0.0179	-1.11
2	-271	-0.0233	-0.0230	+1.29
3	110	+0.0440	+0.0442	-0.45
4	-150	-0.0470	-0.0478	-1.70
5	63	+0.0692	+0.0695	-0.43
6	-127	-0.0580	-0.0582	-0.34

- - - -

(1) Measured doubling times were determined from analyzing reactimeter traces of the neutron flux.

1414 069

TABLE 4.3-1

4.4 ALL RODS OUT CRITICAL BORON CONCENTRATION

4.4.1 PURPOSE

The all rods out critical boron concentration measurement is performed to obtain an accurate value for the excess reactivity loaded in the TMI Unit I core and to provide a basis for the verification of calculated reactivity worths. This measurement was performed at system conditions of 532°F and 2155 psig.

4.4.2 TEST METHOD

The reactor coolant system was borated such that control rod groups 1-6 and 8 were positioned at 100% withdrawn and group 7 was maintaining criticality at approximately 80% withdrawn. Once steady state conditions were established, control rod group 7 was withdrawn to 100% and the resultant reactivity change was measured. The measured boron concentration with group 7 partially inserted was then adjusted to the all rods out configuration using the result of the rod worth measurement to determine the reactivity worth, in terms of ppm boron, of the inserted control rods.

4.4.3 TEST RESULTS

The results of the measurement at 532°F are tabulated below.

ALL RODS OUT CRITICAL BORON CONCENTRATION

ppm boron

Moderator Temperature	Calculated Result	Measured Result
532°F	1634	1617

The measured boron concentration with group 7 positioned at 75% was 1609 ppm. An additional 8 ppm was added to this value that is derived from 0.084% $\Delta k/k$ due to group 7 withdrawal to 100%, using a differential boron worth of 1.058% $\Delta k/k$ per 100 ppm boron.

4.4.4 CONCLUSIONS

The above results show that the measured boron concentrations are in excellent agreement with predictions and are well within the acceptance criterion of $\pm$100 ppm.

1414 070

4.5 TEMPERATURE COEFFICIENT MEASUREMENTS

4.5.1 PURPOSE

The moderator temperature coefficient of reactivity can be positive, depending upon the soluble boron concentration in the reactor coolant. Because of this possibility, the Technical Specifications state that the moderator temperature coefficient shall not be positive at full power conditions. The moderator temperature coefficient cannot be measured directly, but it can be derived from the core temperature coefficient and a known fuel temperature (isothermal Doppler) coefficient at the zero power condition. For this reason, the temperature coefficient of reactivity was measured for several different boron concentrations at the zero power conditions of 532°F and 2155 psig to provide comparison of the moderator temperature coefficient with the design calculations prior to operation in the power range.

4.5.2 TEST METHOD

The technique used to measure the isothermal temperature coefficient at zero power was to first establish steady state conditions by maintaining reactor flux, reactor coolant pressure, turbine header pressure and core average temperature constant, with the reactor critical at approximately 3×10^{-9} amps in the intermediate range. (The measurement began with the reactor critical at a slightly higher flux level if a negative feedback effect was expected from a temperature increase or at a lower flux level if a positive feedback effect was expected from a temperature increase.) Equilibrium boron concentration was established in the reactor coolant system, make-up tank and pressurizer to eliminate reactivity effects due to boron changes during the subsequent temperature swings. The reactimeter and the brush recorders were connected to monitor selected core parameters with the reactivity value calculated by the reactimeter and the core average temperature displayed on an L&N two channel recorder.

Once steady state conditions were established, a positive heatup rate was started by closing the turbine bypass valves. After the core average temperature increased by about 10°F, core temperature and flux were stablized and the process was reversed by decreasing the core average temperature to the initial value by opening the turbine bypass valves. This procedure was completed two times at each boron concentration that the coefficient measurement was conducted to establish repeatability in the measured value. Calculation of the temperature coefficient from the measured data was then performed by dividing the change in core reactivity by the corresponding change in core temperature over a specific time period.

4.5.3 TEST RESULTS

Isothermal temperature coefficient measurements were conducted at four different reactor coolant boron concentrations during the zero power test program. The results of the measurements are summarized in Table 4.5-1 and in Figure 4.5-1. The calculated values are included for comparison. Good repeatability was demonstrated in all cases and the measured results compare favorably with the calculated values. All measured temperature coefficients of reactivity were within the acceptance criteria of $\pm 0.4 \times 10^{-4} \Delta k/k^{\circ}F$ of the predicted value. A calculation of the moderator coefficient indicates that it is well within the requirements of Technical Specification 3.1.7.

4.5.4 CONCLUSIONS

The measured values of the temperature coefficient of reactivity at 532°F, zero reactor power are within the acceptance criteria of $\pm 0.4 \times 10^{-4} \Delta k/k/^{\circ}F$ of the predicted value. Calculation of the moderator coefficient indicates that it is well within the limits of Technical Specifications 3.1.7.

1414 072

TABLE 4.5-1

SUMMARY OF TEMPERATURE COEFFICIENT MEASUREMENTS
AT THE ZERO POWER CONDITIONS OF 532°F AND 2155 PSIG

RC Boron Concentration (ppm)	Control Rod Position (% withdrawn)	Temperature Coefficient ($x10^{-4}\Delta k/k/^{o}F$) Measurement (I)	Measurement (II)	Average (I) & (II)	Calculated Results
1601	Gps 1-6 @100 Gp 7 @ 78 Gp 8 @100	+0.450	+0.447	+0.449	+0.488
1461	Gps 1-5 @100 Gp 6 @ 78 Gp 7 @ 0 Gp 8 @ 27	+0.306	+0.302	+0.304	+0.200
1269 (1)	Gps 1-3 @100 Gp 4 @ 95 Gps 5-7 @ 0 Gp 8 @ 27	-0.534 (2)	-0.520 (2)	-0.527	-0.710
1245	Gps 1-3 @100 Gp 4 @ 50 Gps 5-3 @ 0 Gp 8 @ 27	-0.605	-0.603	-0.604	-0.860

- - - - - - - - - - - - - - - - - -

(1) This is an average value based upon RC boron samples of 1261 ppm and 1276 ppm for measurements I and II, respectively.

(2) These results are from the heatup phase only.

1414 073

TEMPERATURE COEFFICIENT OF REACTIVITY

VS BORON CONCENTRATION

@ 532 F, 2155 PSI, 0 EFPD

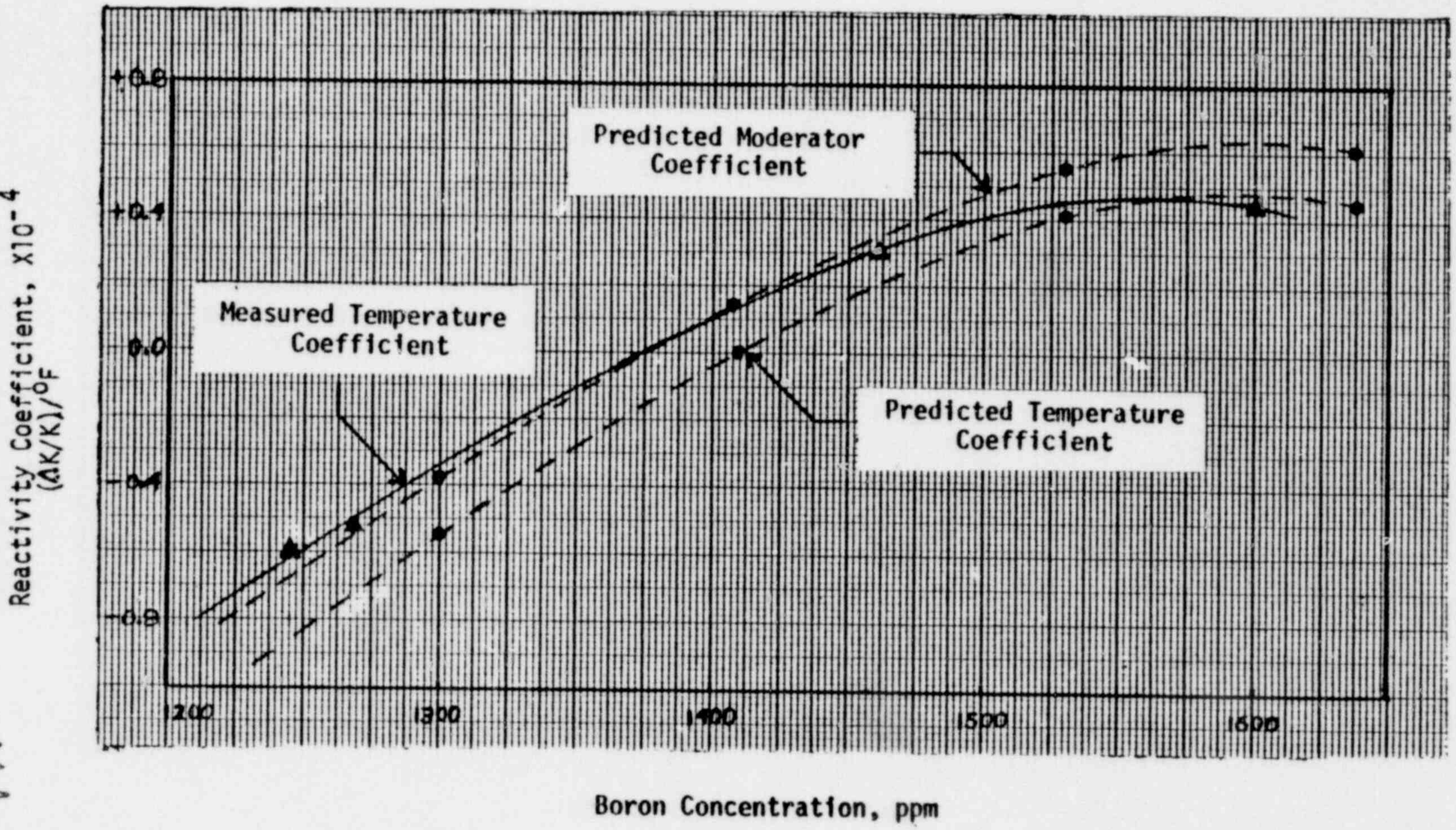


FIGURE 4.5-1

4.6 SOLUBLE POISON WORTH

4.6.1 PURPOSE

Soluble poison in the form of dissolved boric acid is added to the moderator to provide additional reactivity control beyond that available from the control rods. The primary function of the soluble poison control system is to control the excess reactivity of the fuel throughout each core life cycle. The differential reactivity worth of the boric acid in terms of ppm boron was measured during the zero power test.

4.6.2 TEST METHOD

Measurements of the differential boron worths at 532°F were performed in conjunction with the control rod worth measurements. The control rods worths were measured by the boron swap technique in which a boration/deboration rate was established and the control rods were withdrawn/inserted to compensate for the changing core reactivity. The reactimeter was used to provide a continuous reactivity calculation throughout the measurement. The differential boron worth was then determined by summing the incremental reactivity values measured during the rod worth measurements over a known boron concentration range. The average differential boron worth is the measured change in reactivity divided by the change in boron concentration.

4.6.3 TEST RESULTS

Measurements of the soluble poison differential worth were completed at the zero power condition of 532°F. The measured results are plotted in Figure 4.6.-1 along with the calculated differential worths. The measured results are within 1.25% of the calculated worths and within the acceptance criteria limits of ±.11% $\Delta k/k/100$ ppm. The results for only three out of five measurements are reported since the initial and final boron concentrations for two of the measurements were in question. The two results not reported, although within the acceptance criteria, are not considered representative.

4.6.4 CONCLUSIONS

The measured results for the soluble poison differential worth at 532°F were within 1.25% of the predicted values.

1414 075

DIFFERENTIAL REACTIVITY WORTH OF SOLUBLE POISON VS
BORON CONCENTRATION FOR MODERATOR TEMPERATURE OF 532°F

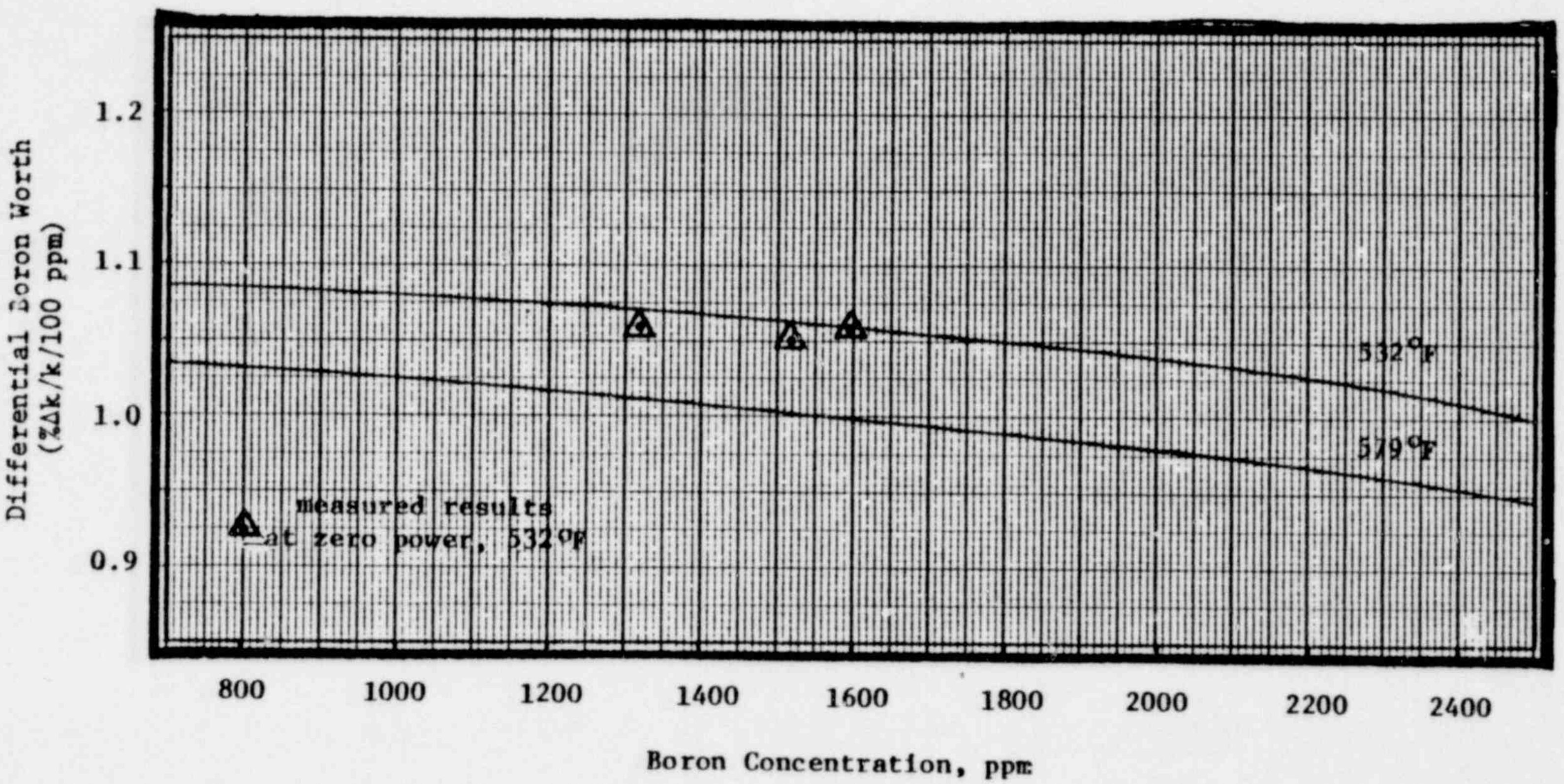


FIGURE 4.6-1

1414 076

4.7 CONTROL ROD GROUP WORTH MEASUREMENTS

4.7.1 PURPOSE

The total amount of excess reactivity controlled at beginning-of-life (BOL), hot (532 F), clean conditions is 17.8%Δk/k. During reactor operations, nearly all of the excess reactivity is controlled by the souble and lumped burnable poison systems. Additional control is provided by moveable control rods. This section provides comparison between the calculated and measured results for the control rod group worths.

The layout of the core according to the standard alphebet-numeric mesh showing the initial location of the control rod groups is shown in Figure 4.7-1. The number of control rods and the reactivity control function of each group is given in Table 4.7-1. The grouping of the control rods shown in Figure 4.7-1 will be used until the core burnup is 250 EFPD. At that time, an interchange between groups 4 and 7 will be made. Calculated and measured BOL control rod group reactivity worths for the normal withdrawal sequence were determined at reactor conditions of zero power, 532°F and 2155 psi. The calculated results were obtained using the PDQ code with either a two or three dimensional description of the core.

4.7.2 TEST METHOD

Control rod group reactivity worth measurements were performed at zero power, 532°F using the rod drop and boron/rod swap methods. The boron/rod swap method was used to measure the differential and integral reactivity worths of control rod groups 5 through 8 and parts of group 4. The total reactivity worth of rod groups 1 through 3 and part of group 4 was measured by the rod drop technique.

The boron swap method consisted of establishing a deboration rate in the reactor coolant system and compensating for the reactivity changes by inserting the control rod groups in incremental steps. In the rod swap technique (similar to the boron swap method), the reactivity changes caused by moving the rod group being measured are compensated for by moving another rod group. The reactivity changes that occurred during the measurements were calculated by the reactimeter and differential rod worths were obtained from the known reactivity worth versus the change in rod group position. The differential rod worths of each group were then summed to obtain the integral rod group worths.

In the rod drop method, critical equilibrium conditions were established with all the control rod groups to be measured withdrawn from the core. The control rod groups being measured were then tripped. The reactivity inserted in the core was calculated by the reactimeter. The total reactivity worth of rod groups 1 through 3 and part of group 4 was measured using the rod drop method.

4.7.3 TEST RESULTS

Control rod group reactivity worths were measured at the zero power, 532°F condition. The boron/rod swap method was used to determine differential and integral rod worths for control rod groups 5 through 8 from 100% to 0% withdrawn, and for group 4 from 100% to 50% withdrawn. The differential and integral worth of group 8 from 27% to 0% withdrawn was measured by the rod swap method using group 7. The rod drop method was used to obtain the total worth of groups 1 through 4 (group 4 from 50% to 0% withdrawn).

The results of the rod drop measurements on rod groups 1 through 4 are given in Table 4.7-2. Based on experience with previous startups, it was predicted that the rod drop measurements at TMI Unit I would yield values approximately 26% less than the correct value when considerably more than 1%Δk/k was being inserted. The deviation is caused by spatial flux changes in the core immediately after the control rods drop and its sign and magnitude are a function of the total amount of reactivity inserted and the detector-control rod geometry. The TMI results were consistent with these expectations, as seen in Table 4.7-2. When a correction factor of 1.35 is applied to the measured value, the measured and calculated worths agree to within 2%.

The integral reactivity worths for control rod groups 4 through 8 are presented in Figures 4.7-2 through 4.7-6. These curves (1) were obtained by integrating the measured differential worth curves. A third order polynomial expansion was used to obtain a "best fit" differential worth curve from the measured rod worth data. The point of maximum reactivity insertion for the group 8 rods occurred at 27% withdrawn (0%Δk/k/% withdrawn differential worth) with a total worth of -0.393%Δk/k at this position. The integral worth of group 8 from 27% to full insertion was measured at +0.215%Δk/k. The group 8 rods were positioned at 27% withdrawn during the reactivity worth measurements on group 1 through 7. Figure 4.7-7 is a plot of the total reactivity worth of groups 4 through 8 for the normal withdrawal sequence.

Table 4.7-3 provides a comparison between the predicted and measured results for the rod worth measurements. The calculated results were used as the best estimate for the worth of groups 1 through 4, based upon the rod drop results discussed above. The results show good agreement between the measured and predicted rod group worths. The maximum deviation between measured dand predicted was -8.33%. Also presented in Table 4.7-3 are the expected control rod group worths at 579°F with the APSRs at 27% withdrawn. These values were obtained by applying the percent deviation between the measured and predicted worths at 532°F to the predicted worths at 579°F.

4.7.4 CONCLUSIONS

Differential and integral control rod group reactivity worths were measured using the boron/rod swap and rod drop methods. The measured results at zero power, 532°F indicate good agreement with the predicted group worths.

- - - -

(1) Zero Power Physics rod worth data was processed by the Babcock and Wilcox computer in Lynchburg, Virginia to supplement on-site analysis. The results of that analysis are presented here as the best estimate of the measured control rod group reactivity worths.

TABLE 4.7-1

REACTIVITY CONTROL
FUNCTION OF CONTROL ROD GROUPS

Rod Group Number	Number Of Rods	Control Function
1	8	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	12	Power Doppler
6	8	Power Doppler
7	9	Transient Xenon
8	8	Axial Power Shaping
	69	

1414 079

TABLE 4.7-2

COMPARISON OF CALCULATED AND MEASURED CONTROL ROD GROUP REACTIVITY WORTHS FROM ROD DROP RESULTS

Moderator Temperature at 532°F, APSRs at 27% Withdrawn

<table>
<tr><th>Rod Group Number</th><th>Withdrawal Interval (% Withdrawn)</th><th>Calculated Worth (%Δk/k)</th><th></th><th>Uncorrected Meas. Worth (%Δk/k)</th><th>Corrected[1] Meas. Worth (%Δk/k)</th><th>Deviation From Calculated (%)</th></tr>
<tr><td>1</td><td>0-100</td><td>0.89</td><td rowspan="4">5.99</td><td rowspan="4">4.33</td><td rowspan="4">5.85</td><td rowspan="4">-2</td></tr>
<tr><td>2</td><td>0-100</td><td>3.01</td></tr>
<tr><td>3</td><td>0-100</td><td>0.74</td></tr>
<tr><td>4</td><td>0-49</td><td>1.35</td></tr>
</table>

(1) Corrected measured worth is based upon an expected 26% deviation between the measured and corrected results.

1414 080

TABLE 4.7-3

COMPARISON OF CALCULATED AND
MEASURED CONTROL ROD GROUP REACTIVITY WORTH

A. Moderator Temperature at 532°F, APSRs at 27% Withdrawn

Rod Group	Number Of Rods	Predicted Worth (%Δk/k)	Measured Worth (%Δk/k)	Percent Deviation[1] (%)
1	8	-0.89	-0.89	NA[3]
2	8	-3.01	-3.01	NA
3	8	-0.74	-0.74	NA
4	8	-1.86	-1.86	NA
5	12	-1.07	-1.03	-3.74
6	8	-1.22	-1.25	+2.46
7	9	-1.20	-1.10	-8.33
8	8	-0.38	-0.39	+3.42
Total	69	-10.37	-10.27	

B. Moderator Temperature at 579°F, APSRs at 27% Withdrawn

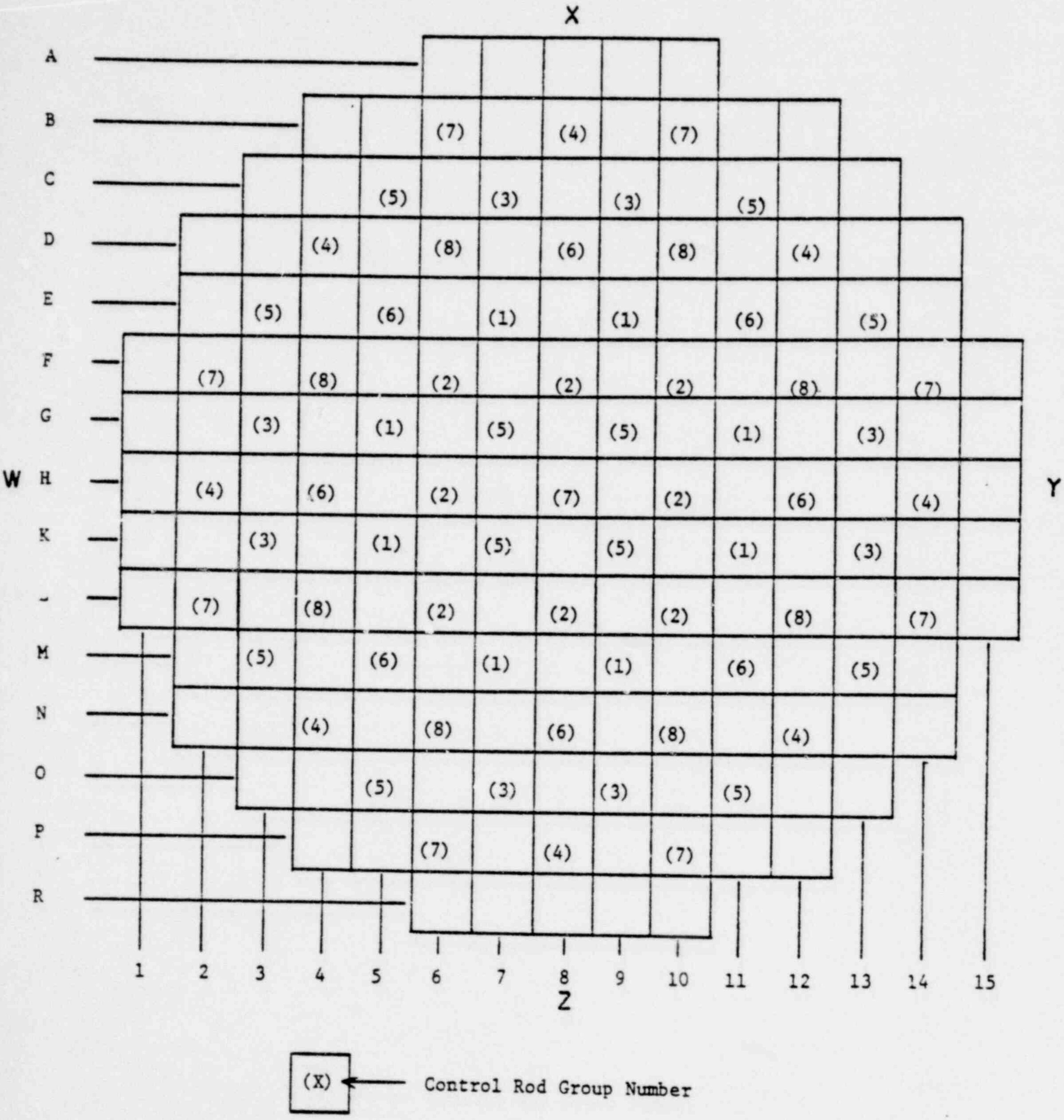
Rod Group	Number Of Rods	Predicted Worth (%Δk/k)	Expected Worth[2] (%Δk/k)	Percent Deviation[1] (%)
1	8	-1.40	-1.40	NA[3]
2	8	-3.59	-3.59	NA
3	8	-0.75	-0.75	NA
4	8	-1.47	-1.47	NA
5	12	-1.40	-1.35	-3.74
6	8	-1.48	-1.52	+2.46
7	9	-1.07	-0.98	-8.33
8	8	-0.44	-0.46	+3.42
Total	69	-11.60	-11.52	

(1) Percent deviation is calculated assuming predicted value is correct

(2) Expected worth is obtained by applying percent deviation between predicted and measured results at 532°F to the predicted results at 579°F

(3) NA denotes that percent deviation is not applicable since the predicted worths are used

CONTROL ROD GROUP LOCATIONS

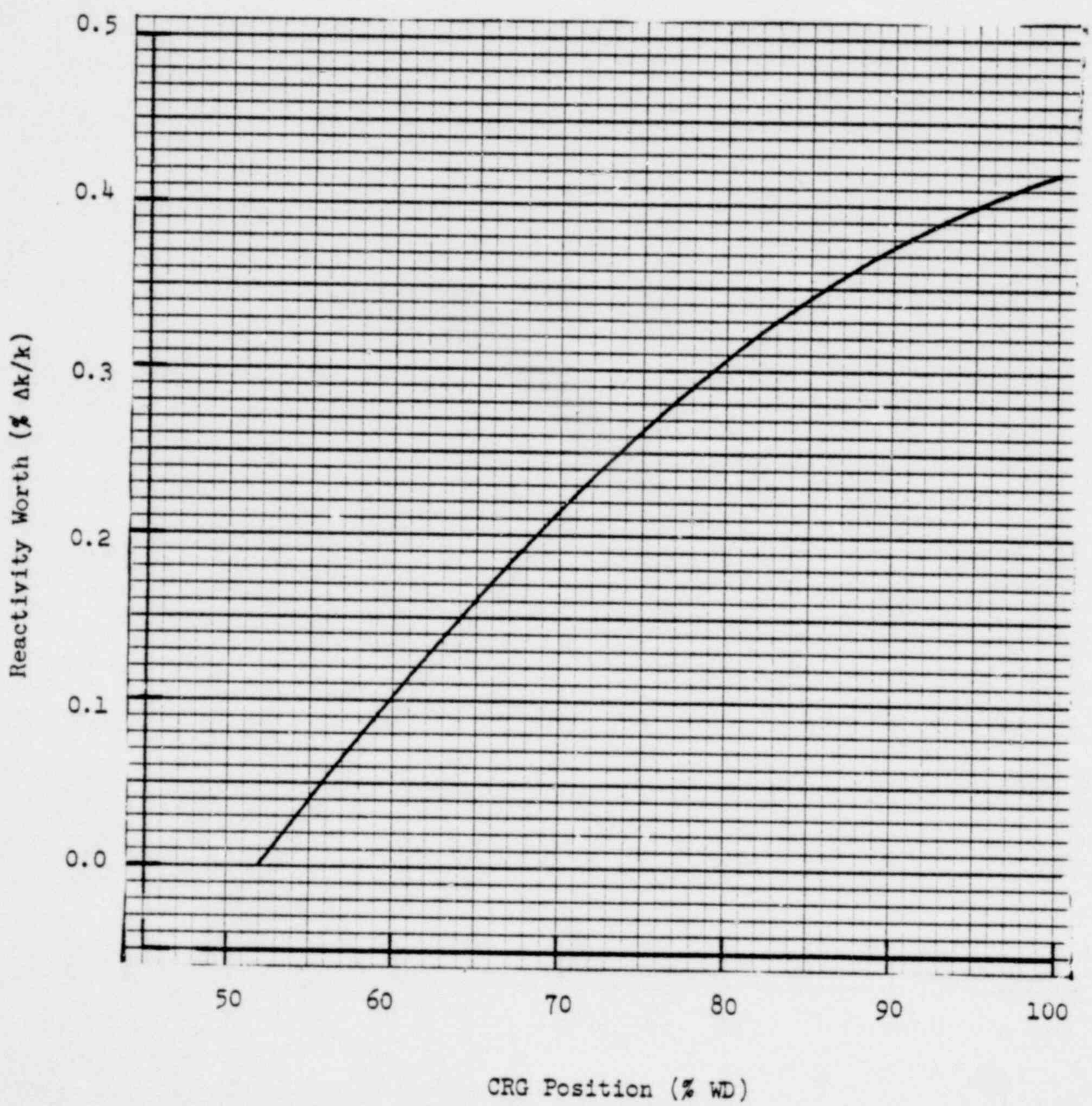


1414 082

Figure 4.7-1

Control Rod Group 4 Integral Worth
At Zero Power, 532°F, 0 EFPD

Total Worth = .419% Δk/k

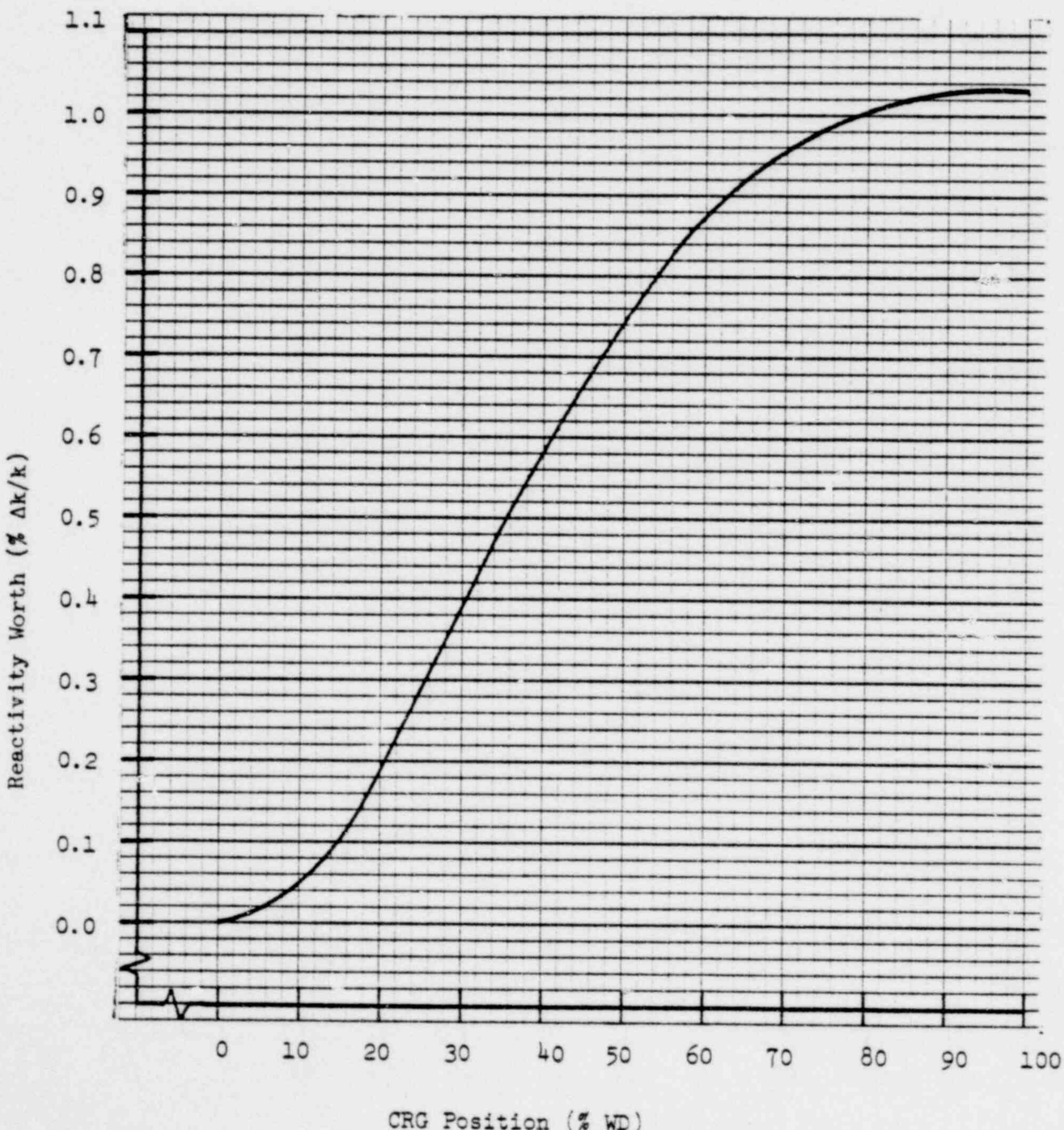


1414 083

FIGURE 4.7-2

Control Rod Group 5 Integral Worth
At Zero Power, 532°F, 0 EFPD

Total Worth = 1.03



1414 084

FIGURE 4.7-3

Control Rod Group 6 Integral Worth
At Zero Power, 532°F, 0 EFPD

Total Worth = 1.25% Δk/k

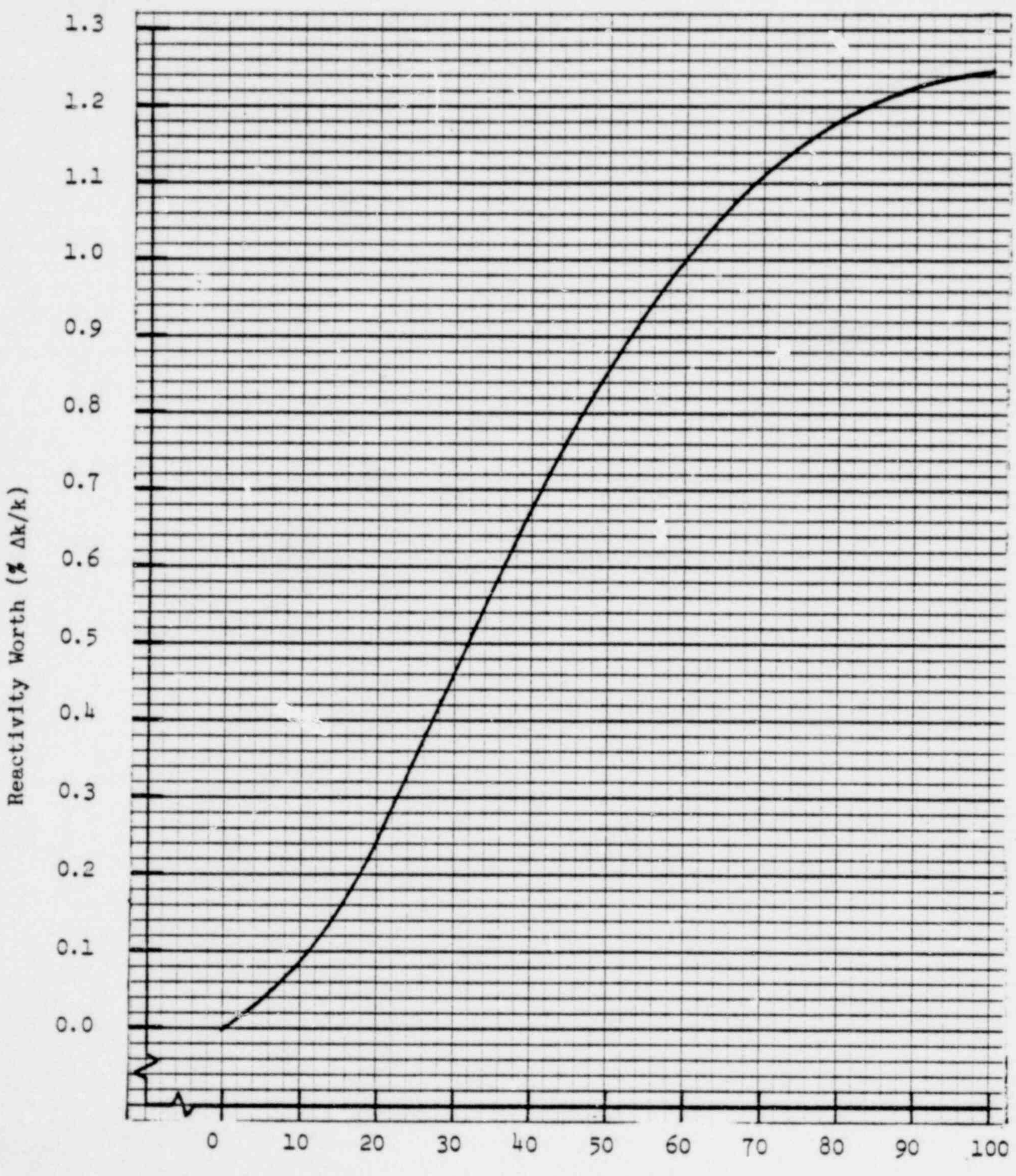
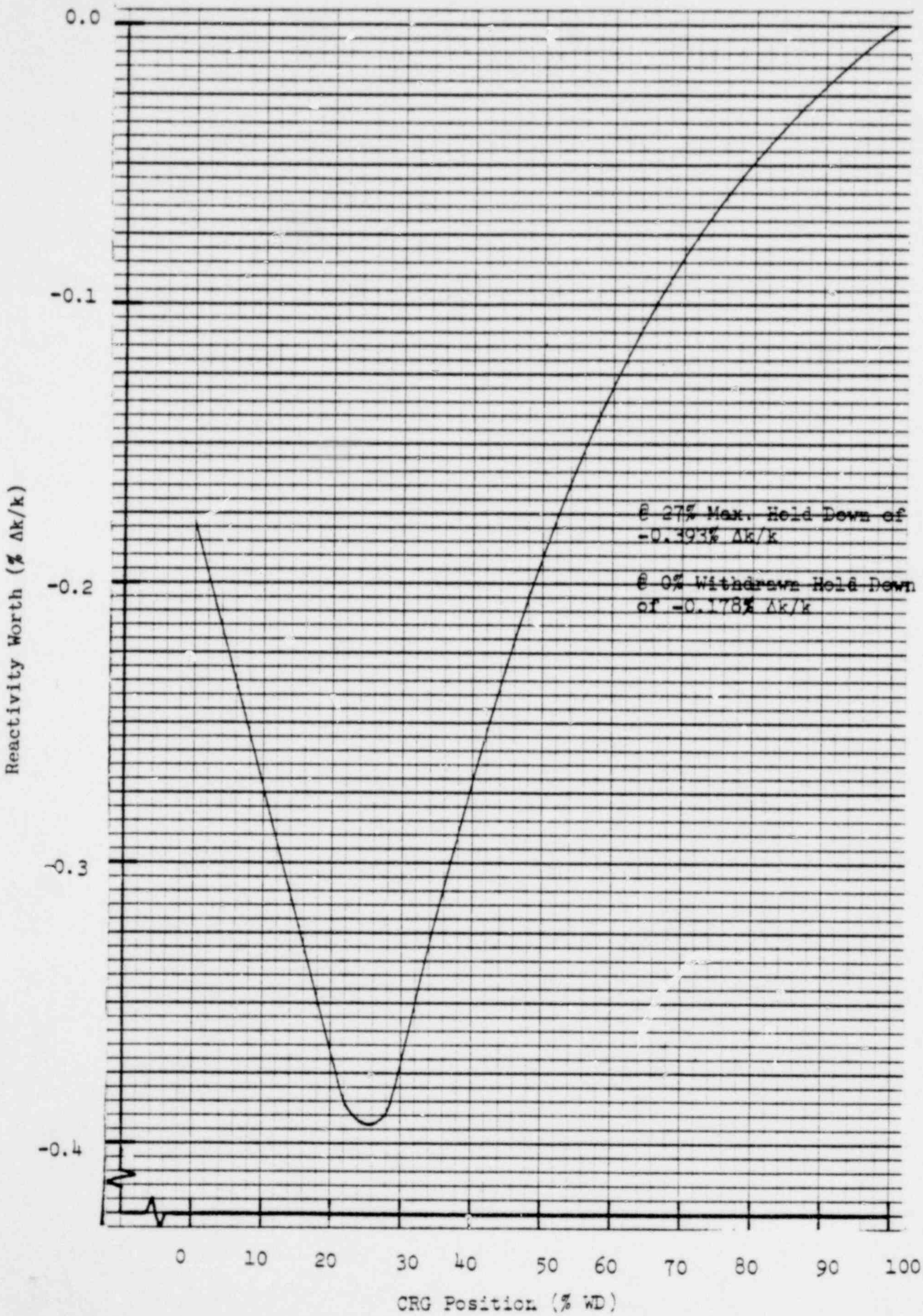


FIGURE 4.7-4

1414 085

Control Rod Group 8 Integral Worth
At Zero Power, 532°F, 0 EFPD



1414 086

FIGURE 4.7-6

Total Reactivity Worth Versus Rod Withdrawal
At Zero Power, 532°F, 0 EFPD

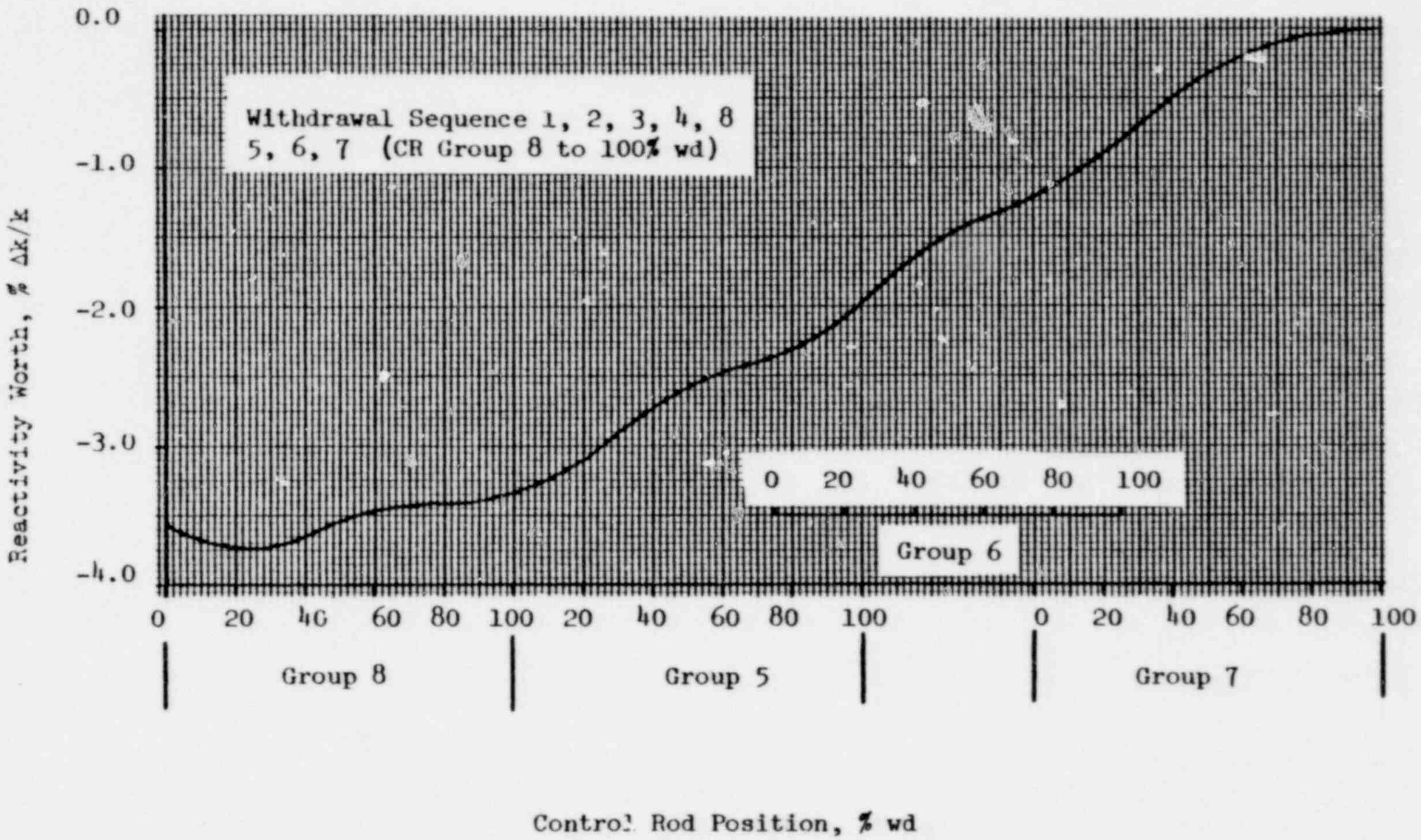


FIGURE 4.7-7

1414 087

4.8 EJECTED CONTROL ROD WORTH

4.8.1 PURPOSE

Technical Specification 3.5.2 states that the maximum worth of a single inserted control rod at zero power conditions of 532°F, 2155 psig shall not exceed 1.0% $\Delta k/k$. A pseudo ejected control rod worth measurement was performed during the zero power test program to verify the safety analysis calculations relating to the hypothetical ejection of the most reactive control rod.

4.8.2 TEST METHOD

Pseudo ejected control rod worths were measured at zero power using two different techniques. The first technique was the boron-swap method during which the boron concentration of the reactor coolant system was slowly and continuously increased. The pseudo ejected rod was withdrawn in quick steps to compensate for the reactivity inserted by the boration and the reactivity change was measured by the reactimeter. The sum of the incremental reactivity changes gives the total worth of the ejected rod. In the second technique, the rod drop method, critical equilibrium conditions were established with the pseudo ejected rod withdrawn to 100%. The ejected rod was then dropped into the core and the neutron flux and reactivity was logged by the reactimeter every 0.2 seconds. The measured instantaneous worth of the dropped rod is taken as the worth of the ejected rod.

4.8.3 TEST RESULTS

Pseudo ejected control rod reactivity worth was measured at the zero power conditions of 532°F, 2155 psig. Rod worth calculations performed for several Doppler and Transient control rods indicated that core location F-2 (and those locations symmetrical to it) was the highest worth rod position. Control rod 8 in group 7 was selected for the ejected rod measurement. Figure 4.8-1 shows the location of rod 7-8 in the core.

Critical equilibrium conditions were established for the boron-swap measurement with an initial RCS boron concentration of 1269 ppm and control rod group 5 at 6%, groups 6 and 7 at 0% and group 8 at 27.5% withdrawn. Control rod 7-8 was withdrawn to 100% to compensate for borating the reactor coolant to 1337 ppm. The worth of rod 7-8 from this measurement was 0.688%$\Delta k/k$.

In the rod drop method, the reactor was just critical with rod group 5 at 13% withdrawn, group 8 at 27.5% and groups 6 and 7 at 0% withdrawn. Control rod 7-8 was at the 100% withdrawn position. Under these conditions, rod 7-8 was dropped into the core and its resultant reactivity worth was obtained from the reactimeter. The worth of the ejected rod by the rod drop method is 0.664%$\Delta k/k$, which compares well with the boron swap result. The calculated and measured results are compared in Table 4.8-1.

4.8.4 CONCLUSIONS

Two different methods were used to measure the pseudo ejected rod worth at zero power, 532°F. The results from the boron-swap and the rod drop techniques compare favorably. The best estimate for the measured value, 0.688%$\Delta k/k$, is below the calculated worth, but this is more conservative with respect to an ejected rod accident. The Technical Specification requirement that the value not exceed 1.0% $\Delta k/k$ is satisfied.

TABLE 4.8-1

COMPARISON OF PREDICTED AND MEASURED PSEUDO EJECTED ROD WORTHS AT THE ZERO POWER, 532°F CONDITION

I. Calculated

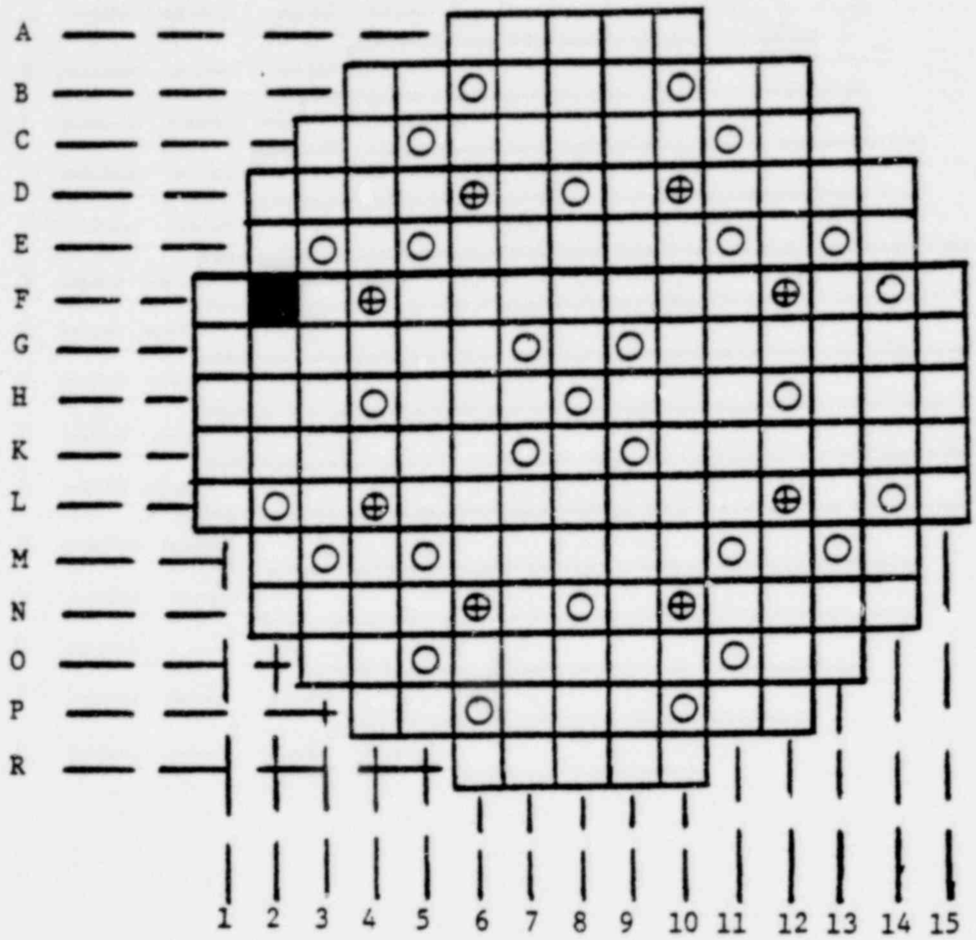
Rod Positions (% withdrawn)	Reactivity Worth (% $\Delta k/k$)
Groups 1-4 @100% Groups 5-7 @ 0% Group 8 @37.5%	1.0%

II. Measured

Method	Rod Positions (% withdrawn)	Reactivity Worth (% $\Delta k/k$)
Boron-Swap	Groups 1-4 @100% Group 5 @ 6% Groups 6-7 @ 0% Group 8 @27.5%	0.688%
Rod Drop	Groups 1-4 @100% Group 5 @ 13% Groups 6-7 @ 0% Group 8 @27.5%	0.664%

1414 089

CONTROL ROD LOCATIONS FOR EJECTED ROD MEASUREMENT AT 532 F, 2155 PSI, 0 EFPD



⊕ APSR Location

○ CR Groups 5, 6, 7

Rod Ejected	Calculated Ejected Rod Worth	Rods Inserted	Measured Ejected Rod Worth
F-2	1.0%Δk/k	○,⊕ Doppler, Transient and APSR Groups	0.688%Δk/k (1) 0.664%Δk/k (2)

(1) Boron Swap Result (2) Rod Drop Result

1414 090

FIGURE 4.8-1

4.9 SHUTDOWN MARGIN

4.9.1 PURPOSE

Technical Specification 3.5.2 states that the available shutdown margin shall not be less than 1%Δk/k with the most reactive control rod stuck out of the core. The purpose of the stuck rod worth measurement at zero power, 532°F was to verify that the calculated stuck rod worths are conservative compared to the measured value.

4.9.2 TEST METHOD

The minimum available shutdown margin and the worth of a simulated stuck control rod were measured by performing two rod drop measurements. In the first measurement, all control rods not in the core were dropped (except APSRs). The simulated stuck rod was then withdrawn to 100%, and the remainder of the safety groups were then withdrawn to establish critical equilibrium conditions at the same boron concentration as the first measurement. All control rods except the stuck rod and the APSRs were then tripped. The difference in the reactivity inserted in the two measurements was taken as the stuck control rod worth. The minimum shutdown margin available was obtained directly from the second rod drop.

4.9.3 TEST RESULTS

The most reactive control rod at zero power, 532°F was calculated to be rod 7 in group 4 (core location H-2) and those control rods symmetrical to it. Rod 4-7 was selected for the stuck rod measurement and all drops were made with the APSRs at 26% withdrawn. Figure 4.9-1 shows the core location of control rod 4-7. For the first drop, critical equilibrium conditions were established with group 4 at 49% withdrawn at a 1197 ppm boron concentration. The reactivity inserted in the core by dropping all the rods was 4.33%Δk/k. For the same boron concentration, control rod 4-7 was withdrawn to 100% and critical conditions were established with group 3 at 85% withdrawn. For the second drop, all control rods except rod 4-7 were tripped. The reactivity measured from inserting all control rods except the simulated stuck rod was 2.16%Δk/k. The reactivity measured for the second drop gives the minimum shutdown margin available with the most reactive control rod stuck out. The difference in measured reactivity inserted in the two drops is the measured worth of the stuck rod.

Correction factors were applied to the measured reactivity value from the reactimeter to correct for changes in the spatial flux distribution immediately after the rod drops. Table 4.9-1 lists the corrected and uncorrected measured worths and provides comparison with the predicted worths for the stuck rod. The corrected measured results for drops one and two are 5.85%Δk/k and 2.01%Δk/k, respectively. This results in a stuck rod worth value of 3.84%Δk/k, which compares favorably with the predicted value of 3.91%Δk/k. The minimum available shutdown margin with the most reactive control rod stuck out of the core was measured to be 2.01%Δk/k. If a 10% uncertainty is assigned to the measurement, the minimum shutdown margin is at least 1.8%Δk/k and this ensures that the Technical Specification requirement is met.

1414 091

4.9.4 CONCLUSIONS

Minimum shutdown margin verification and stuck control rod worth measurements were completed for the zero power condition at 532°F. The measured value of the most reactive control rod stuck out of the core with all other control rods inserted (except APSRs) was 3.84%$\Delta k/k$. The shutdown margin available under this condition was at least 1.8%$\Delta k/k$ which guarantees that the Technical Specification limit of 1.0%$\Delta k/k$ is satisfied.

1414 092

TABLE 4.9-1

MEASURED AND CALCULATED WORTH OF STUCK CONTROL ROD AT ZERO POWER, 532 F

Drop No.	Rod Groups Dropped(1)	Uncorrected Measured Worth (%Δk/k)	Correction Factor	Corrected Measured Worth (%Δk/k)	Predicted Worth (%Δk/k)
1	1, 2, 3 & 4 (4 at 49%)	4.33	1.35	5.85	5.94
2	1, 2 & 3 (3 at 85%, Rod 4-7 remains at 100%)	2.16	0.93	2.01	2.03(2)
			Stuck Rod Worth	3.84	3.91

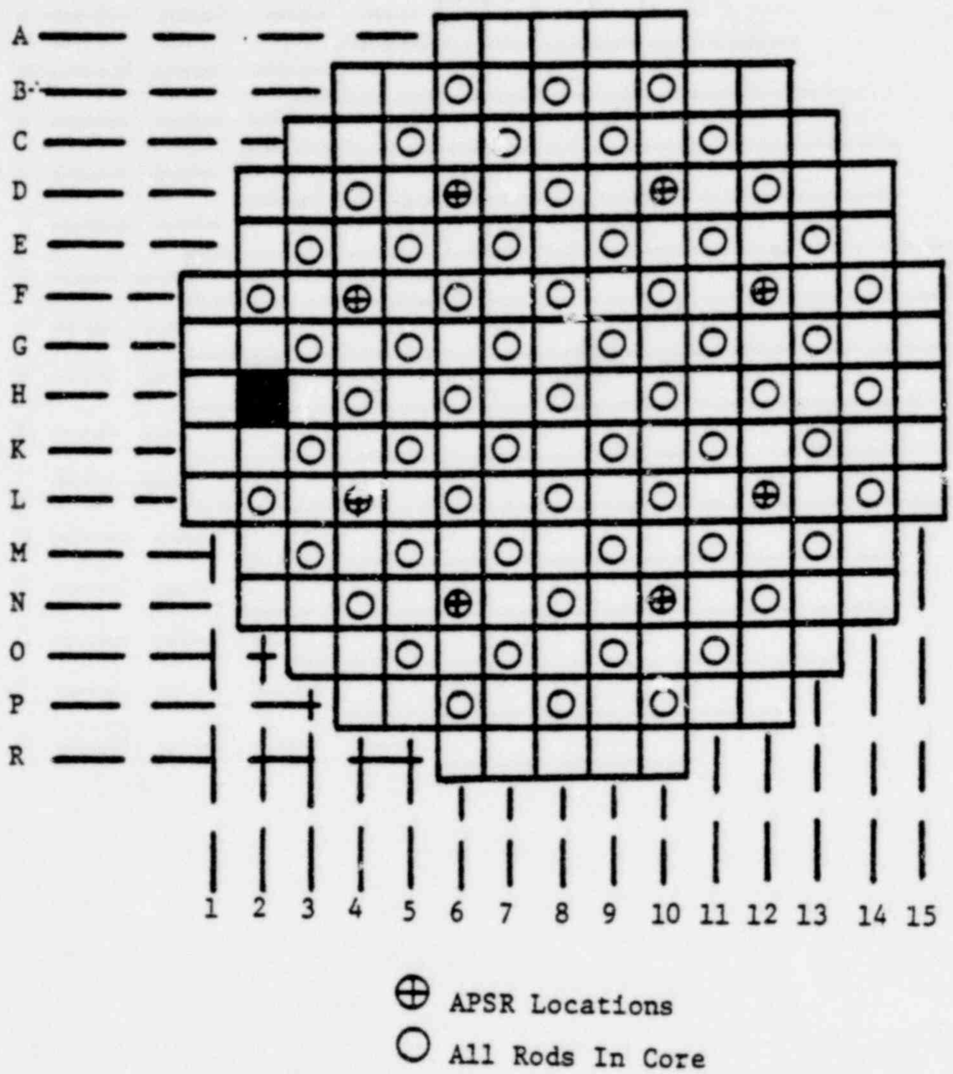
\- - - - -

(1) APSRs were at 26% withdrawn for both measurements

(2) Based upon Stuck Rod Worth of 3.91% Δk/k

1414 093

CONTROL ROD LOCATION FOR STUCK ROD
MEASUREMENT AT 532 F, 2155 PSI, 0 EFPD



Stuck Control Rod	Corrected Drop 1 Results (All Rods)	Corrected Drop 2 Results (All but 4-7)	Measured Rod Worth (%Δk/k)	Predicted Rod Worth (%Δk/k)
CRA 4-7 (H-2)	5.85	2.01	3.84	3.91

1414 094

FIGURE 4.9-1

5.0 CORE PERFORMANCE - MEASUREMENTS AT POWER

This section presents the results of the physics measurements that were conducted with the reactor at power. Testing was conducted at the four major power plateaus of 15%, 40%, 76% and 100% of 2535 megawatts thermal core power, as determined from primary and secondary calorimetric measurements. Operation in the power range began on June 15, 1974. Power escalations occurred as the required testing at each plateau was successfully completed.

Periodic measurements and calibrations were performed on the plant nuclear instrumentation during the escalation to full power. The four power range detector channels were calibrated based upon primary and secondary plant heat balance measurements. Testing of the incore nuclear instrumentation was performed to ensure that all detectors were functioning properly and that the detector outputs were processed correctly by the plant computer. Core axial imbalance determined from the incore instrumentation system was used to calibrate the out of core detector imbalance indication. Radiation surveys of the biological shield and reactor and auxiliary buildings were conducted to obtain base line data on accessible work areas while the reactor is operating at power.

The major physics measurements performed during power escalation consisted of determining the moderator and power Doppler coefficients of reactivity, determining the worth and associated power distributions effected by simulated dropped and ejected control rods, and obtaining detailed radial and axial core power distribution measurements for several core axial imbalances. Values of minimum DNBR and maximum linear heat rate were monitored throughout the test program to ensure that core thermal limits would not be exceeded.

A summary of the tests reported in this section, including the respective section number and power level at which they were performed, is given in Table 5.0-1. The core power history and integrated burnup up to August 27, 1974 is presented in Figures 1.0-1 and 1.0-2, respectively.

1414 095

TABLE 5.0-1

SUMMARY OF TESTS REPORTED IN SECTION 5.0

Section Number	Title of Section - Test Procedure Number	Test Power Levels, % FP ≈5	15	25	35	40	50	65	76	85	95	100
5.1	Biological Shield Survey - TP 800/3					X						X
5.2	NI Calibration at Power - TP 800/2	X	X			X	X		X			X
5.3	Incore Detector Testing - TP 800/24		X	X	X	X			X			
5.4	Power Imbalance Detector Correlation - TP 800/18					X			X			
5.5	Rod Reactivity Worth Measurements - TP 800/20					X			X			X
5.6	Reactivity Coefficients at Power - TP 800/5					X			X			X
5.7	Dropped Control Rod Test - TP 800/31					X			X			
5.8	Pseudo Ejected Control Rod Test - TP 800/33					X						
5.9	Core Power Distributions - TP 800/11		X			X			X			X
5.10	NSS Heat Balance - TP 800/22	X	X	X	X	X	X	X	X	X	X	X
5.11	Reactivity Depletion Versus Burnup - TP 800/16											X
5.12	Neutron Noise Measurements					X			X			X

1414 096

5.1 BIOLOGICAL SHIELD SURVEY

5.1.1 PURPOSE

The purpose of the biological shield survey was to measure the radiation levels in all accessible locations of the plant adjacent to the biological shield and to obtain base line radiation levels for comparison with future measurements of radiation levels during plant operation.

5.1.2 TEST METHOD

The biological shield survey was conducted at zero reactor power and at 40% and 100% of full power. The Reactor Building outside of the biological shield or areas designated as access areas were marked off in horizontal and vertical zones and readings were taken in discrete sections. All areas in the Auxiliary Building were also surveyed. The surveys were conducted using portable ionization and GM counters for gamma radiation and BF_3 counters for neutrons. All readings were taken within one inch of the shield wall. Readings were taken after fifteen hours of steady state operation at the specified power was attained.

5.1.3 TEST RESULTS

The results of the biological shield survey at each power level where the test was conducted are summarized in the table below.

Date	Power Level	Gamma/Neutron Average (mRem/hr.)	Gamma/Neutron Maximum (mRem/hr.)
6/6/74	0%	<0.03/0	0.15/0
7/1/74	40%	<2.34/1.0	14/4.8
8/8/74	100%	<5.88/8.7	40/60

The above results apply to the inside of the Reactor Building only in those areas outside of the biological shield. The maximum radiation level measured was 60mr/hr neutron at elevation 365', which is above the shield area.

5.1.4 CONCLUSIONS

The maximum radiation levels found in all accessible areas were below 100mRem/hr, and therefore, the biological shield meets all design criteria.

1414 097

5.2 NUCLEAR INSTRUMENTATION CALIBRATION AT POWER

5.2.1 PURPOSE

The purpose of Nuclear Instrumentation Calibration at Power was to calibrate the power range nuclear instrumentation indication to within ±2% FP of the reactor thermal power as determined by a heat balance and to within ±5 percent incore axial offset as determined by the incore monitoring system. Additional purposes during the power escalation program were as follows:

(a) To adjust the high power level trip setpoint when required by the power escalation procedure.

(b) To verify that at least one decade overlap exists between the intermediate and power range nuclear instrumentation.

Two acceptance criteria are specified for nuclear instrumentation calibration at power as listed below.

(1) The power range nuclear instrumentation indicates power level within ±2% FP of the power level determined by heat balance and within ±5 percent of the incore axial offset as determined by the incore detectors.

(2) The high power level trip bistable is set to trip at the desired value, ±0.5% FP.

5.2.2 TEST METHOD

As required during power escalation, the top and bottom linear amplifier gains were adjusted to maintain power range nuclear instrumentation channel power indication within ±2% of the power calculated by a heat balance. During top and bottom linear amplifier gain adjustment, the ratio of their gains was maintained constant as long as the indicated axial imbalance was within ±5% of incore imbalance; if not, their gains were adjusted to correct imbalance and heat balance mismatch at the same time.

Data was also taken to verify overlap between the intermediate and power range channels. The required overlap was a minimum of one decade between these two nuclear instrumentation ranges.

When directed by the power escalation procedure and/or the unit startup procedure, the high flux trip bistable setpoint was adjusted. The major settings during power escalation are given below:

Test Plateau %FP	Bistable Setpoint %FP
15	50
50	60
76	95
100	104.75

1414 098

5.2.3 TEST RESULTS

An analysis of test results indicated that changes in Reactor Coolant System boron and xenon buildup or burnout affected the power as observed by the nuclear instrumentation. This was as expected since the power range nuclear instrumentation measures reactor neutron leakage which is directly related to the above changes in system conditions. Changes in these system conditions resulted in a nuclear power range indication increase or decrease of approximately 5 to7%FP. Each time that it was necessary to calibrate the power range nuclear instrumentation, the acceptance criteria of calibration to within ±2.0%FP of the heat balance power was met without any difficulty. Also, each time it was necessary to calibrate the power range nuclear instrumentation, the ±5% axial offset criteria as determined by the incore monitoring system was also met. Table 5.2-1 is a summary of the data taken during calibration at different power levels during power escalation testing. In all cases, the nuclear instrumentation was adjusted to within ±2.0%FP of the heat balance and to within ±5% incore axial offset.

The high flux trip bistable was adjusted to 50, 60, 95 and 104.75% FP prior to escalation of power to 15, 50, 76 and 100% FP, respectively. Acceptance criteria of adjusting the setpoint to the above values within ±0.5% FP was met each time without difficulty.

The overlap measured during the startup program included the total span of the power range, exceeding the one-decade overlap requirement. Figure 5.2-1 shows the overlap of all three nuclear instrumentation channels.

5.2.4 CONCLUSIONS

The power range channels were calibrated to within two percent of heat balance power several times during the startup program. These calibrations were required due to power level, boron, and/or control rod configuration changes during the program. Acceptance criteria for nuclear instrumentation calibration at power were met in all instances.

1414 099

SUMMARY OF NUCLEAR INSTRUMENTATION CALIBRATION
AT POWER RESULTS PERFORMED DURING POWER ESCALATION

Heat Balance Power (%FP)	Incore Imbalance (%FP)	Power Before and After Calib., % FP				Imbalance Before and After Calib., % FP			
		NI-5	NI-6	NI-7	NI-8	NI-5	NI-6	NI-7	NI-8
12.96	-----	16.87	16.87	17.0	16.5	-7.53	-4.00	-4.44	-4.39
12.96	-----	12.2	13.0	13.0	14.0	-5.83	-3.28	-3.15	-3.70
30.40	-----	28.0	27.6	27.5	27.5	-7.63	-2.75	-3.13	-3.38
30.40	-----	29.1	29.0	28.5	29.3	-8.00	-3.13	-3.25	-3.88
40.05	1.42	39.1	39.6	38.8	40.0	-2.88	3.31	1.84	2.81
40.01	0.47	40.1	39.5	39.7	39.0	0.53	-0.91	-1.05	-0.94
75.95	0.59	76.8	77.1	76.9	76.9	3.90	4.50	3.20	4.00
76.40	1.38	76.0	76.6	76.1	76.6	0.40	0.60	1.10	0.40
85.85	1.30	83.0	84.2	82.6	84.0	-3.70	-2.60	-2.00	-3.07
85.85	1.30	85.5	84.1	85.5	85.1	0.30	2.00	1.20	0.60
95.19	-1.30	93.6	91.7	93.7	92.9	-3.70	-1.90	-2.40	-3.30
95.30	-2.21	96.0	94.4	95.5	94.9	0.00	0.20	-0.30	0.30
99.6	-2.46	99.0	98.0	99.0	99.0	0.34	0.41	-0.16	0.59
99.6	-2.46	99.0	98.0	99.0	99.0	0.34	0.41	-0.16	0.59

TABLE 5.2-1

NUCLEAR INSTRUMENTATION FLUX RANGES

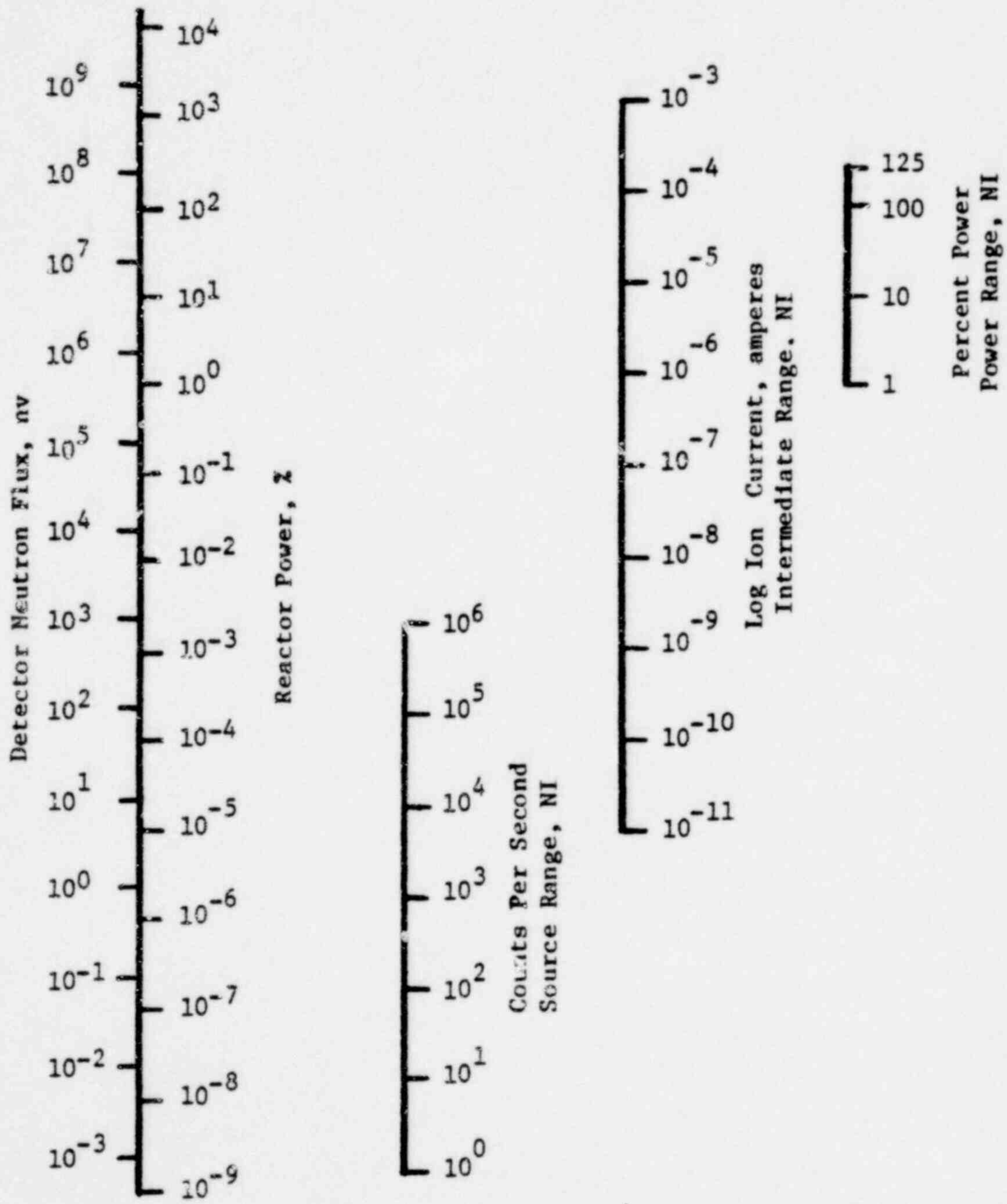


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Figure 5.2-1

5.3 INCORE DETECTOR TESTING

5.3.1 PURPOSE

A system of self-powered-neutron-detectors (SPNDs) is installed in the TMI Unit I reactor core. These detectors monitor core power density within the core and their outputs are monitored and processed by the plant computer to provide accurate readings of relative neutron flux. Although the incore detector system serves no safety related function, it does provide detailed core power distribution data which will be used throughout core life for physics and fuel performance calculations. In addition, information from these detectors formed an integral part of the startup test program.

Tests conducted on the incore detector system during power escalation were performed to:

(a) Verify that the output from each detector and its response to increasing reactor power was as expected.

(b) Verify that the background, length and depletion corrections applied by the plant computer are correct.

(c) Calibrate the backup incore recorder.

5.3.2 TEST METHOD

5.3.2.1 Incore Detector System

Power distribution within the core is measured at 364 locations (7 axial positions in 52 fuel assemblies) by the Incore Detector System's self-powered neutron detector. The 52 incore monitor assemblies are placed at preselected radial positions as shown in Figure 5.3-1. Seventeen detector assemblies are positioned to act as symmetry monitors and the remaining 35 assemblies, with 5 of the 17 symmetry monitors, monitor every other fuel assembly position assuming quatercore symmetry. Each assembly contains seven equally spaced flux detectors corresponding to seven axial core elevations to provide measurement of axial flux shape.

The self-powered incore detectors use rhodium wire detectors which undergo electron emmission when placed in a neutron environment. The capture of a neutron by rhodium - 103 produces the radioactive isotope rhodium - 104. The radioactive decay of rhodium - 104 emits a beta particle and creates a daughter product that requires one more orbital electron than the parent. This orbital electron is the source of the self-powered detector signal when the only free electron path to the detector is an electrical conductor in series with a current measuring instrument. Figure 5.3-2 shows the basic components of the self-powered neutron detector.

Since beta decay of rhodium - 104 is not the only source of electrons in the core, the current measured from the individual rhodium detectors must be corrected for a background current. The background current is measured by background detectors in each of the 52 incore monitor assemblies.

The outputs from the detectors are read and processed by the plant computer. The computer applies background, as built manufacturing and depletion correction factors to each detector reading. The incore detector information is then used in core physics calculations and/or provided for display to the reactor operators. The

readings from 36 incore detectors are also monitored by two 24 point recorders. The recorders provide power distribution information to the operator at times the plant computer is not available.

5.3.2.2 Incore Detector Tests

The response of the incore detectors versus power level was determined and a comparison of the symmetrical detector outputs made at reactor powers of 15, 25 35 and 40% FP. Once steady state conditions were achieved at each of the above power levels, a printout of corrected and uncorrected SPND maps for all detectors was obtained from the plant computer. Figures 5.3.-3 and 5.3-4 are sample printouts of the uncorrected and corrected maps, respectively. The corrected and uncorrected rhodium detector readings and the background readings, in units of nanoamps, were then plotted versus reactor power level to verify that each detector was responding as expected. The values of all symmetrical detectors were compared to verify that they were the same within allowable deviations.

The plant computer applies background, as built manufacturing and depletion correction factors to each rhodium detector, as mentioned above. Hand calculations were performed at 40% FP to verify the computer calculated corrections using uncorrected SPND outputs and SPNDI values from performance data output segment number 6. SPNDI is the accumulated nanoamp sum for each detector.

Data was collected to calibrate the backup incore recorder at 40% and 76% FP. The readings from two 24 point recorders located in the control room were recorded while obtaining a printout of the corrected detector readings from the plant computer. The recorder indication was then adjusted to agree with the corrected detector outputs.

5.3.3 TEST RESULTS

Incore detector testing during the startup program indicates that the detectors are responding as expected. Typical results of the tests are shown in Tables 5.3-1 and 5.3-2 and Figure 5.3-5.

Tables 5.3-1 and 5.3-2 show the comparison made between two sets of symmetical detectors at 40% FP. Corrected SPND values for each detector at all seven axial levels were recorded and the highest and lowest detector readings at each level were determined. The acceptance criteria for the test required that the difference between the average value and the highest and the lowest detector readings be within 5% of the average value for a given level. The 5% acceptance criteria was met in almost all cases; however, some detector readings were greater than 5% from the average value. The maximum difference between a detector reading and the average value was 11.6%.

The differences that were observed between the symmetrical detector readings were attributed to errors caused by a loss of some of the detector signal. The amount of signal loss varies for each SPND, but is proportional to the value of the dropping resistors used to measure the detector current. The results of measurements performed at 76% FP indicate that a reduction in the size of the dropping resistor will decrease the error associated with the measured signal. Plans are in progress to replace the present resistors.

Both the corrected and uncorrected readings from each SPND and the background detector signals were plotted at 15%, 25%, 35% and 40% FP. An example of the response of an SPND with increasing reactor power is shown in Figure 5.3-5. The outputs of the SPNDs increase linearly with power, as expected, and show some variations for different core locations and control rod position.

Hand calculations were performed at 40% FP to verify the length, background and depletion correction factors applied to each SPND by the plant computer. The hand calculations were performed for detector string 1, 7, 35 and 52 by obtaining the SPNDI (accumulated detector burnup) values from the plant computer and then applying the appropriate length, background and depletion corrections. The correction factors derived from the hand calculations agreed to within 2% of the computer calculated values.

Calibration of the backup incore recorders was performed at 40% and 76% FP. The 36 detector signals were recorded from the backup recorders at the same time that corrected detector readings were obtained from the plant computer. A hand calculation of $(P/\bar{P}c)$ was performed from the computer data and these values were factored into the adjustments made to the backup recorder readings. The advantage of using this technique is that in additon to having a readout of relative neutron flux on the backup recorder, the information displayed also incicates flux peaking in the core. A sample calculation for the backup incore calibration is given in Tables 5.3.-3 through 5.3-5.

5.3.4 CONCLUSIONS

Incore detector testing during power escalation demonstrated that all detectors were functioning as expected that symmetrical detector readings agreed within acceptable limits and that the computer applied correction factors are accurate. The backup incore recorder was calibrated and operational above 80% FP as required by the Technical Specifications.

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SYMMETRICAL DETECTOR COMPARISON

6/30/74 40% FP

Det. No.	Core Location	Corrected Nanoamps						
		1	2	3	4	5	6	7
5	E-9	331	500	486	418	376	304	154
7	E-7	334	502	496	423	392	298	144
9	G-5	329	502	489	415	400	318	156
11	K-5	327	495	481	420	396	315	153
13	M-7	322	497	485	414	375	311	156
16	M-9	344	505	493	422	391	303	142
19	K-11	330	502	487	417	393	308	152
25	G-11	331	505	480	421	390	308	154
	Average	331	501	487	419	389	308	151
	Highest	344	505	496	423	400	318	156
	Lowest	322	495	480	414	375	298	142
	(1) Difference	13	4	9	4	11	10	4
		9	6	7	9	25	10	6

(1) Difference is taken between the average value and the highest and the lowest readings.

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TABLE 5.3-1

SYMMETRICAL DETECTOR COMPARISON

6/30/74 40% FP

Det. No.	Core Location	Corrected Nanoamps						
		1	2	3	4	5	6	7
22	F-13	224	300	286	221	227	191	97
28	C-10	229	309	287	226	225	192	97
32	C-6	228	300	283	223	221	188	92
35	F-3	230	310	287	226	229	191	95
39	L-3	226	305	282	225	220	194	96
43	O-6	228	311	289	224	218	191	93
47	O-10	229	308	283	222	217	179	95
50	L-13	225	304	277	228	226	193	98
	Average	227	306	284	224	223	190	95
	Highest	230	310	289	228	229	194	98
	Lowest	224	300	277	221	217	179	92
	(1) Difference	3	4	5	4	6	4	3
		3	6	7	4	6	11	3

(1) Difference is taken between the average value and the highest and the lowest readings.

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TABLE 5.3-2

F̄ Calculation Sheet 1

SPND DATA and CORRECTION FACTORS NECESSARY for a HAND CALCULATION of
CORE POWER DISTRIBUTION from INCORE DETECTOR READINGS

REACTOR : TMI-#1 Initials of Analyst DAL

Control Rod Group Positions

Gps 1-4 100 % wd Gp 6 73 % wd
Gp 5 98 % wd Gp 7 0 % wd
Gp 8 30 % wd

Core Power Level 76 % FP
Boron Concentration 1215 ppm
Core Burnup 9.02 EFPD
Axial Imbalance -13 % FP

Xenon Conditions
Equilibrium: 2-D / 3-D / neither
Reactivity Worth NA % Δk/k
Max Quadrant Tilt 0 %

Computer Printout
Date 7/17/74
Time 0400
Format 48

1/8 core FA** location	Incore det.* number	Corrected Incore Det. Read. (na)--Corr. SPND, Group 48							Enrichment		Fuel Assembly Type		Power/Signal Correction	
		Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7	%	factor	Rods/LBP	Correction factor	Factor	Comments
H-8	1	334	432	462	433	394	330	174	2.72	1.000	Gp-7	1.202	1.202	Same all levels
G-8	2	444	631	643	598	564	445	230	2.7[illegible]	1.000	LBP_1	1.024	1.024	same all levels
F-8	4	582	876	900	821	783	583	308	2.0[illegible]	0.805	Gp-2	0.97*	0.7809*	same all levels*
E-6	10	491	715	690	659	599	458	208	2.7[illegible]	1.000	LBP_2	1.024	1.024	same all levels
D-8	14	562	868	812	760	753	339	155	2.0[illegible]	0.805	Gp-6	.97/1.202	.78/.968	1-5/6 & 7
C-8	21	454	672	644	597	559	400	203	2.7[illegible]	1.000	LBP_2	1.024	1.024	same all levels
B-6	30	445	674	681	633	594	448	196	3.[illegible]	1.077	Gp-4	0.97*	1.0447*	same all levels*
A-8	37	296	442	449	421	410	309	141	3.0[illegible]	1.077	no rods	0.97	1.0447	same all levels
G-9	3	557	793	807	796	715	589	308	2.0[illegible]	0.805	Gp-5	.97	.7809	Same all levels
F-10	12	593	858	826	743	732	569	270	2.0[illegible]	0.805	Gp-2	0.97*	0.7809*	same all levels*
E-11	26	531	807	691	583	687	297	128	2.0[illegible]	0.805	Gp-6	.97/1.202	.781/.968	1-5/6 & 7
D-12	41	432	642	578	549	538	449	203	2.0[illegible]	0.805	Gp-4	0.97*	0.7809*	same all levels*
C-13	52	185	274	261	238	232	193	92	3.0[illegible]	1.077	no rods	0.97	1.0447	same all levels
F-9	6	488	[illegible]	697	637	640	485	260	2.7[illegible]	1.000	LBP_1	1.024	1.024	same all levels
E-9	5	575	859	800	738	708	558	289	2.[illegible]	0.805	Gp-1	0.97*	0.7809*	same all levels*
D-9	20	457	672	581	528	586	420	176	2.72	1.000	LBP,	1.024	1.024	same all levels
C-9	29	505	735	665	607	646	488	235	2.06	0.805	Gp-3	0.97*	0.7809*	same all levels*
B-9	31	348	487	474	438	440	330	166	2.72	1.000	LBP,	1.024	1.024	same all levels
A-9	45	238	369	372	336	325	265	133	3.05	1.077	no rods	0.97	1.0447	same all levels
E-10	17	469	719	630	569	600	449	210	2.72	1.000	LBP_3	1.024	1.024	same all levels
D-10	27	516	788	425	350	617	500	227	2.[illegible]6	0.805	Gp-8	.97/1.202	.781/.968	1,2,5 & 7/3&4
C-10	28	377	526	443	396	444	353	174	2.72	1.000	LBP_2	1.024	1.024	same all levels
B-10	44	207	274	270	250	248	198	99	3.05	1.077	Gp-7	1.202	1.295	Same all levels
A-10	46	150	215	221	209	196	153	77	3.05	1.077	no rods	0.97	1.0447	same all levels
D-11	33	417	597	506	453	586	369	190	2.72	1.000	LBP_3	1.024	1.024	same all levels
C-11	42	419	610	571	508	558	433	226	2.[illegible]6	0.805	Gp-5	.97	.781	Same all levels
B-11	49	230	353	337	302	296	260	129	3.05	1.077	no rods	0.97	1.0447	same all levels
C-12	48	277	407	389	347	344	287	147	3.05	1.077	LBP_2	1.013	1.091	same all levels
B-12	51	158	234	233	220	215	164	74	3.05	1.077	no rods	0.97	1.0447	same all levels

** 1/8 core FA location is location within standard eighth core which is symmetric to actual detector location.

* Group 1-4 rods are 100% withdrawn whenever core power distribution is taken.

TABLE 5.3-3

$\overline{F}$ Calculation Sheet 2

CORE POWER DISTRIBUTION from INCORE DETECTOR READINGS

REACTOR : TMI-I

Initials of Analyst DAL

Control Rod Group Positions		Core Power Level 76 % FP	Xenon Conditions	Computer Printout
Gps 1-4 100 % wd	Gp 6 73 % wd	Boron Concentration 1215 ppm	Equilibrium: 2-D / 3-D / neither	Date 7/17/74
Gp 5 98 % wd	Gp 7 0 % wd	Core Burnup 9.02 EFPD	Reactivity Worth NA % Δk/k	Time 0400
	Gp 8 30 % wd	Axial Imbalance -13 % FP	Max Quadrant Tilt 0 %	Format 48

1/8 core FA** location	Incore detector number**	Adjusted Incore Detector Readings (na)—Relative Power—F level							Adjusted Sum 1-7	Weighting factor	Relative total
		Level 1	Level 2	Level 3	Level 4	Level 5	Level 6	Level 7			
H-8	1	401.5	519.[illegible]	555.3	520.5	473.6	396.7	209.1	3075.9	1	3075.9
G-8	2	454.7	646.1	658.4	612.4	577.5	455.7	235.5	3640.3	4	14561.3
F-8	4	454.5	684.1	702.8	641.1	611.4	455.3	240.5	3789.7	4	15158.8
E-8	10	502.8	732.2	706.6	674.8	613.4	469.0	213.0	3811.7	4	15646.7
D-8	14	438.9	677.8	634.1	593.5	588.0	328.2	150.1	3407.1	4	13628.4
C-8	21	464.9	688.1	659.5	611.3	572.4	409.6	207.9	3613.7	4	14454.8
B-8	30	464.9	704.1	711.4	661.3	620.6	468.0	204.8	3835.1	4	15340.4
A-8	37	309.2	461.8	469.1	439.8	428.3	322.8	147.3	2578.3	4	10313.2
G-9	3	435.0	619.3	630.2	621.6	558.3	460.0	240.5	3565.3	4	14261.1
F-10	12	463.1	670.0	645.0	580.2	571.6	444.3	210.8	3585.1	4	14340.4
E-11	26	515.1	782.8	670.3	565.5	666.4	357.0	153.9	2995.7	4	11982.9
D-12	41	337.3	501.3	451.4	428.7	420.1	350.6	158.5	2648.0	4	10592.1
C-13	52	193.3	286.2	272.7	248.6	242.4	201.6	96.1	1540.9	4	6163.7
F-9	6	499.7	718.8	713.7	652.2	655.4	496.6	266.2	4002.8	8	32022.5
E-9	5	449.0	670.8	624.7	576.3	552.9	435.7	225.7	3535.1	8	28281.1
D-9	20	468.0	688.1	594.9	540.7	600.1	430.1	180.2	3522.6	8	28180.5
C-9	29	394.4	574.0	519.3	474.0	504.5	381.1	183.5	3030.6	8	24244.8
B-9	31	356.4	498.7	485.4	448.5	450.6	337.9	170.0	2747.4	8	27979.1
A-9	45	248.6	385.5	388.6	351.0	339.5	276.8	138.9	2129.1	8	17032.8
E-10	17	480.3	736.3	645.1	582.7	614.4	459.8	215.0	3733.5	8	29868.0
D-10	27	402.9	615.3	411.4	338.8	481.8	390.5	177.3	2818.3	8	22246.3
C-10	28	386.0	538.6	453.6	405.5	454.7	361.5	178.2	2368.5	8	18948.1
B-10	44	268.1	354.8	349.7	323.8	321.2	256.4	128.2	2002.1	8	16016.6
A-10	46	156.7	224.6	230.9	218.3	204.8	159.8	80.4	1275.6	8	10204.6
D-11	33	427.0	611.3	518.1	463.9	600.1	377.9	194.6	3126.3	8	25010.2
C-11	42	327.2	476.3	445.9	396.7	435.7	338.1	176.5	2576.8	8	20774.6
B-11	49	240.3	368.8	352.1	315.5	309.2	271.6	134.8	1992.2	8	15937.9
C-12	48	216.3	317.8	303.8	271.0	268.6	224.1	114.8	2398.0	8	19184.1
B-12	51	165.1	244.5	243.4	229.8	224.6	171.3	77.3	1356.0	8	10848.2

Sum = 506598.5

$$\overline{F}_{core} = \frac{\text{Sum of Relative Totals}}{177 \times 7}$$

$\overline{F}_{core}$ = 408.877

** 1/8 core FA location is location within standard eighth core which is symmetric to actual detector location.

TABLE 5.3-4

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TABLE 5.3-5

P/$\overline{P}$ CALIBRATION SHEET for BACKUP RECORDER A

Computer Printout

Date 7/17/74 Time 0400

Format 48

Reactor : TMI-I

Core Power Level 76 % FP

Initials of Analyst DAL

Recorder pin number	Incore Detector location	Incore Detector number	Corrected Incore Det. Reading**	Symmetric FA location in 1/8 core	Power/Signal correction factor	Adjusted Det. Read. P_{level}	P_{level} / $\overline{P}_{core}$	Calculated Recorder Reading MR_c	Initial Recorder Reading MR_i	Correction factor (MR_c/MR_i)	Present Recorder Read. MR_p	Correct Reading $MR_p (MR_c/MR_i)$
1	G-11-6	25	573	E- 9-6	0.7809*	447.	1.093	83.1	70	1.187	.234MV	.274MV
2	H- 8-6	1	330	H- 8-6	1.024	338.	.826	62.8	52	1.208	.162MV	.202MV
3	F- 3-6	35	352	C-10-6	1.024	360.	.880	66.9	65	1.062	.215MV	.228MV
4	G-11-4	25	753	E- 9-4	0.7809*	558.	1.438	109.3	102	1.072	.348MV	.373MV
5	H- 8-4	1	433	H- 8-4	1.202	520.	1.272	96.6	72	1.342	.238MV	.321MV
6	F- 3-4	35	398	C-10-4	1.024	408.	.998	75.8	68	1.115	.239MV	.255MV
7	G-11-2	25	864	E- 9-6	o.7809*	675.	1.651	125.5	125	1.004	.417MV	.419MV
8	H- 8-2	1	432	H- 8-2	1.202	519.	1.269	96.5	71	1.359	.237MV	.322MV
9	F- 3-2	35	525	C-10-2	1.024	538.	1.315	100.0	82	1.219	.267MV	.325MV
10	C- 6-2	32	511	C-10-2	1.024	523.	1.279	97.2	86	1.130	.287MV	.324MV
11	C-10-2	28	526	C-10-2	1.024	539.	1.318	100.2	90	1.113	.299MV	.333MV
12	F-13-2	23	515	C-10-2	1.024	527.	1.289	97.9	98	.999	.324MV	.324MV
13	L-13-2	50	525	C-10-2	1.024	538.	1.316	100.0	94	1.064	.316MV	.336MV
14	O-10-2	47	534	C-10-2	1.024	547.	1.338	101.6	98	1.037	.331MV	.343MV
15	O- 6-2	42	530	C-10-2	1.024	543.	1.328	100.9	91	1.109	.307MV	.340MV
16	L- 3-2	39	496	C-10-2	1.024	508.	1.242	94.4	87	1.085	.267MV	.290MV
17	K-11-6	19	581	E- 9-6	0.7809*	454.	1.110	84.4	70	1.205	.231MV	.278MV
18	M- 9-6	16	556	E- 9-6	0.7809*	434.	1.061	80.7	72	1.120	.232MV	.260MV

** Corrected Incore Detector Reading (nanoamps)--
Corrected SPND signals, Group 48.

* Group 1 rods are 100% withdrawn whenever core power distribution is taken.

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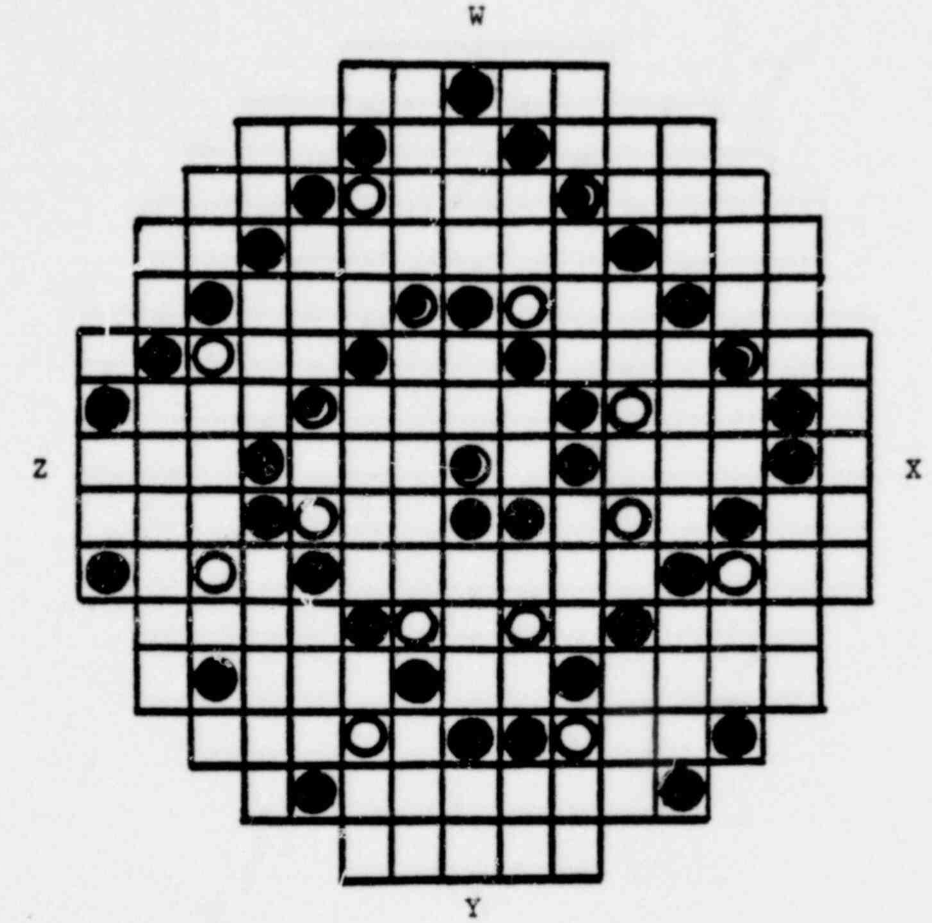
INCORE DETECTOR LOCATIONS



Symmetry monitors

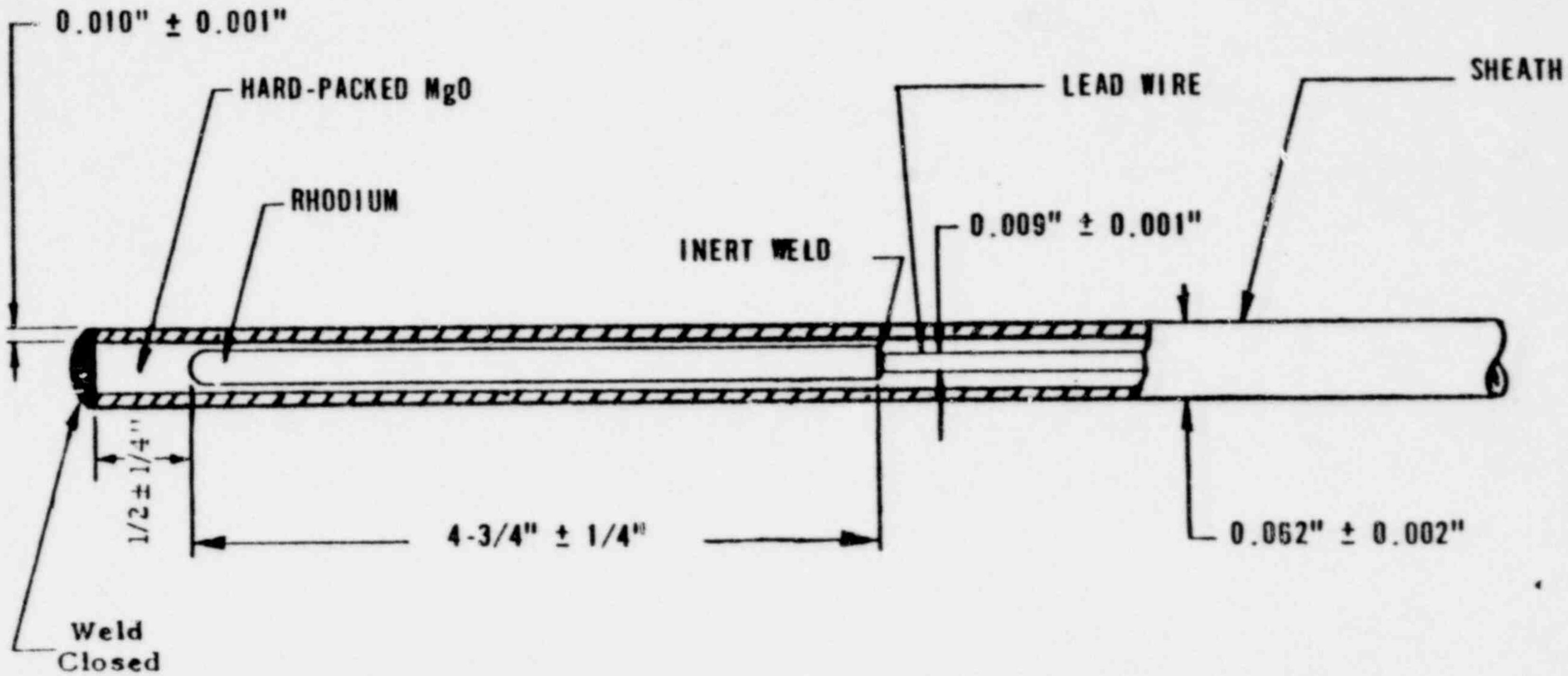
Total core monitors based on ¼ core symmetry

Combination total core and symmetry monitors

FIGURE 5.3-1

FIGURE 5.3 - 2

Incore Self-Powered Neutron Detector



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UNCORRECTED SPND MAP

UNCORRECTED SPND MAP LEVEL 6 ALL LOCATIONS IN NANOAMPS 06:01:01 06/20/74

```
                          31   30
                           99  130
                       32         29    28          52
                     *  99*       139 * 101*         54
                   33                  27                51
                   107                 145                41
              34          7        5      26
              108       * 160*  * 165*    124
          35              6    4             24    23
        * 101*            136  164           148 * 102*
      36           9    8          3      25      22
      106       * 172*  136       165   * 165*    131
  37               10           1    2            21
   83              137          86  124           122
                   11                     19    20
                 * 171*                 * 165*  123
     38    39         12                  18      50
      49 * 104*       166                 127   * 103*
          40              13      16   17               49
          121           * 168*  * 163* 127              66
               41              14   15
               129             135  117
                  42    43               47      48
                  114 * 102*           *  95*    79
                        44
                        54
                           45          46
                            78         41
```

FIGURE 5.3 - 3

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CORRECTED SPND MAP

CORRECTED SPND MAP LEVEL 5 ALL LOCATIONS IN NANOAMPS 06:01:01 06/29/74

```
                          31  30
                          139 182
                     32         29     28          52
                   * 127*       191 * 127*         71
                 33                  27                  51
                 143                 140                  62
          34             7      5        26
          149        * 228*  * 216*     196
     35                  6   4                24    23
   * 131*                187 234              168 * 130*
  36           9      8        3        25        22
  147       * 231*   189      222    * 222*       187
37             10          1    2                 21
124            186        119  173                177
               11                       19    20
             * 229*                   * 228*  171
 38    39           12                  18        50
 65  * 127*         226                 170     * 130*
      40              13      16      17               49
      158          * 216*  * 226*     168              83
           41              14   15
           163             227  159
             42   43                   47         48
             151 * 124*              * 124*       103
                  44
                  72
                     45                46
                     108               59
```

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INCORE DETECTOR
RESPONSE VS
REACTOR POWER

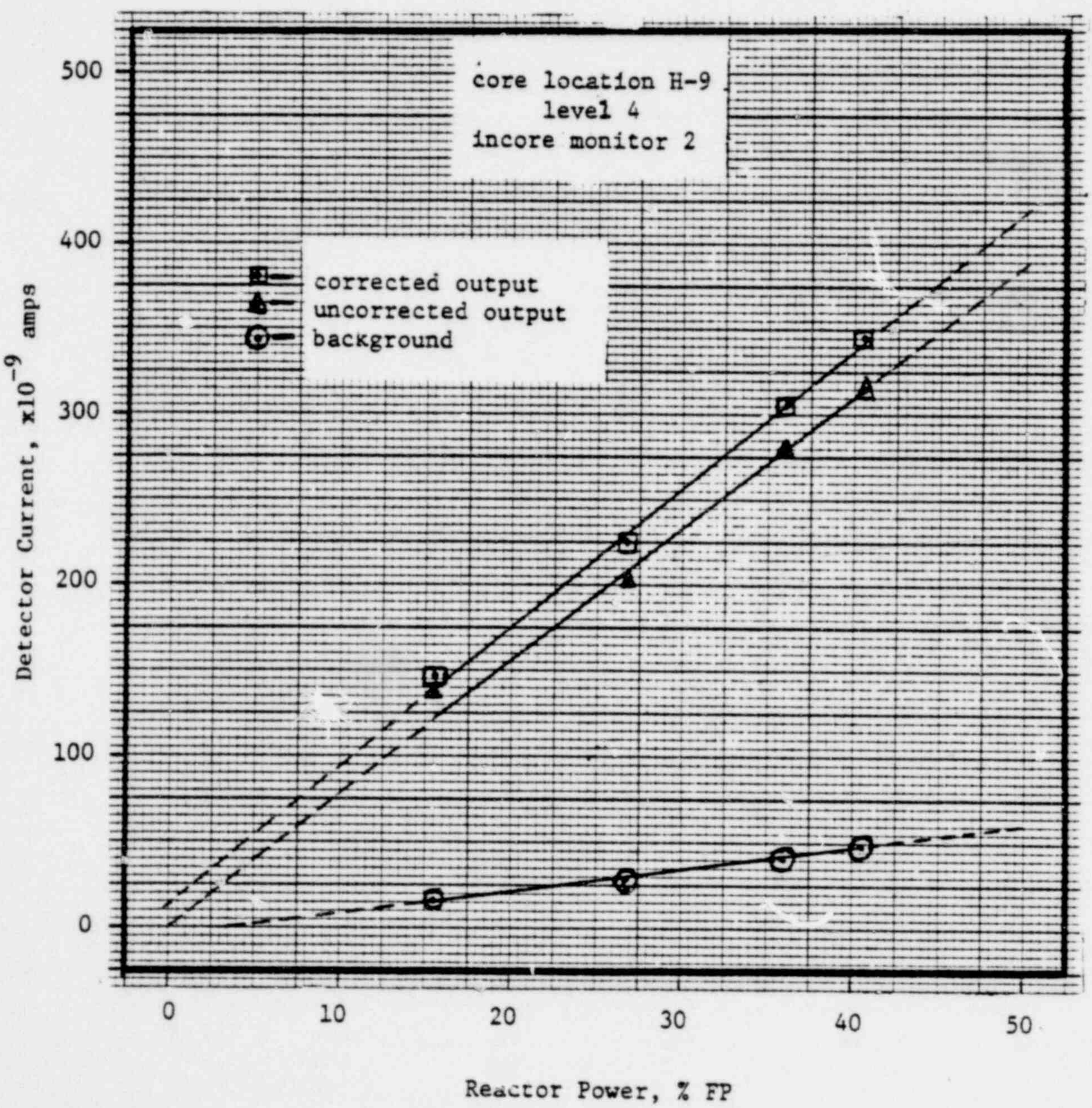

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FIGURE 5.3-5

5.4 POWER IMBALANCE DETECTOR CORRELATION TEST

5.4.1 PURPOSE

The Power Imbalance Detector Correlation Test has three objectives:

(a) To determine the relationship between the induced power distribution as indicated by out-of-core instrumentation and the actual incore power distribution.

(b) To verify the adequacy of the imbalance system trip setpoints.

(c) Verify the adequacy and accuracy of backup imbalance calculations using OP 1203/7, "Power Imbalance and Quadrant Power Tilt Calculations Using the Backup Incore Detector System".

5.4.2 TEST METHOD

This test was conducted at 40% and 76% FP to determine the relationship between the core axial imbalance as indicated by the incore detectors and the out-of-core detectors. Based upon this correlation, it could be verified that the minimum DNBR and maximum linear heat rate limits would not be exceeded by operating within the flux/delta flux/flow envelope set in the Reactor Protection System.

The method employed to conduct the test at both power levels was the same. The sequence is outlined below:

(a) Steady State conditions were established at the desired power level with core xenon concentrations at equilibrium.

(b) The Incore Monitoring System was verified as operational and the backup recorders were checked as having been calibrated in accordance with TP 800/24, "Incore Detector Testing".

(c) The unit computer was verified as operational with applicable Nuclear Steam System (NSS) programs functioning properly.

(d) Prior to conducting this procedure, Nuclear Instrumentation Calibration at Power was used to calibrate the out-of-core detector imbalance to read within 0%, ±1% of the incore imbalance.

(e) Baseline data consisting of the following was collected:

(1) Computer Group 33, Nuclear Instrumentation
(2) Computer Group 55, Imbalance/Tilt/Insertion
(3) Computer Group 34, 3-D Power Map
(4) Computer Group 45, Uncorrected SPND Map for levels 1 thru 7
(5) Computer Group 32, Heat Balance
(6) Core burnup and RCS boron concentration

(f) Once the base line data was acquired, an imbalance was established using the group 8 control rods (APSRs). During group 8 movement, the integrated control system automatically compensated for the reactivity changes by repositioning group 6 to maintain constant power level.

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(g) The imbalance previously established was observed for a minimum period of twenty minutes prior to obtaining the following data:

(1) Computer group 31, Fluid Conditions
(2) Specified Operator Trend Group
(3) Backup Incore Detector Recorder data
(4) Computer group 20, Worst Case Thermal Conditions
(5) Computer group 51, Normalized SPNP Map
(6) Specified Operator Trend Group (second printout)
(7) Computer group 55, Imbalance/Tilt/Insertion

(h) A new imbalance was then established and the same data was recorded once again; this procedure was repeated until the maximum positive and negative imbalances had been established and the required data recorded.

The differences between the measurements at 40% and 76% FP were:

(a) The power imbalance limits observed were +20%/-25% at 40% FP and +14%/-23% at 76% FP.

(b) The measurement at 40% FP used a gain factor of 3.20 set into the scaled difference amplifiers of the power range detector channels. The measurements at 76% FP used the gain factor determined during the 40% FP measurement to verify that the measured gain factor met the acceptance criteria.

As each imbalance condition was established, core power distribution and worst case thermal information was obtained from the plant computer to ensure safe conduct of the test. A plot was maintained of incore offset versus out-of-core offset.

Based upon previous startup experience, it was determined that the relationship between the incore detector (ICD) and out-of-core (OCD) offsets was linear and of the form given in Equation 5.4-1.

$$OCO = M \times K \times ICO + B \quad \text{(Equation 5.4-1)}$$

Where: OCO = Out-of-Core Offset, % FP
ICO = Incore Offset, % FP
M = Slope of line
B = Intercept at Zero ICO
K = Gain Factor (difference amplifier)

The experimental slope could be determined from the plot of ICD versus OCD offset. Once the experimental slope was known, the difference amplifier gain (K factor) required to meet the acceptance criteria was determined from Equation 5.4-2.

$$K = M_2/M_1 \quad \text{(Equation 5.4-2)}$$

Where: K = Gain Factor (difference amplifier)
M_1 = Experimentally determined slope
M_2 = Desired Slope

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5.4.3 TEST RESULTS

The relationship between the ICD and OCD offsets was determined at 40% and 76% FP by performing an imbalance scan with the APSRs. The measured results at 40% FP yielded an average slope of 0.734 for the ICD versus OCD offset relationship. Initially, the minimum permissible slope was specified as 0.835, but was later increased to 0.920 to provide adequate margins to core imbalance limits. Using the measured slope of 0.734 and an acceptance criteria (desired) slope of 0.920, the minimum K factor was found to be 4.00. This value, however, would only give marginal assurance that the minimum acceptance criteria would be met. With an upper limit placed on the acceptable slope of 1.0 (K = 4.35) to avoid unnecessary restrictions on the OCD imbalance, it was decided to use a K factor of 4.033, which corresponds to a slope of 0.925.

After adjustment of the difference amplifier K factors to 4.033, the imbalance scan was performed at 76% FP to verify the results of the 40% FP measurement. The average slope measured on the four out-of-core detectors was 0.991, corresponding to an "actual" difference amplifier gain of 4.072. This value compared well with the 40% FP results and the values established at 40% were accepted as providing the more conservative results.

A comparison of the incore detector (ICD) offset versus the out-of-core (OCD) detector offset obtained for each NI channel is shown in Table 5.4-1. The data taken at 40% FP was obtained with a difference amplifier gain setting of 3.20 while the 76% FP data reflects the experimentally determined K factors.

Core power distribution measurements were taken in conjunction with each of the imbalance measurements discussed above and the values of minimum DNBR and maximum linear heat rate were determined, extrapolated to the applicable overpower trip setpoint and compared to the acceptance criteria. The measured values of linear heat rate were multiplied by an uncertainty factor of 1.432 (discussed in section 5.9) to provide adequate conservatism in the comparison with the acceptance criteria. The worst case values of minimum DNBR and maximum linear heat rate determined at 40% and 76% FP are listed in Table 5.4-2, along with the extrapolations to the overpower trip setpoints for the next plateau in the power escalation sequence.

TABLE 5.4-2

Nominal Power (%)	Measured DNBR	Measured LHR x1.432(kw/ft)	Extrapolations		
			Power (%)	DNBR	LHR
40	8.48	7.90	95	3.6	18.8
76.4	4.56	13.56	105.5	3.6	18.73

The worst case DNB ratio was greater than the minimum limit of 1.55 and the maximum value of linear heat rate was less than the fuel melt limit of 19.6 kw/ft after extrapolation to 105.5% FP (the overpower trip setpoint). These results show that Technical Specification limits have been met and that adequate protection is provided by the Reactor Protection System trip setpoints for the allowed axial imbalances during power operation.

Backup imbalance calculations using OP 1203/7 agreed with computer calculated imbalances within the limits specified in the test. Table 5.4-3 lists the computer calculated imbalances as well as imbalances obtained using the incore detector backup recorders.

TABLE 5.4-3

Nominal Power (%)	Computer Calculated Imbalance (%)	Backup Recorder Imbalance (%)
40	-17.40	-16.32
40	-13.66	-13.41
40	+0.56	+0.24
40	-13.6	-11.19
40	+3.67	+1.55
76	+11.86	+15.18
76	+8.64	+9.53
76	-19.11	-15.02
76	-0.66	+0.17

5.4.4 CONCLUSIONS

Backup imbalance calculations performed in accordance with OP 1203/7 provide a reliable alternate method to computer calculated values of imbalance.

Utilization of the Difference Amplifier Gain "K" factor as determined at 40% power, resulted in good agreement between Incore and Out-of-Core Detector Offset indications.

The two most important values verified as a result of this test were Minimum DNBR and Maximum Linear Heat Rate. Both of these parameters were well within Technical Specification limitations.

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TABLE 5.4-1

MEASURED ICD AND OCD OFFSETS AT 40% AND 76% FP

Nominal Power (%)	ICD Offset (%)	OCD Offset (%) NI-5	NI-6	NI-7	NI-8
40	-42.04	-33.01	-36.62	-33.92	-35.28
40	-32.63	-30.07	-33.59	-31.33	-32.08
40	-26.11	-28.77	-31.52	-28.37	-30.50
40	-15.32	-17.28	-21.34	-19.58	-21.05
40	-12.89	-18.20	-22.01	-19.48	-21.48
40	-0.67	-3.34	-7.13	-7.06	-7.25
40	1.70	-1.92	-5.47	-5.41	-5.46
40	8.85	6.70	3.60	2.60	2.63
76	-24.60	-26.57	-28.53	-24.40	-27.07
76	-17.42	-20.69	-21.34	-18.17	-20.29
76	-8.15	-10.00	-10.58	-8.4	-9.71
76	-0.85	-2.26	-1.84	-1.06	-1.72
76	+0.72	+0.53	+0.79	+1.45	+0.53
76	+7.17	+5.55	+5.92	+6.10	+5.42
76	+11.29	+6.87	+6.84	+7.26	+6.46
76	+15.20	+18.75	+17.62	+16.28	+18.04

$$\text{where ICD OFFSET} = \frac{\text{POWUP-POWLW}}{\text{POWUP+POWLW}} \times 100\%$$

$$\text{OCD OFFSET} = \frac{\text{CHANNEL IMBALANCE}}{\text{CHANNEL POWER}} \times 100\%$$

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5.5 ROD WORTH AT POWER

5.5.1 PURPOSE

The purpose of the Rod Worth at Power test was to define a method for measuring differential control rod reactivity worths during equilibrium and transient power operation. This test was conducted in conjunction with Reactivity Coefficient, Pseudo Rod Ejection and Dropped Rod measurements.

5.5.2 TEST METHOD

The fast insert/withdrawal technique was used to measure differential rod worth under steady state and transient conditions. After placing the Diamond Control Station in hand, the selected control rods were withdrawn for six seconds and immediately inserted for six seconds. The Diamond Control Station was placed back in auto after waiting a minimum of 15 seconds following completion of rod motion. The reactimeter was used to calculate the change in core reactivity during the measurement. The differential worth was then obtained by dividing the known change in core reactivity by the change in rod position. The measured results were used to verify previously generated differential rod worth data.

The following conditions were established in the reactor coolant system (RCS) prior to recording test data:

(a) RCS Average Temperature = 579°F ±1°F.

(b) RCS Pressure = 2155 psig ±25 psig.

(c) Reactor Power constant within ±1% for 20 minutes.

(d) The RCS make-up tank was filled to ~90% and approximately two hours elapsed to allow system boron concentrations to equalize. The pressurizer was constantly sprayed for eight hours prior to the test in order to prevent any potential pressurizer out-surge from affecting test results. Pressurizer, make-up tank and RCS boron samples were obtained just prior to recording test data to verify equilibrium and to obtain a baseline boron concentration.

5.5.3 TEST RESULTS

The results of differential rod worth measurements performed during power escalation testing are presented in Table 5.5-1. The core power level, RCS boron concentration and control rod positions that existed for each measurement are shown and the predicted results are included for comparison. Measured differential worths were well within the acceptance criteria limit of ±20% of the predicted values.

5.5.4 CONCLUSIONS

Differential control rod reactivity worth measurements were performed as required during the power test program. Measured differential rod worths agreed with the design values well within the acceptance criteria limit of ±20%. The maximum measured deviation was less than 4% from the design worth.

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TABLE 5.5-1

MEASURED DIFFERENTIAL ROD WORTHS AT POWER

Power Level (%FP)	Boron Concentration (ppmb)	Rod Group Position, %wd				Differential Rod Worth, 10^{-3} %Δk/k/%wd	
		1-5	6	7	8	Measured	Predicted
40	1192	100	72.2	0	33	9.44	9.77
40	1188	100	65.5	0	13	11.39	NA(1)
76	1120	100	85.5	15.0	20	15.03	15.3
100	1092	100	86.0	11.0	0	11.55	11.44

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(1) NA denotes that the predicted differential worth was not available.

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5.6 REACTIVITY COEFFICIENTS AT POWER

5.6.1 PURPOSE

Four coefficients of reactivity normally associated with light water reactors are defined as follows:

(a) The Temperature coefficient of reactivity is defined as the fractional change in core net reactivity per unit change in fuel and moderator temperature.

(b) The Moderator Temperature coefficient of reactivity is defined as the fractional change in core net reactivity per unit change in moderator temperature.

(c) The power doppler coefficient of reactivity is defined as the fractional change in core net reactivity per unit change in core power.

(d) The doppler coefficient of reactivity is defined as the fractional change in core net reactivity per unit change in fuel temperature.

The purpose of this test was to measure the temperature and the Power Doppler coefficients of reactivity at 40%, 76% and 100% of full power. The Moderator Temperature and the Doppler coefficients, which can be derived from the measured coefficients, were then calculated to verify that design limits were not exceeded.

5.6.2 TEST METHOD

Reactivity coefficients were determined at 40%, 76% and 100% of full power by measuring the change in core net reactivity caused by varying core average temperature and power level. The following prerequisite conditions were established at the start of each test to minimize reactivity effects not directly related to the coefficient being measured:

(a) Equilibrium xenon conditions were established.

(b) Stable unit operating conditions were maintained - reactor power was held constant within $\pm 1\%$ full power, reactor coolant temperature and pressure were maintained at 579 $\pm 2^{\circ}$F and 2155 ± 25 psig, respectively.

(c) The soluble boron concentration differences between the Reactor Coolant, the Make-up Tank and the pressurizer were maintained in equilibrium.

After the initial conditions listed above were established, the differential reactivity worth of the controlling rod group(s) were obtained using the Rod Worth at Power test to verify previously generated rod worth data.

The temperature coefficient was measured by adjusting the Bailey Control Station to increase reactor coolant average temperature by 5°F. The actual temperature change, rod movement, reactivity additions and any power changes were recorded on the plant computer, the reactimeter and the brush recorders. The individual reactivity effects were summed and then divided by the change in core average temperature to obtain the temperature coefficient. The measurement was repeated by decreasing core average temperature back to its initial value.

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The power doppler coefficient of reactivity was measured in a similar manner. After the prerequisite conditions were established, reactor power was decreased by 5% while the data was recorded. The reactivity effects of the power change were summed and then divided by the change in power to obtain the power doppler coefficient. The measurement was repeated by increasing core power by 5% to the initial level.

The moderator temperature and doppler coefficients of reactivity were calculated from the measured temperature and power doppler results.

5.6.3 TEST RESULTS

The results of the reactivity coefficient measurements at power are summarized in Tables 5.6-1 and 5.6-2. Table 5.6-1 presents the measured temperature and power doppler coefficient results and provides the calculated results for comparison. The moderator temperature and doppler coefficients were calculated from the measured results and are listed in Table 5.6-2. The measured coefficients are also plotted in Figures 5.6-1 and 5.6-2.

The measured temperature coefficients are all within the acceptance criteria limit of $\pm 0.4 \times 10^{-4} \Delta k/k/^{o}F$ of the calculated values and are more negative. The moderator temperature coefficients calculated from the test results at 40%, 76% and 100% full power are all negative and trend more negative with increasing core power and burnup. Based upon the predicted results, the moderator coefficient will not be positive above 95% FP if the soluble poison concentration is less than 1155 ppm. Measured results from the Reactivity Depletion versus Burnup test (Section 5.11) show that the soluble poison concentration will not exceed 1155 ppm under equilibrium xenon, beginning of life, normal control rod configuration and 100% full power conditions. These results show that the moderator temperature coefficient will not be positive during power operation at or above 95% FP.

Comparison of the measured power doppler coefficient with the calculated values shows that the predicted reactivity deficit versus power is slightly less than predicted. The estimated reactivity deficit from 0% to 100% full power based upon the measured results is $0.90\%\Delta k/k$ as compared to the calculated value of $1.32\%\Delta k/k$. The acceptance criteria limit that the power doppler coefficient be more negative than $-0.55 \times 10^{-4} \Delta k/k/\%$ FP was met.

5.6.4 CONCLUSIONS

The measured results indicate that the moderator temperature coefficient will be negative during power operation above 95% FP. The results of the power doppler coefficient measurements indicate that the least negative value of the coefficient is $-0.00710\%\Delta k/k/\%$ FP at full power conditions.

TABLE 5.6-1

MEASURED COEFFICIENTS OF REACTIVITY AT POWER

A. Temperature Coefficient of Reactivity

Core Power (% FP)	Boron Concentration (ppmb)	Avg. Differential Rod Worth (% Δk/k/% Wd)	Rod Group Position, % Wd				Temperature Coefficient, X10-4 Δk/k/°F	
			1-5	6	7	8	Measured	Calculated
40	1192	0.00940	100	75	0	33	-0.135	-0.020
76	1120	0.01503	100	85	10	20	-0.251	-0.158
100	1090	0.01155	100	90	20	0	-0.329	-0.246

B. Power Doppler Coefficient of Reactivity

Core Power (% FP)	Boron Concentration (ppmb)	Avg. Differential Rod Worth (% Δk/k/% Wd)	Rod Group Position, %Wd				Power Doppler Coefficient, X10-4 Δk/k/% FP	
			1-5	6	7	8	Measured	Calculated
40	1192	0.00940	100	75	0	33	-0.890	-1.37
76	1120	0.01503	100	85	10	20	-0.849	-1.28
100	1090	0.01155	100	90	20	0	-0.734	-1.16

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TABLE 5.6-2

SUMMARY OF REACTIVITY COEFFICIENTS AT POWER

Core Power (% FP)	Boron Concentration (ppmb)	Coefficient of Reactivity, $X10^{-4}$ $\Delta k/k$			
		Temperature	Power Doppler	Moderator	Doppler
40	1192	-0.135	-0.890	-0.008	-0.127
76	1120	-0.251	-0.849	-0.144	-0.107
100	1090	-0.329	-0.734	-0.222	-0.107

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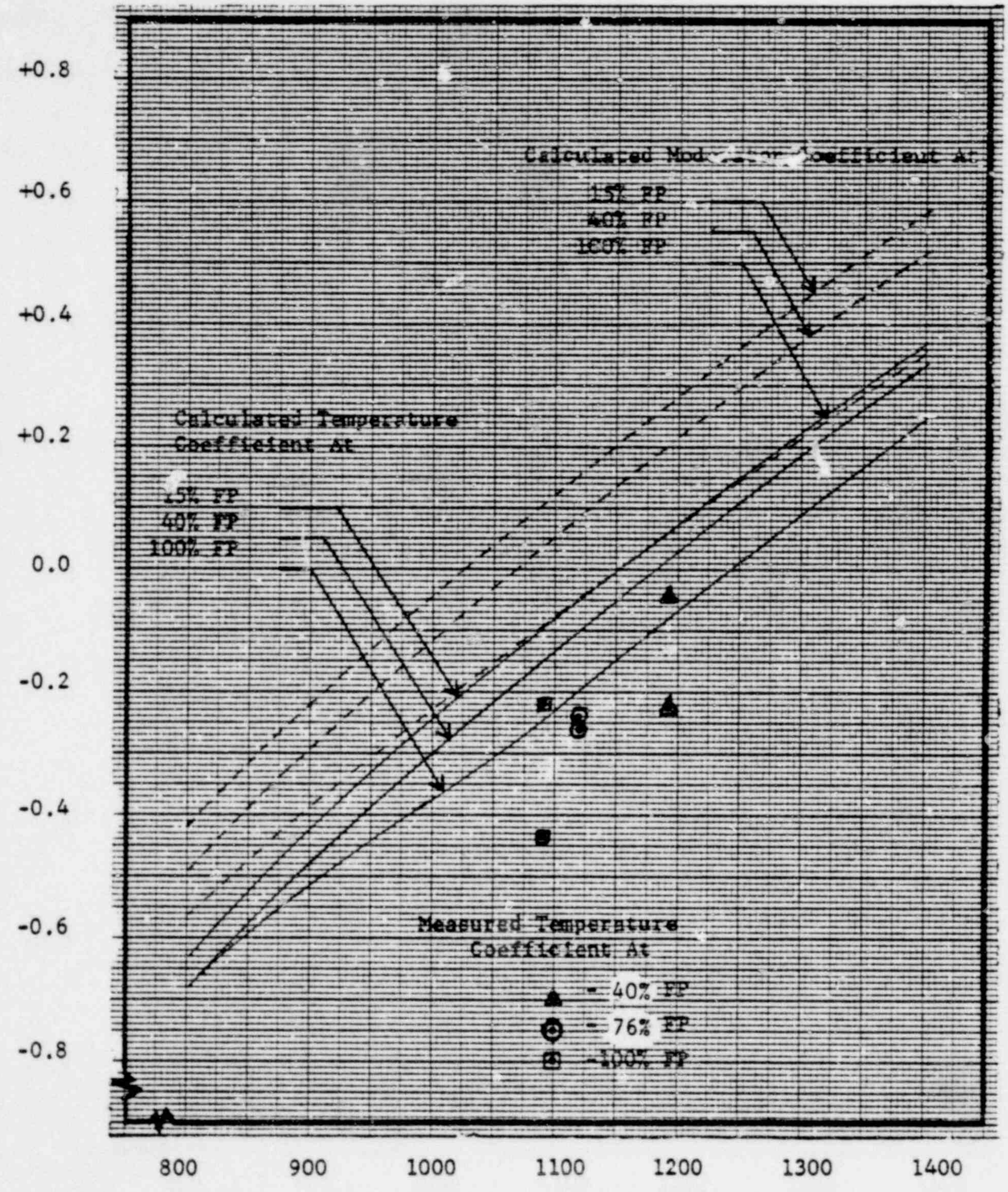
MEASURED AND CALCULATED TEMPERATURE
COEFFICIENTS VERSUS BORON CONCENTRATION
FOR VARIOUS POWER LEVELS



Soluble Poison Concentration, ppmb

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FIGURE 5.6-1

POWER DOPPLER COEFFICIENT OF REACTIVITY
VERSUS POWER LEVEL

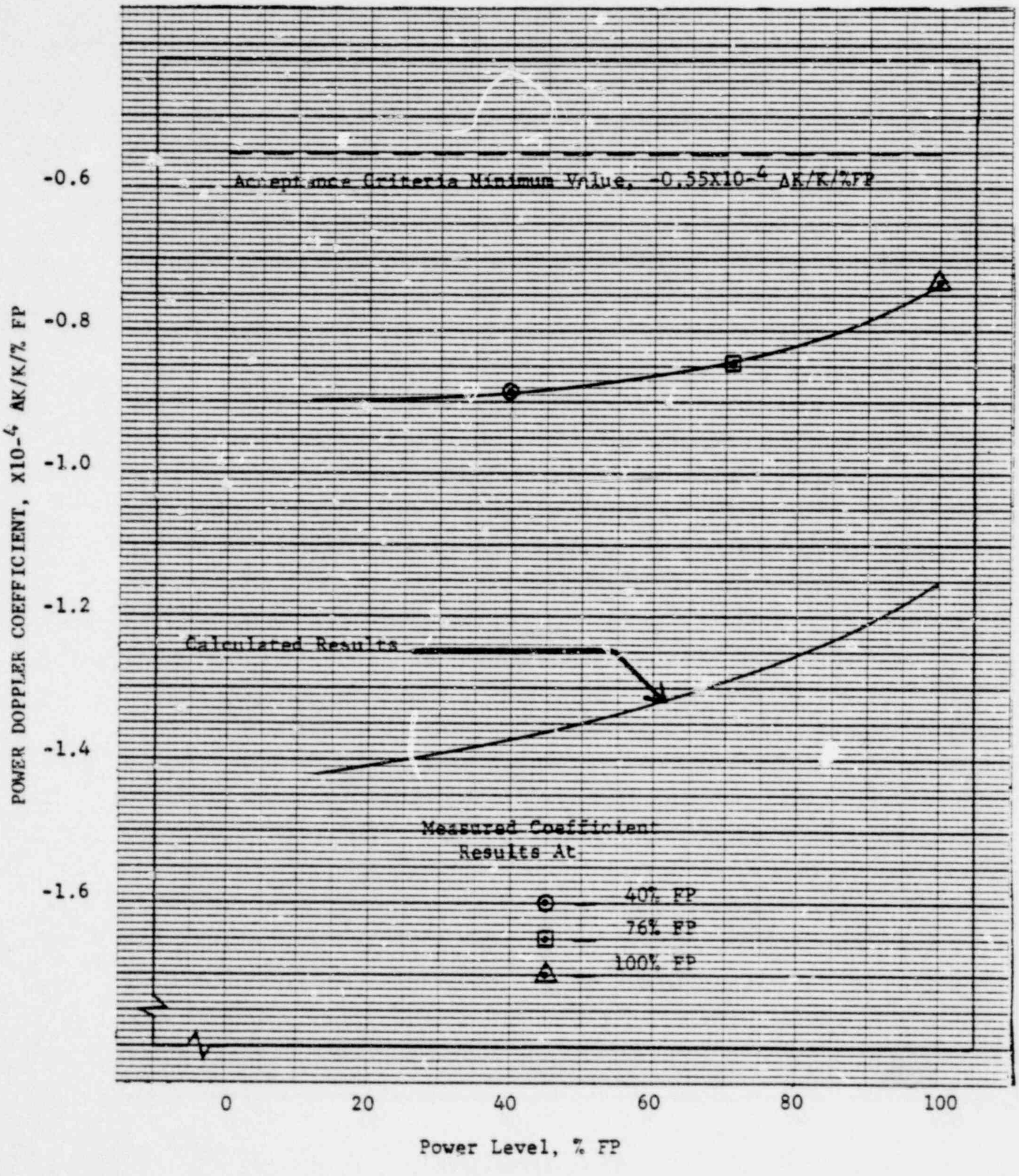

FIGURE 5.6-2

5.7 DROPPED CONTROL ROD TEST

5.7.1 PURPOSE

The purposes of the Dropped Control Rod Test conducted at 40% and 76% of full power were as follows:

(a) To verify the safety analysis relating to the accidental dropping of a control rod which is normally withdrawn from the core at full power. The measurement was performed at 40% FP.

(b) To demonstrate the capability of the rod drive control system to detect a control rod that deviates from its group average position by preset limits, to verify that reactor power is automatically reduced to a specified power level and that automatic control rod withdrawal is inhibited above a specified power level when a control rod asymmetric fault exists; this test was performed at 76% FP.

(c) To verify the adequacy and accuracy of backup quadrant tilt calculations using OP 1203/7, "Power Imbalance and Quadrant Power Tilt Calculations Using the Backup Incore Detector System", at 40% FP.

(d) To insure reactor stability following an induced xenon oscillation at 40% FP.

5.7.2 TEST METHOD

The results of core thermal calculations show that control rod 7 in group 6 (core location H-4), and those rods symmetric to it, would produce the most adverse thermal effects if it were dropped into the core during operation at power. The pseudo dropped control rod test was conducted at 40% FP by measuring the worth of control rod 6-7 as it was inserted in the core and by measuring the resultant core power distributions with the dropped rod at 0% and 50% withdrawn.

The worth of control rod 6-7 was determined by performing a rod swap with control rod group 7. Differential reactivity worths were measured on group 7 using the fast insert/withdraw method at 20% insertion intervals as rod 6-7 was inserted into the core. The measured differential worth data for group 7 was then integrated to obtain the worth of rod 6-7. Core power distribution and worst case thermal conditions data was obtained from the plant computer when rod 6-7 reached 0% and 50% withdrawn. Data from the backup incore recorders was taken concurrent with the computer data and was used to verify the tilt calculations of OP 1203/7.

Control rod 6-7 was realigned with its group after completing the reactivity worth and power distribution measurements. Quadrant power tilt was monitored using the incore detector system for 24 hours following the test to verify that the induced xenon oscillation decayed away.

Measurements at 76% FP were performed to demonstrate the response of the rod drive control system to an asymmetric rod. Control rod 6-7 was selected for individual control and inserted past the 7 in. and 9 in. deviation limits. Control rod maps were obtained from the plant computer when the asymmetric rod alarm and fault conditions occurred. The Diamond Control Station was then put back into automatic and the subsequent power runback and rod withdrawal inhibit were verified.

5.7.3 TEST RESULTS

The results of the pseudo dropped control rod test conducted at 40% and 76% full power are summarized in Figures 5.7-1 through 5.7-4 and Tables 5.7-1 and 5.7-2.

Figure 5.7-1 shows the location of control rod 6-7 and its position relative to each core quadrant. During the insertion of rod 6-7 from its group average position to 0% withdrawn, control rod group 7 was withdrawn a total of 4% to compensate for the reactivity effects of the dropped rod. The initial and final positions of group 7 were 15% and 17% withdrawn, respectively, which was due to a periodic adjustment of the goup 6 and 7 positions to maintain proper overlap between the groups. The results of the differential rod worth measurements on group 7 yield a total dropped control rod worth of 0.094%Δk/k.

Results of the core power distribution measurements with control rod 6-7 inserted in the core are summarized in Figure 5.7-2. These results show the radial core power distribution with rod 6-7 at 85%, 50% and 0% withdrawn. A large depression of the flux in the quadrant containing the dropped rod was observed, as expected. The maximum measured radial peaking factor was 1.708, in the core quadrant opposite to the dropped rod. The maximum quadrant power tilts measured were 14.87% and 14.36% in the XY and WX quadrants, respectively, with rod 6-7 fully inserted. Figure 5.7-3 is a plot of quadrant power tilt versus rod 6-7 position. The quadrant power tilt was subsequently monitored after aligning rod 6-7 with the group 6 average position. The quadrant tilt returned to less than 4% within 24 hours of realignment of the dropped rod which demonstrates the core stability to an induced radial xenon oscillation. Figure 5.7-4 shows the quadrant power tilt versus time after the test.

Worst case thermal conditions were determined during the power distribution measurements taken for the dropped rod test and are summarized in Table 5.7-1. The values of maximum linear heat rate and minimum DNBR were measured with rod 6-7 at 85%, 50% and 0% withdrawn. The minimum value of DNBR was 9.50 and extrapolation of this value to 100% FP resulted in a DNBR of 3.8, which is well above the acceptance criteria limit of 1.55. The measured value of maximum linear heat rate was 5.02 kw/ft which, when multiplied by the 1.432 uncertainty factor, becomes 7.19 kw/ft. This value extrapolated to 100% full power gives a maximum linear heat rate of 18.44 kw/ft which is less than the fuel melt limit of 19.6 kw/ft. The results of this measurement show that even when the worst case uncertainties are assumed, sufficient margin exists to the core thermal limits with a control rod dropped in the core at 100% FP and a resultant quadrant power tilt of 14.87%. Further, this evaluation does not incorporate the reduction in total core power that would occur due to the reactivity inserted by the dropped rod.

The results of the quadrant power tilt calculation performed with data from the backup incore recorders are summarized in Table 5.7-2. The quadrant power tilts determined from the backup recorders were in good agreement with the tilts determined from the incores and within the acceptance criteria limit of ±5% tilt.

The asymmetric rod alarm and fault conditions occurred with rod 6-7 at 4.3% and 5.7% below its group average position. These results compare well with the acceptance criteria limits of 5 ±2% and 6.4 ±2%, respectively. Reactor power was automatically reduced from 76% to 56% FP at a runback rate of 32.3% FP/min when the asymmetric rod was detected by the rod drive system.

Automatic control rod withdrawal was inhibited above 60% FP following the runback, as required.

5.7.4 CONCLUSIONS

The dropped control rod test performed at 40% and 76% FP met all of the required acceptance criteria. The following conclusions were drawn as a result of the measurements.

(a) The core power distributions and thermal conditions that developed from a control rod which was dropped into the core during operation at power showed adequate margins to minimum DNBR and maximum LHR limits. The measured worth of the dropped rod was 0.094%$\Delta k/k$.

(b) Quadrant power tilt calculations performed using backup incore recorder data were accurate in comparison to the computer calculated values.

(c) The rod drive control system responded properly to an asymmetric rod condition.

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TABLE 5.7-1

CORE THERMAL CONDITIONS MEASURED DURING THE
DROPPED CONTROL ROD TEST AT 40% FULL POWER

Core Power (%FP)	CR H-4 Position (% wd)	Min. DNBR	Max.[1] LHR (kw/ft)	Extrap. to 100% FP DNBR	Extrap. to 100% FP LHR (kw/ft)
39	86	11.14	5.64	4.34	14.46
39	49	10.27	6.04	4.01	15.49
39	0	9.50	7.19	3.8	18.44

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(1) Values of maximum LHR include the 1.432 uncertainty factor.

TABLE 5.7-2

COMPARISON OF BACKUP RECORDER AND INCORE DETECTOR TILT CALCULATIONS AT 40% FP

CR 7-4 Position (% wd)	Data Source	Quadrant Power Tilt (%)			
		XY	YZ	ZW	WX
86	Recorder	1.58	-1.58	1.06	-1.06
86	Incores	0.26	0.25	0.71	0.20
Difference (Recorder-Incores)		1.32	-1.83	0.35	-1.26
49	Recorder	5.37	3.76	-1.08	-8.00
49	Incores	5.69	5.65	-6.01	-5.33
Difference (Recorder-Incores)		-0.32	-1.89	4.93	-2.73
0	Recorder	14.87	12.30	-12.30	-14.36
0	Incores	13.85	13.98	-14.07	-13.76
Difference (Recorder-Incores)		1.02	-1.68	1.77	-0.60

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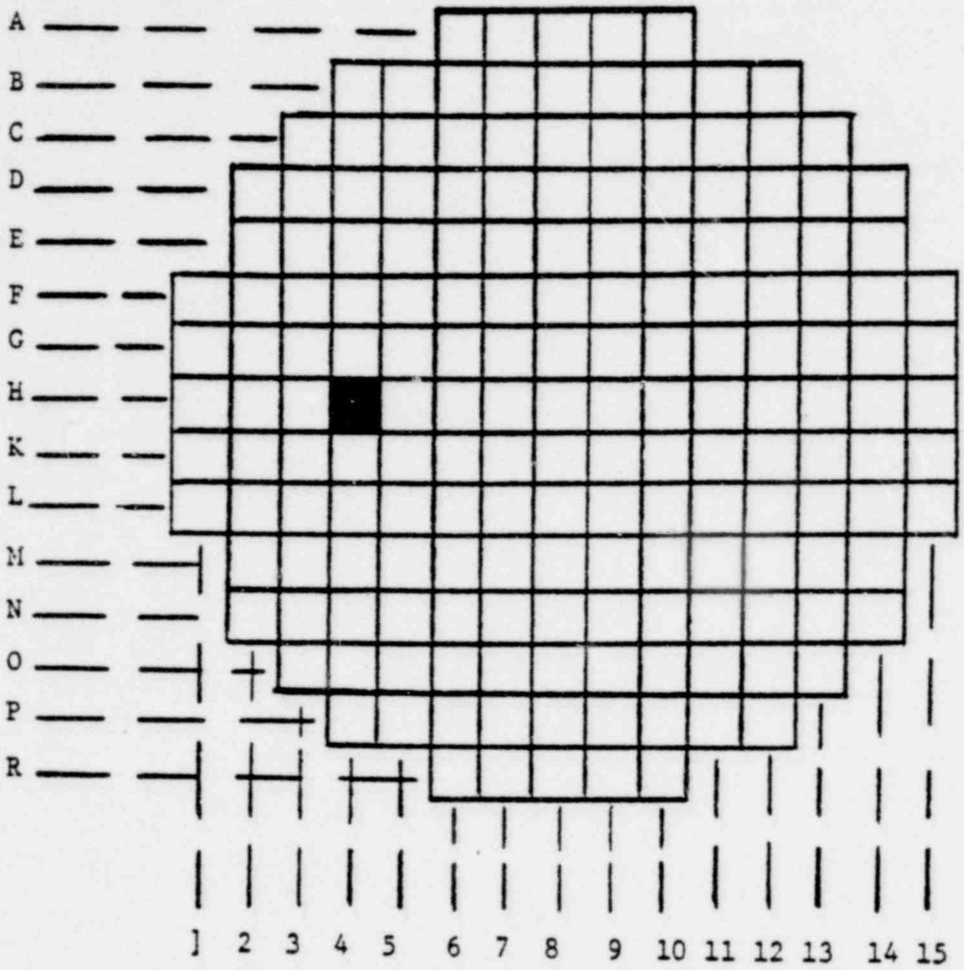
PSUEDO DROPPED CONTROL ROD LOCATION

FOR MEASUREMENT AT 40% FP



Control Rod	Core Location	TMI Unit I Measured Worth (%Δk/k)
6 - 7	H-4	0.094

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FIGURE 5.7-1

MEASURED RADIAL CORE POWER DISTRIBUTIONS
WITH DROPPED CONTROL ROD AT 85%, 50% AND
0% WITHDRAWN FOR EQUILIBRIUM XENON, 40% FP CONDITIONS

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						0.477 0.600 0.717	0.791 0.818 0.850	0.940 0.890 0.771	0.791 0.851 0.845	0.477 0.661 0.692					
B				0.479 0.432 0.527	0.722 0.734 0.768	0.773 0.790 0.826	1.001 1.031 1.053	1.394 1.440 1.479	1.001 1.066 1.089	0.773 0.854 0.882	0.722 0.816 0.812	0.479 0.516 0.512			
C			0.542 0.488 0.568	0.841 0.873 0.903	0.907 0.884 0.855	0.982 0.978 0.973	1.069 1.068 1.078	1.286 1.256 1.214	1.069 1.109 1.151	1.007 1.057 1.111	0.907 0.982 1.010	0.841 0.976 1.034	0.542 0.572 0.612		
D		0.479 0.438 0.484	0.341 0.866 0.865	0.919 0.876 0.807	1.097 1.046 0.993	1.002 0.948 0.893	1.239 1.219 1.209	1.198 1.218 1.255	1.239 1.268 1.316	1.002 1.039 1.091	1.097 1.167 1.249	0.919 1.008 1.071	0.841 0.984 1.072	0.479 0.510 0.551	
E		0.722 0.717 0.673	0.857 0.807 0.697	1.106 1.008 0.902	1.064 0.970 0.857	1.281 1.201 1.123	1.203 1.183 1.172	1.336 1.230 1.120	1.198 1.225 1.271	1.281 1.311 1.380	1.064 1.118 1.200	1.106 1.174 1.280	0.857 0.964 1.043	0.722 0.833 0.902	
F	0.477 0.584 0.165	0.781 0.753 0.612	1.005 0.879 0.712	1.034 0.894 0.707	1.269 1.139 0.990	1.209 1.138 1.064	1.338 1.303 1.274	1.261 1.264 1.280	1.338 1.371 1.434	1.209 1.266 1.367	1.269 1.314 1.401	1.034 1.094 1.175	1.001 1.075 1.165	0.781 0.912 0.969	0.477 0.662 0.769
G	0.791 0.765 0.707	1.044 0.892 0.676	1.056 0.903 0.698	1.230 1.067 0.867	1.235 1.071 0.889	1.325 1.219 1.108	1.189 1.140 1.087	1.225 1.219 1.219	1.189 1.216 1.264	1.325 1.363 1.444	1.214 1.273 1.360	1.230 1.296 1.397	1.056 1.133 1.227	1.044 1.087 1.117	0.791 0.878 0.953
H	0.940 0.803 0.606	1.394 1.324 1.232	1.286 1.131 0.933	1.198 1.099 0.983	1.336 1.130 0.887	1.261 1.192 1.111	1.225 1.182 1.130	1.083 1.110 1.141	1.225 1.263 1.315	1.261 1.347 1.462	1.336 1.343 1.368	1.198 1.347 1.541	1.286 1.382 1.495	1.394 1.538 1.708	0.940 0.934 0.887
K	0.791 0.740 0.686	1.044 0.870 0.658	1.056 0.885 0.683	1.230 1.054 0.857	1.224 1.063 0.884	1.325 1.216 1.107	1.189 1.142 1.091	1.225 1.226 1.227	1.189 1.225 1.274	1.325 1.374 1.455	1.227 1.286 1.374	1.230 1.304 1.405	1.056 1.136 1.230	1.044 1.082 1.111	0.791 0.862 0.935
L	0.477 0.541 0.581	0.781 0.715 0.581	0.970 0.847 0.686	1.034 0.871 0.689	1.269 1.125 0.980	1.209 1.133 1.062	1.338 1.307 1.281	1.261 1.275 1.293	1.338 1.387 1.452	1.209 1.284 1.386	1.269 1.330 1.418	1.034 1.104 1.185	1.000 1.073 1.162	0.781 0.892 0.948	0.477 0.618 0.723
M		0.722 0.675 0.639	0.857 0.771 0.670	1.106 0.982 0.882	1.064 0.955 0.847	1.281 1.196 1.123	1.207 1.307 1.181	1.336 1.243 1.135	1.214 1.241 1.289	1.281 1.327 1.398	1.064 1.129 1.211	1.106 1.172 1.278	0.857 0.942 1.020	0.722 0.781 0.848	
N		0.479 0.410 0.464	0.841 0.841 0.847	0.919 0.858 0.794	1.097 1.036 0.988	1.002 0.947 0.895	1.239 1.188 1.217	1.198 1.228 1.268	1.239 1.277 1.327	1.002 1.043 1.096	1.097 1.156 1.239	0.919 0.975 1.038	0.841 0.917 1.005	0.479 0.398 0.437	
O			0.542 0.493 0.576	0.841 0.876 0.909	0.907 0.889 0.862	0.983 0.984 0.980	1.069 1.225 1.084	1.286 1.257 1.215	1.069 1.099 1.143	0.980 1.030 1.085	0.907 1.156 0.958	0.841 0.884 0.043	0.542 0.429 0.469		
P				0.479 0.475 0.566	0.722 0.764 0.794	0.773 0.807 0.840	1.001 1.033 1.053	1.394 1.423 1.460	1.011 1.023 1.045	0.773 0.777 0.804	0.722 1.156 0.689	0.479 0.333 0.330			
R						0.477 0.807 0.741	0.791 0.817 0.839	0.940 0.847 0.721	0.791 1.099 0.748	0.477 0.590 0.537					

W — Y

Z

x.xx	Rod 6-7 at 85% wd
x.xx	Rod 6-7 at 50% wd
x.xx	Rod 6-7 at 0% wd

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FIGURE 5.7-2

QUADRANT POWER TILT DURING
DROPPED CONTROL ROD TEST AT 40% FP

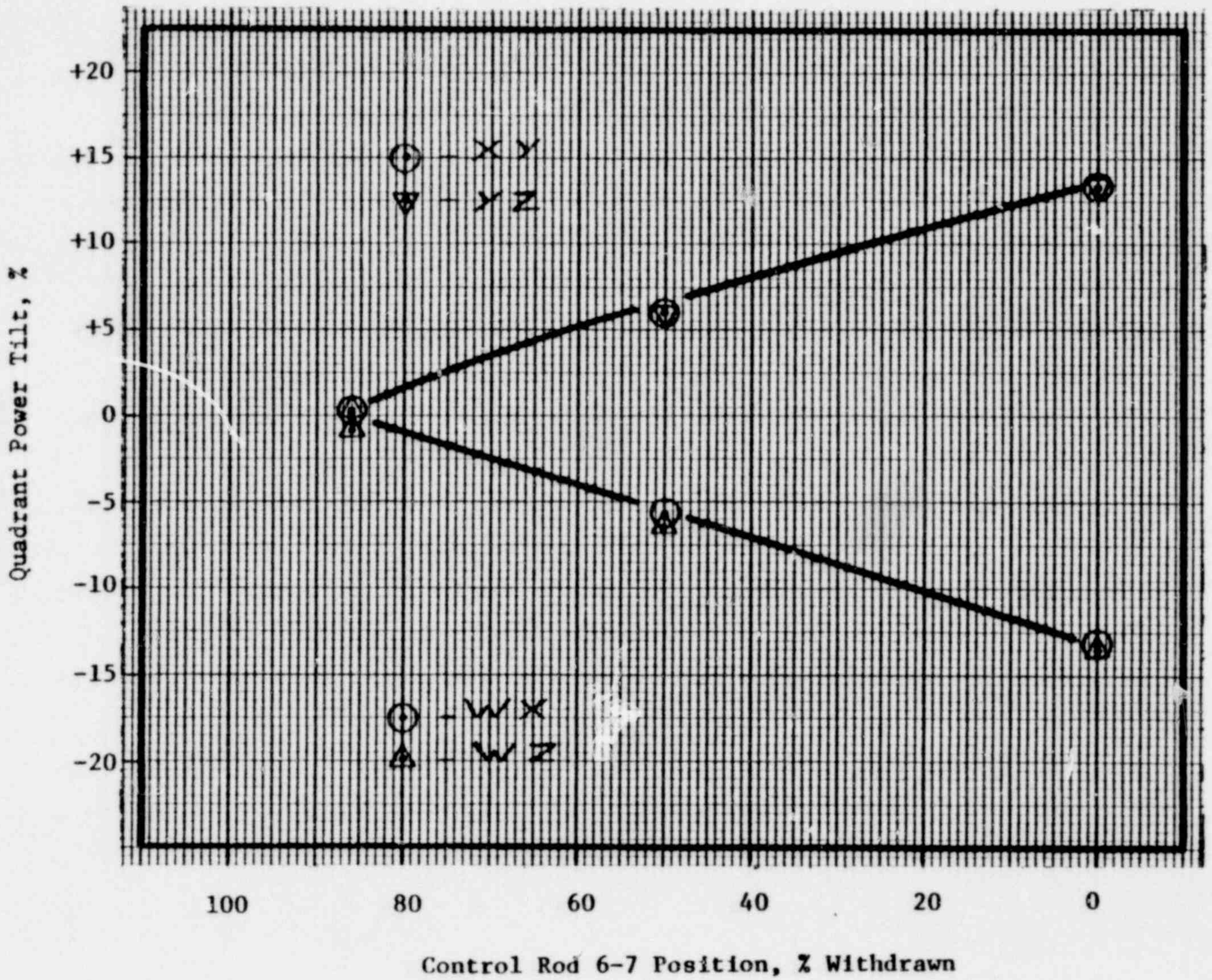

FIGURE 5.7-3

1414 135

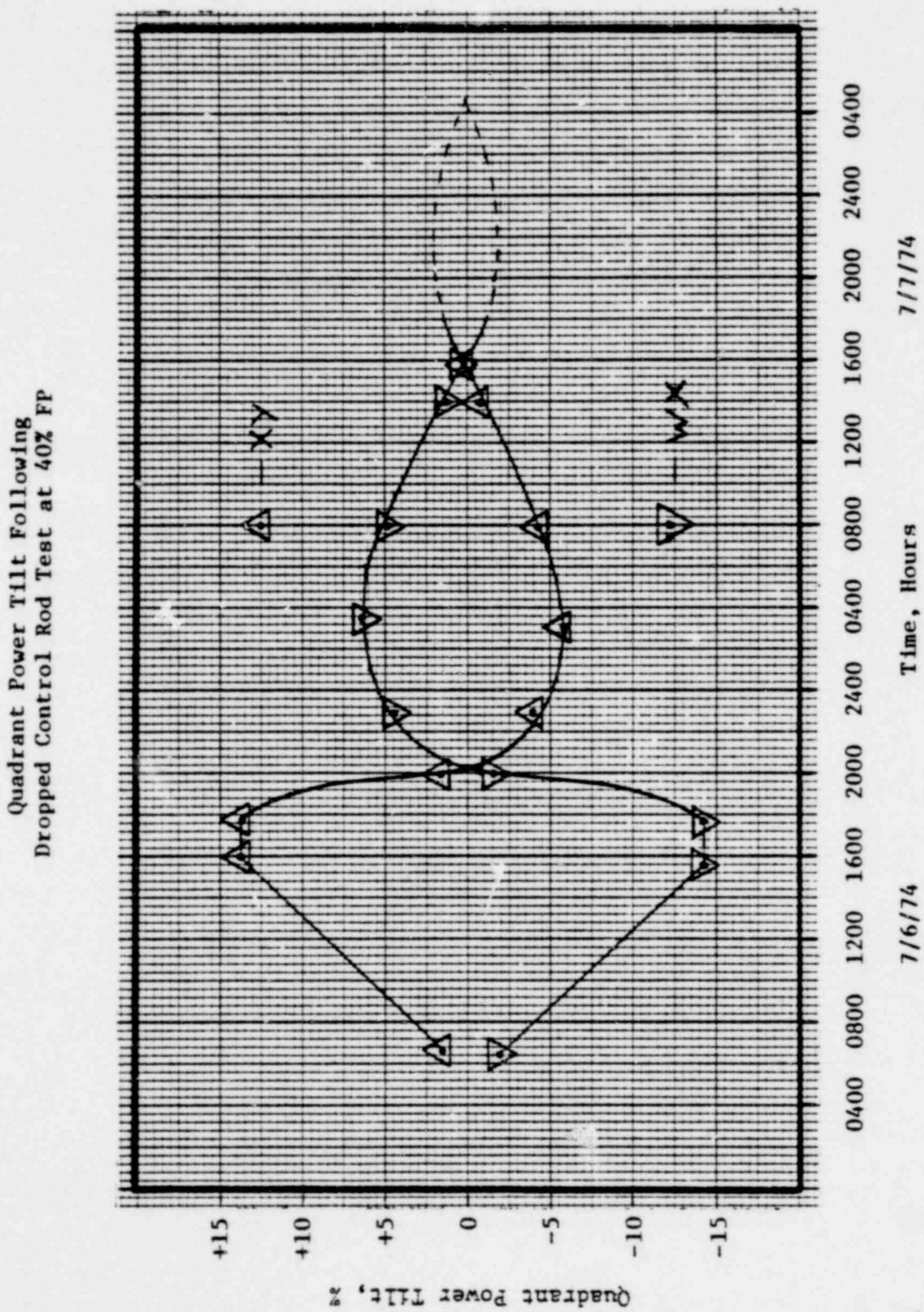

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FIGURE 5.7-4

5.8 PSEUDO CONTROL ROD EJECTION TEST

5.8.1 PURPOSE

The purpose of the Pseudo Control Rod Ejection Test was to verify the safety analysis relating to the accidental ejection of a control rod normally inserted in the core during power operation. This was accomplished by measuring the rod worth and associated core power distribution as the pseudo ejected control rod was withdrawn from the core. The ability of the rod drive control system to detect a control rod that deviates from its group average position was also determined during this test.

5.8.2 TEST METHOD

The test was conducted at 40% FP with equilibrium xenon established in the core. Control rod groups 1 through 5 were positioned at 100% withdrawn, group 6 at 75%, group 7 at 0% and group 8 at 13% to maintain core axial imbalance at 0 $\pm$2%. All Integrated Control System Stations were in automatic, except the Diamond Control Station which was in the manual sequence mode of operation.

Control rod 7-1 (core location H-8) was calculated to be the most reactive control rod in a rod ejection accident and this rod was selected for the measurement. Control rod 7-1 was withdrawn to 100% (in small increments) by performing a rod swap with control rod group 6. Differential reactivity worth measurements using the fast insert/withdrawal technique were performed on group 6 with control rod 7-1 positioned at 0% withdrawn and again at every 20% withdrawal interval. The total reactivity worth of rod 7-1 from 0% to 100% withdrawn was then obtained by integrating the differential worth data measured on group 6. Core power distribution measurements were taken with control rod 7-1 positioned at 100% withdrawn.

The asymmetric rod alarm and fault verification was conducted by obtaining a control rod map from the plant computer when the asymmetric rod alarm and fault conditions occurred as rod 7-1 was withdrawn from the core.

5.8.3 TEST RESULTS

The Pseudo Control Rod Ejection test was performed at 40% full power using control rod 7-1 (core location H-8) as the ejected rod. Figure 5.8-1 shows the location of control rod 7-1 in the core. A comparison of radial core power distributions with control rod 7-1 at 0% and 100% withdrawn is given in Figure 5.8-2.

Control rod group 6 was inserted from 74.2% to 54.5% withdrawn to compensate for the reactivity effects of removing rod 7-1 from the core. An integration of the measured differential worth data on group 6 gives 0.229%Δk/k as the reactivity worth of the ejected rod. A reactor power level of 40.59% full power was established at the start of the measurement and was maintained constant using the excore instrumentation. However, the total effect of withdrawing control rod 7-1 from the core was partially masked by the other group 7 rods and actual core power increased to 46.08% of full power with the ejected rod fully withdrawn. Using the measured value for the Power Coefficient of $-8.9\times10^{-5}\Delta k/k/\%$ FP, a correction factor of 0.0489%Δk/k was added to the integral worth result to give 0.278%Δk/k as the total ejected rod worth. This result is well below the acceptance criteria value of 0.49%Δk/k.

The effects of the pseudo ejected rod on core power distribution are summarized in Figure 5.8-2 and Tables 5.8-1 and 5.8-2. Core power distribution and thermal conditions were measured with control rod 7-1 at 0% and 100% withdrawn. The measured values of minimum DNBR and maximum LHR were 4.85 and 13.12 kw/ft, respectively, with rod 7-1 fully withdrawn. The measured value of LHR includes the uncertainty factor of 1.432 discussed in section 5.9.3.2. The maximum measured radial power peak was 2.38 in core location H-8. The effect of the ejected rod on quadrant power tilt was small, since the rod was located in the center of the core. Table 5.8-2 presents the results of a hand calculation of quadrant tilt which show a maximum positive tilt of +0.73% in the YZ quadrant. Core axial power imbalance shifted to the bottom of the core during the measurement due to group 6 insertion from 74.2% to 54.5% withdrawn.

The control rod asymmetric alarm and asymmetric fault conditions were verified during the withdrawal of rod 7-1. The control rod asymmetric alarm should occur when rod 7-1 is 5% ±1% above the group 7 average position and the asymmetric fault condition should occur when rod 7-1 is 6.5% ±1% above the group average. Control rod group 7 was at 2.66% withdrawn when the asymmetric alarm occurred with 7-1 at 8.0% withdrawn. Control rod group 7 was at 2.88% withdrawn when the asymmetric fault occurred with rod 7-1 at 10.0% withdrawn.

5.8.4 CONCLUSIONS

The measured worth of the pseudo ejected control rod was 0.278%Δk/k, which is well below the 0.65%Δk/k limit set in Technical Specifications 3.5.2. The measured values of maximum LHR and minimum DNBR were 13.12 kw/ft and 4.85, respectively, with the ejected rod at 100% withdrawn. The maximum measured radial power peak was 2.38 in the fuel assembly containing the ejected control rod.

The asymmetric control rod condition was detected by the rod drive control system within the acceptance criteria limits.

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TABLE 5.3-1

CORE POWER DISTRIBUTION AND THERMAL HYDRAULIC DATA

DURING THE PSEUDO ROD EJECTION TEST

ROD POSITION (% Wd)	INCORE IMBALANCE (% Wd)	QUADRANT TILT				MAXIMUM RADIAL PEAK	MAXIMUM LHR (1) (kw/ft)	MINIMUM DNBR	FUEL ASSEMBLY LOCATION
		WX	XY	WZ	YZ				
00	-1.90	-0.28	+0.26	-0.21	+0.22	1.40	6.95	9.28	K-10
100	-8.37	-0.28	+0.01	-0.15	+0.42	2.38	13.12	4.84	H-08

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(1) The values of LHR include the 1.432 uncertainty factor.

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HAND CALCULATION of QUADRANT POWER TILT from CORRECTED INCORE DETECTOR READINGS

Control Rod Group Positions
Gps 1-4 100% wd Gp 6 54.5% wd
Gp 5 100% wd Gp 7 2.0% wd
Gp 8 13.0% wd

Core Power 40 % FP
Boron Conc. 1188 ppm
Core Burnup 3.75 EFPD

Xenon Conditions
ρ Worth 1.92 %Δk/k
Equil. Conc. Yes
(yes or no)

Date 7/5/74, Time 1220, Initials of Analyst JAM

QUADRANT WX		QUADRANT XY		QUADRANT WZ		QUADRANT YZ	
Core Loc.	Level	Core Loc.	Level	Core Loc.	Level	Core Loc.	Level
C-6 1	211	C-10 1	213	K-5 1	368	K-11 1	376
2	235	2	243	2	511	2	524
3	242	3	248	3	549	3	554
4	283	4	286	4	566	4	552
5	215	5	220	5	394	5	396
6	135	6	142	6	270	6	262
7	48	7	53	7	114	7	114
E-7 1	381	E-9 1	375	L-3 1	203	L-13 1	206
2	518	2	519	2	231	2	240
3	549	3	534	3	264	3	250
4	550	4	549	4	293	4	302
5	389	5	365	5	201	5	219
6	239	6	257	6	136	6	146
7	92	7	112	7	47	7	59
F-3 1	206	F-13 1	204	M-7 1	364	M-9 1	395
2	240	2	236	2	523	2	527
3	259	3	256	3	544	3	541
4	293	4	286	4	537	4	554
5	223	5	227	5	368	5	395
6	140	6	147	6	269	6	257
7	55	7	59	7	117	7	101
G-5 1	372	G-11 1	371	O-6 1	210	O-10 1	213
2	526	2	517	2	245	2	244
3	553	3	538	3	250	3	247
4	542	4	551	4	283	4	283
5	393	5	387	5	209	5	207
6	272	6	261	6	137	6	126
7	113	7	109	7	49	7	54

WX TOTAL = 8274 XY TOTAL = 8265 WZ TOTAL = 8252 YZ TOTAL = 8344

$$\text{QUAD AVERAGE} = \frac{\text{WX TOT} + \text{WZ TOT} + \text{XY TOT} + \text{YZ TOT}}{4} = 8283.75$$

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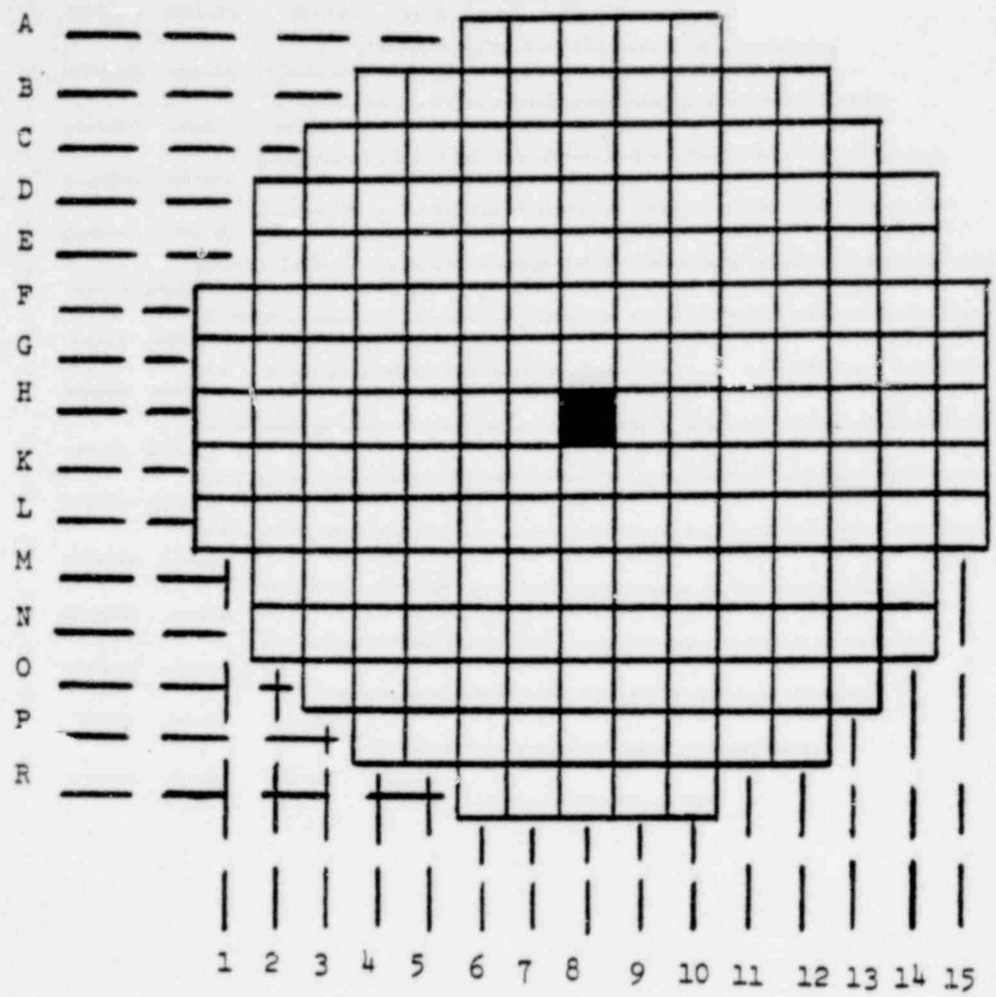
$$\text{TILT WX} = \left[\frac{\text{WX TOTAL}}{\text{QUAD AV.}} - 1\right] \times 100 = -.12\% \qquad \text{TILT WZ} = \left[\frac{\text{WZ TOTAL}}{\text{QUAD AV.}} - 1\right] \times 100 = -.38\%$$

$$\text{TILT XY} = \left[\frac{\text{XY TOTAL}}{\text{QUAD AV.}} - 1\right] \times 100 = -.23\% \qquad \text{TILT YZ} = \left[\frac{\text{YZ TOTAL}}{\text{QUAD AV.}} - 1\right] \times 100 = +.73\%$$

TABLE 5.8-2

EJECTED ROD WORTH AND LOCATION AT 40% FP PLATEAU



CR Group No.	Core Location	Calculated CR Worth % Δk/k	Measured CR Worth % Δk/k	Technical Specification Limit @100% FP % Δk/k
7	H-8	0.49	0.278	0.65

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FIGURE 5.8-1

COMPARISON OF CORE POWER DISTRIBUTIONS WITH EJECTED ROD AT 0% AND 100% WITHDRAWN, UNDER EQUILIBRIUM XENON, 40% FP CONDITIONS

CORE CONDITIONS

	H-8 at 0%		H-8 at 100%			H-8 at 0%		H-8 at 100%	
GPS 1-4 at	100	%Wd	100	%Wd	CORE POWER LEVEL	40	%FP	40	%FP
5 at	100	%Wd	100	%Wd	BORON CONC.	1188	ppm	1188	ppm
6 at	75	%Wd	54.5	%Wd	CORE BURNUP	3.75	EFPD	3.75	EFPD
7 at	2	%Wd	2	%Wd	AXIAL IMBALANCE	-1.90	%FP	-8.57	%FP
8 at	13	%Wd	13.0	%Wd	MAX. QUADRANT TILT.	+ .28	%	+ .42	%

Date 7/5/74, Time 1220, Initials of Analyst DAL

1.106 2.376	1.300 1.982	1.339 1.627	1.383 1.458	1.176 1.093	1.273 1.166	1.375 1.253	.942 .850
	1.271 1.721	1.404 1.647	1.265 1.317	1.225 1.167	1.041 .959	1.028 .932	.780 .704
		1.268 1.371	1.282 1.277	.983 .915	.971 .889	.721 .663	.456 .413
			1.025 .938	1.089 .987	.853 .767	.696 .632	
				.920 .826	.840 .757	.476 .429	
					.545 .490		

℄ ℄

X.XX X.XX	Ejected Rod at 0% Wd Ejected Rod at 100% Wd

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FIGURE 5.8-2

5.9 CORE POWER DISTRIBUTIONS

5.9.1 PURPOSE

Detailed core power distribution measurements were performed under steady state conditions during the power escalation program to verify that the core axial imbalance, quadrant power tilt, maximum linear heat rate and minimum DNBR do not exceed their specified limits. A summary of the steady-state core power distributions presented in this section is given in Table 5.9-1.

The specific acceptance criteria applied to the measured core power distributions are listed below:

(a) The combination of reactor thermal power and reactor power imbalance (power fraction in top half of the core minus power fraction in bottom half of the core) shall not exceed the safety limit as defined by the locus of points for the specified flow established in Technical Specification Figure 2.1-2 (Figure 5.9-1).

(b) The quadrant power tilt, as defined below, does not exceed the limits specified in Technical Specification 3.5.2.4.

$$\text{Quadrant Power Tilt} = \left[\frac{\text{Power in any Core Quadrant}}{\text{Average Quadrant Power}} - 1\right] \times 100$$

(c) The minimum value of DNBR shall be greater than 1.55.

(d) The maximum linear heat rate shall not exceed the curve of Figure 5.9-2 (LOCA limit) for all combinations of reactor thermal power and power imbalance that lie within the area bounded by the curve of Figure 5.9-3. For any combinations of reactor power and power imbalance that lie outside of the curve of Figure 5.9-3, the maximum linear heat rate shall not exceed 19.60 kw/ft (fuel melt limit).

5.9.2 TEST METHOD

Core power distribution measurements were performed at the major power plateaus of the test program (15%, 40%, 76% and 100% full power) under steady state conditions for specified control rod configurations. To provide the best comparison between measured and predicted results, three-dimensional equilibrium xenon conditions were established for the measurements at 40, 76 and 100% full power. The first step to establish three-dimensional xenon equilibrium was to borate/deborate the reactor coolant system, as required, to maintain the controlling rod groups within plus or minus two percent of the desired position during and after escalation to the next power plateau. The axial power shaping rods were moved, as required, to establish the predicted offset on the core. Once the measurement power level was achieved, the axial power shaping rods were maintained in a constant position for six to eight hours prior to taking data.

Data collected for the measurements consisted of detailed power distribution information at 364 core locations from the incore detector system and the worst case core thermal conditions calculated by the plant computer. The calculated core power distributions for various core power levels, axial imbalances, control rod configurations, burnup, boron concentrations and xenon conditions were provided for comparison with test results from the three-dimensional PDQ-7 code with thermal feedback.

5.9.3 TEST RESULTS

5.9.3.1 Steady State Power Distributions

Steady-state equilibrium xenon core power distribution measurements were performed at the major test plateaus of the power escalation sequence for specific control rod patterns, boron concentrations, axial imbalances and core burnups. The measured results are tabulated in Tables 5.9-2 through 5.9-5. The tables present a complete 1/8 core power distribution map using the corrected incore detector outputs from 7 levels of 29 fuel assemblies which describe the entire core, assuming eighth core symmetry. A summary of the four cases studied in this report is given in Table 5.9-1 which gives the core power level, core burnup, control rod pattern, boron concentration, xenon conditions, axial imbalance, maximum quadrant tilt, minimum DNBR, maximum LHR and power peaking data for each measurement.

Core power distribution calculations at steady-state equilibrium xenon conditions were performed using the three-dimensional PDQ-07 code with thermal feedback. The four cases reported in this section are compared with the PDQ-07 results in Figure 5.9-4 through 5.9-7 to demonstrate the degree of agreement between the measured and calculated radial core power distributions. As can be seen from these figures, the comparison between measured and calculated distributions shows good agreement. Differences between measured and calculated results are attributed to differences in core conditions that were assumed for the calculations as compared to those that existed at the time of measurement.

The results of measured core power distributions show maximum local peaking factors between 1.80 and 1.87 for the four cases with the maximum value of 1.87 measured at 15% FP and the minimum value of 1.80 measured at 100% FP. In all cases, the maximum local peaking factor was below the 2.67 limit given in Technical Specification 3.1. Calculations were performed to determine the maximum linear heat rate using the maximum local peaking factor measured in the four cases. The values of maximum linear heat rate were extrapolated to the overpower trip setpoint (105.5% FP) and the central fuel melt limit (112% FP). Examination of Table 5.9-6 shows that all extrapolated values of maximum linear heat rate were below the Technical Specification values of 17.1 kw/ft for the LOCA limit and 19.6 kw/ft for the fuel melt limit. Similarly, the extrapolated values of DNBR were well above the Technical Specifications minimum value of 1.30.

5.9.3.2 Minimum DNBR and Maximum LHR Calculations

5.9.3.2.1 Minimum DNBR Determination

Minimum values of DNBR were calculated for the core power distributions taken during the test program. The results of the DNBR calculations, are plotted in Figure 5.9-8. The results show that the minimum DNBR is greater than the acceptance criteria value of 1.55. The following analysis was used to determine the DNBR whenever steady state core power distributions were obtained.

(a) From each core power distribution, the fuel assembly which yielded the worst-case DNBR was selected.

(b) Upon selection of the worst-case assembly, a radial peaking factor was calculated and segment power levels were converted into axial peaking factors.

(c) The radial peaking factors were adjusted by a factor of 1.05 (the local pin peaking multiplier).

(d) The radial and axial flux data from (b) and (c) were incorporated into the standard hot channel analysis and analyzed for operation at full power.

All cases studied showed that the measured minimum DNBR values are greater than the minimum test acceptance criteria limit of 1.55. A minimum DNBR margin of 109 percent was observed after extrapolation to 112% FP.

5.9.3.2.2 Maximum Linear Heat Rate Determination

Analysis for determining maximum linear heat rate was performed in conjunction with minimum DNBR analysis. After selection of the worst case assemblies and determination of the radial and axial peaking factors, the maximum linear heat rate for each of the measured core power distributions was determined by Equation 5.9-1.

$$MLHR = \frac{P_R \times P_A \times P_L \times Q_{Rate} \times FNT}{NA \times NP \times AL} \qquad \text{(Equation 5.9-1)}$$

where: MLHR = Maximum Linear Heat Rate (kw/ft)
P_R = Radial Peaking Factor
P_A = Axial Peaking Factor
P_L = Local Pin Peaking Multiplier (1.05)
Q_{Rate} = Rated Core thermal power (2535×10^3 KW)
FNT = Fraction of power generated in fuel (0.973)
NA = Number of fuel assemblies in core (177)
NP = Number of fuel pins in each assembly (208)
AL = Active length of fuel pin (12 ft)

The results of the maximum linear heat rate calculations performed for the power distributions presented in this section are summarized in Table 5.9-6. The results of these worst case values were extrapolated to the overpower trip setpoint (105.5% FP) and fuel melt (112% FP) limits for each measurement. Substantial linear heat rate margins were observed with a minimum margin of 42% after extrapolation to 112% FP.

In addition to the above analysis, an uncertainty factor was applied to the calculated values of linear heat rate during the startup test program to introduce additional conservatism in the comparison with acceptance criteria. The worst case values of linear heat rate determined during any core power distribution measurement were multiplied by 1.432 to account for uncertainties not incorporated in Equation 5.9-1. These uncertainties include:

(a) nuclear uncertainty
(b) power uncertainty
(c) fuel densification effect
(d) power spike effect
(e) quadrant power tilt effect

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In each case where uncertainties were considered for incorporation into the 1.432 factor, worst case assumptions were used to provide adequate conservatism. For example, the quadrant power tilt factor makes allowances for a quadrant tilt of 4% even though the measured values for tilt were much less than 4% during the test

program. The modified values of linear heat rate were then extrapolated and compared to the LOCA or fuel melt limit as specified in section 5.9.1 (d).

Table 5.9-6 includes the modified values of maximum linear heat rate determined during the four cases studied in this section. As can be seen from the table, the LOCA and fuel melt limits were not exceeded even with the worst case assumptions used.

5.9.3.3 Quadrant Power Tilt and Axial Power Imbalance

Table 5.9-1 presents the maximum observed quadrant power tilts measured by the incore detector system during the core power distribution measurements. Quadrant power tilt limits are established by the Technical Specifications in conjunction with control rod position limits to assure that the design peak heat rate criterion is not exceeded during normal power operation. The quadrant power tilt measured during operation at power has been well within the limits established in the Technical Specifications.

The core axial power imbalances measured in conjunction with the core power distribution of this section show that a substantial margin exists between the measured reactor power/power imbalance combinations and the limits set forth in Technical Specifications Figure 2.1-2 (Figure 5.9-1). The core power imbalance measured by this incore detector system for each power distribution studied is summarized in Table 5.9-1.

5.9.4 CONCLUSIONS

Core power distribution measurements were conducted at 15%, 40%, 76% and 100% of full power during the power escalation sequence under steady state equilibrium xenon conditions for specified control rod configurations. Comparison of the measured power distributions with the PDQ-07 results shows good agreement. For the three cases studied at 40%, 76% and 100% full power, the three largest measured and calculated radial peaks were chosen. In each case, the measured values were within 8% of the calculated results.

The results of the minimum DNBR and maximum LHR analyses are given in Table 5.9-6. The margins to the minimum DNBR limit of 1.55 and maximum LHR value of 17.1 kw/ft were 109% and 42%, respectively, after extrapolation to 112% FP. All quadrant power tilts and axial core imbalances measured during the power distribution tests were within the Technical Specifications and normal operational limits.

Measured Core Power Distributions and Core Thermal Conditions for Various Control Rod Patterns and Core Power Levels of 15%, 40%, 76% and 100% FP.

	15%	40%	76%	100%
Date	6/20/74	7/01/74	7/21/74	8/09/74
Time	0713	1010	0844	0103
Power level, % FP	16.4	40.47	77	99
Xenon Equilibrium	2-D	3-D	3-D	3-D
Rod Positions, % wd				
1-5	100	100	100	100
6	76	75	77	93
7	0	0	5	19
8	34	33	27	14.5
Core Burnup, EFPD	0.45	1.8	9.87	21.2
Boron Concentration, ppmb	1340	1207	1090	1101
Axial Imbalance, % FP	-2.0	-5.1	-12.43	-2.3
Maximum Quadrant Tilt, %	0.5	0.41	1.18	1.39
Minimum DNBR	25.82	10.28	5.42	4.01
Maximum LHR, kw/ft	1.86	4.49	8.33	10.06
Maximum Peaks				
Radial	1.44	1.40	1.39	1.37
Radial x Axial	1.87	1.80	1.81	1.83

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TABLE 5.9-1

MEASURED CORE POWER DISTRIBUTION RESULTS

Control Rod Group Positions

Gps 1-4	100% wd	Gp 6	76% wd
Gp 5	100% wd	Gp 7	0% wd
		Gp 8	34% wd

Core Power Level	16.4% FP
Boron Concentration	1340 ppm
Core Burnup	0.45 EFPD
Axial Imbalance	-2.0% FP

Xenon Conditions

Equilibrium Conc.	No	Yes or No
Reactivity Worth	NA	% $\Delta k/k$
Max Quadrant Tilt	0.5	%

1/8 Core FA Loc.	Incore Det. No.	Pmax/Pcore Local	P/P Fuel Assembly
H-08	1	1.15	1.11
G-08	2	1.73	1.32
F-08	4	1.74	1.32
E-08	10	1.83	1.41
D-08	14	1.61	1.19
C-08	21	1.72	1.31
B-08	30	1.87	1.44
A-08	37	1.22	0.94
G-09	3	1.67	1.27
F-10	12	1.66	1.29
E-11	26	1.43	1.06
D-12	41	1.15	0.89
C-13	52	0.66	0.52
F-09	6	1.83	1.42
E-09	5	1.62	1.26
D-09	15	1.60	1.25
C-09	29	1.38	1.04
B-09	31	1.31	1.07
A-09	45	1.00	0.78
E-10	17	1.68	1.30
D-10	27	1.35	0.99
C-10	28	1.24	0.97
B-10	44	0.87	0.69
A-10	46	0.56	0.44
D-11	33	1.43	1.08
C-11	42	1.09	0.84
B-11	49	0.80	0.64
C-12	48	1.03	0.80
B-12	51	0.56	0.44

TABLE 5.9-2

MEASURED CORE POWER DISTRIBUTION RESULTS

Control Rod Group Positions

Gps 1-4 100% wd Gp 6 75% wd

Gp 5 100% wd Gp 7 0% wd

Gp 8 33% wd

Core Power Level 40.47% FP

Boron Concentration 1207ppm

Core Burnup 1.8EFPD

Axial Imbalance -5.1% FP

Xenon Conditions

Equilibrium Conc. Yes Yes or No

Reactivity Worth 1.91 % $\Delta k/k$

Max Quadrant Tilt 0.41 %

1/8 Core FA Loc.	Incore Det. No.	Pmax/Pcore Local	P/P Fuel Assembly
H-08	1	1.39	1.12
G-08	2	1.66	1.29
F-08	4	1.74	1.32
E-08	10	1.78	1.38
D-08	14	1.59	1.17
C-08	21	1.66	1.29
B-08	30	1.80	1.38
A-08	37	1.20	0.93
G-09	3	1.61	1.26
F-10	12	1.64	1.26
E-11	26	1.47	1.04
D-12	41	1.16	0.92
C-13	52	0.69	0.55
F-09	6	1.80	1.40
E-09	5	1.59	1.25
D-09	15	1.59	1.23
C-09	29	1.33	1.04
B-09	31	1.24	1.04
A-09	45	1.00	0.79
E-10	17	1.72	1.29
D-10	27	1.42	1.00
C-10	28	1.25	0.97
B-10	44	0.88	0.72
A-10	46	0.55	0.45
D-11	33	1.42	1.09
C-11	42	1.11	0.85
B-11	49	0.86	0.69
C-12	48	1.07	0.84
B-12	51	0.60	0.47

TABLE 5.9-3

MEASURED CORE POWER DISTRIBUTION RESULTS

Control Rod Group Positions

Gps 1-4	100% wd	Gp 6	77% wd
Gp 5	100% wd	Gp 7	5% wd
		Gp 8	27% wd

Core Power Level	77% FP
Boron Concentration	1090 ppm
Core Burnup	9.87 EFPD
Axial Imbalance	-12.43% FP

Xenon Conditions

Equilibrium Conc.	Yes	Yes or No
Reactivity Worth	2.44	% Δk/k
Max Quadrant Tilt	1.18	%

1/8 Core FA Loc.	Incore Det. No.	Pmax/Pcore Local	P/P Fuel Assembly
H-08	1	1.41	1.14
G-08	2	1.67	1.29
F-08	4	1.76	1.33
E-08	10	1.81	1.38
D-08	14	1.68	1.21
C-08	21	1.70	1.26
B-08	30	1.74	1.35
A-08	37	1.15	0.91
G-09	3	1.58	1.26
F-10	12	1.66	1.26
E-11	26	1.52	1.05
D-12	41	1.22	0.92
C-13	52	0.69	0.54
F-09	6	1.79	1.39
E-09	5	1.65	1.28
D-09	15	1.70	1.24
C-09	29	1.41	1.05
B-09	31	1.23	0.99
A-09	45	0.95	0.75
E-10	17	1.81	1.30
D-10	27	1.50	0.99
C-10	28	1.32	0.97
B-10	44	0.87	0.74
A-10	4[illegible]	0.57	0.46
D-11	33	1.48	1.10
C-11	42	1.16	0.85
B-11	49	0.91	0.71
C-12	48	1.09	0.84
B-12	51	0.59	0.48

TABLE 5.9-4

MEASURED CORE POWER DISTRIBUTION RESULTS

Control Rod Group Positions

Gps 1-4 100% wd Gp 6 93% wd
Gp 5 100% wd Gp 7 19% wd
Gp 8 14.5% wd

Core Power Level 99% FP
Boron Concentration 1101ppm
Core Burnup 21.2EFPD
Axial Imbalance -2.3% FP

Xenon Conditions
Equilibrium Conc. Yes Yes or No
Reactivity Worth 2.62 % $\Delta k/k$
Max Quadrant Tilt 1.39 %

1/8 Core FA Loc.	Incore Det. No.	Pmax/Pcore Local	P/P Fuel Assembly
H-08	1	1.50	1.20
G-08	2	1.54	1.29
F-08	4	1.63	1.28
E-08	10	1.83	1.37
D-08	14	1.64	1.23
C-08	21	1.60	1.25
B-08	30	1.65	1.33
A-08	37	1.09	0.89
G-09	3	1.58	1.23
F-10	12	1.61	1.23
E-11	26	1.45	1.08
D-12	41	1.26	0.92
C-13	52	0.66	0.52
F-09	6	1.75	1.35
E-09	5	1.60	1.26
D-09	15	1.65	1.24
C-09	29	1.33	1.05
B-09	31	1.15	1.00
A-09	45	0.88	0.75
E-10	17	1.83	1.30
D-10	27	1.41	1.00
C-10	28	1.26	1.01
B-10	44	0.96	0.81
A-10	46	0.58	0.48
D-11	33	1.48	1.12
C-11	42	1.18	0.86
B-11	49	0.88	0.72
C-12	48	1.07	0.83
B-12	51	0.61	0.47

TABLE 5.9-5

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TABLE 5.9-6

MINIMUM DNBR AND MAXIMUM LINEAR HEAT RATE

MEASURED FOR STEADY STATE, EQUILIBRIUM XENON CONDITIONS

DATE TIME	POWER LEVEL (%FP)	INCORE IMBALANCE (%FP)	MAXIMUM PEAK (Pmax/Pcore)	AXIAL LOCATION (SEGMENT)	FUEL ASSEMBLY	MAXIMUM LHR(1) (kw/ft)	MAXIMUM LHR(2) (kw/ft)	MINIMUM DNBR
6/20/74	16.4	- 2.0	1.87	4	B-08	1.86	2.66	25.82
0713	105.5					11.97	17.11	4.01
	112.0					12.70	18.17	3.78
7/01/74	40.47	- 5.1	1.80	3	B-08	4.49	6.43	10.28
1010	105.5					11.70	16.76	3.94
	112.0					12.43	17.79	3.71
7/21/74	77	-12.43	1.81	2	E-08	8.33	11.93	5.42
0844	105.5					11.41	16.35	3.96
	112.0					12.12	17.35	3.73
8/09/74	99	- 2.3	1.83	4	E-08	10.06	14.41	4.01
0103	105.5					10.72	15.36	3.60
	112.0					11.38	16.30	3.25

(1) Worst case values of LHR without uncertainties

(2) Worst case values of LHR multiplied by 1.432 to account for measurement uncertainties

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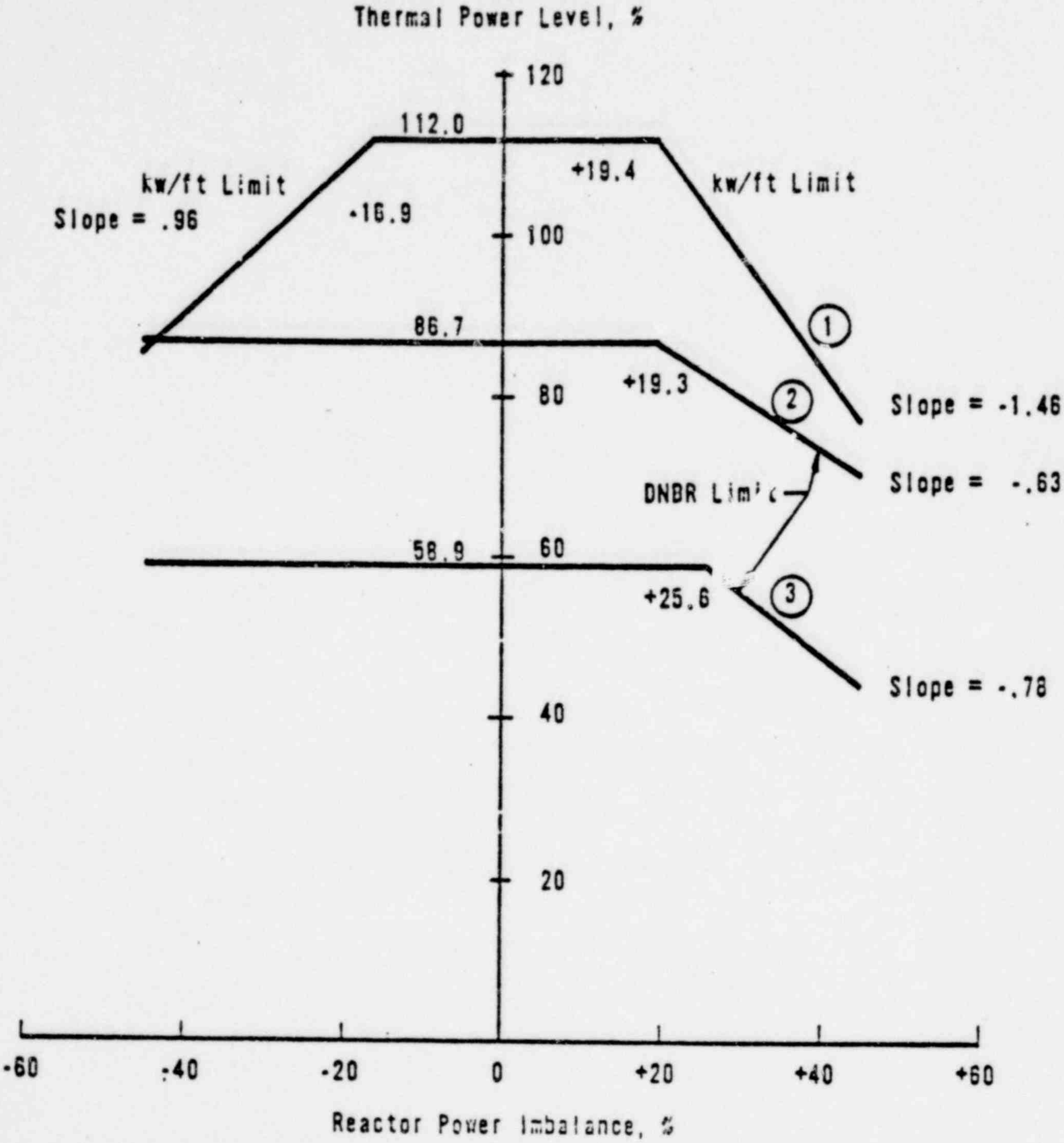
CORE PROTECTION SAFETY LIMITS

THREE MILE ISLAND NUCLEAR STATION UNIT I



CURVE	REACTOR COOLANT FLOW (LB/HR)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6

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FIGURE 5.9-1

LOCA LIMITED MAX MUM ALLOWABLE
LINEAR HEAT RATE
THREE MILE ISLAND NUCLEAR STATION UNIT 1

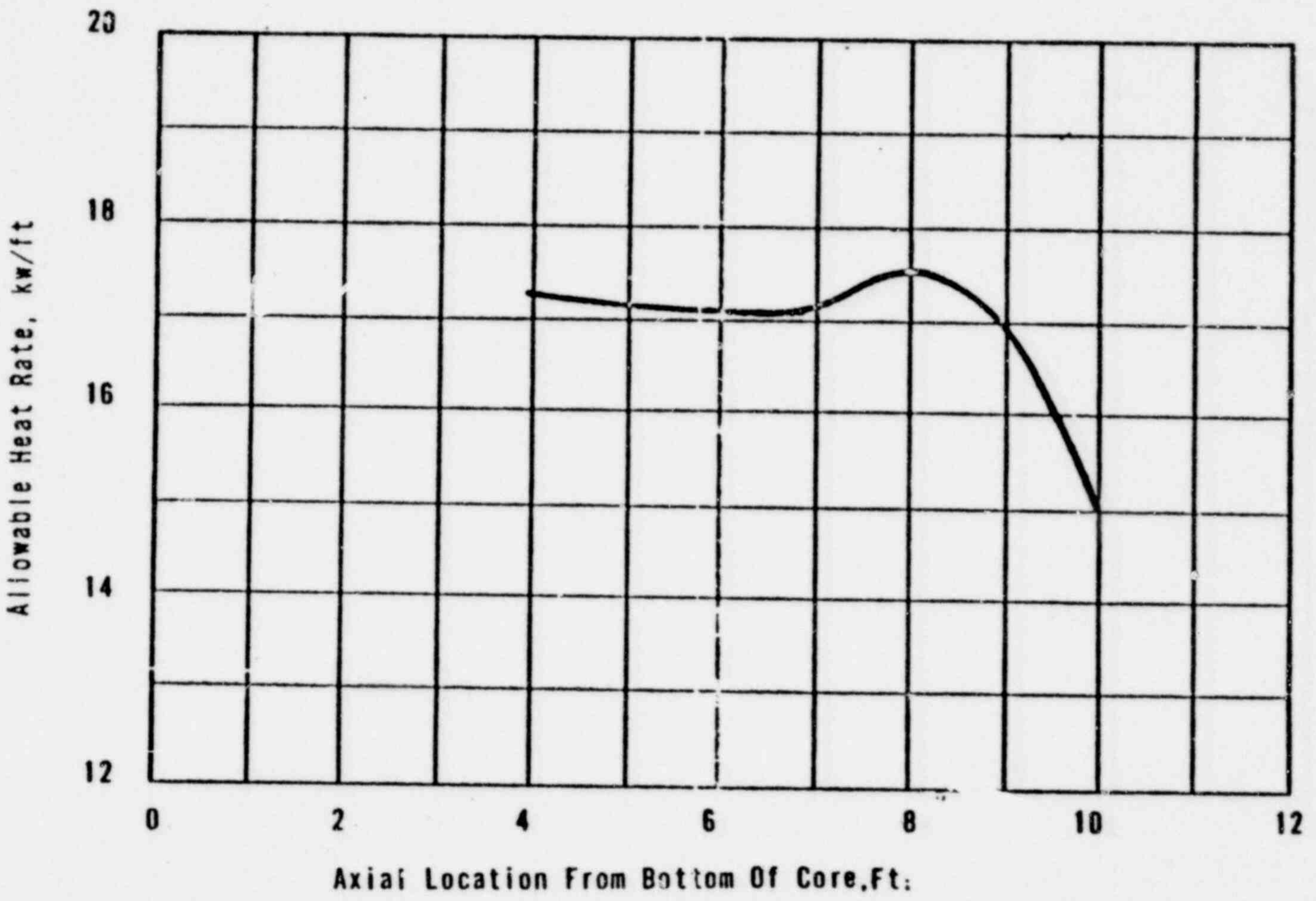

FIGURE 5.9-2

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OPERATIONAL POWER IMBALANCE ENVELOPE
THREE MILE ISLAND NUCLEAR STATION UNIT 1

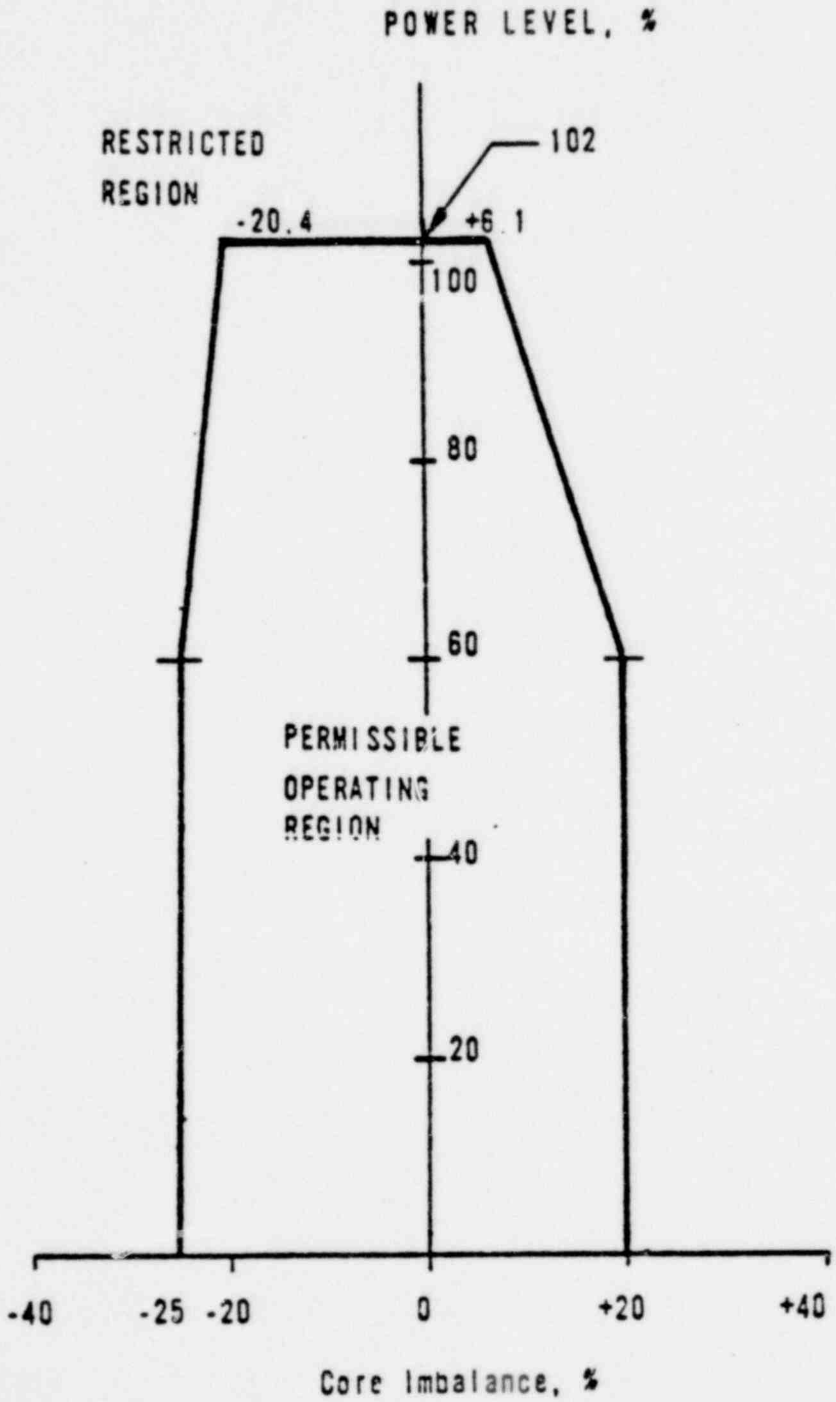

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FIGURE 5.9-3

COMPARISON OF MEASURED AND CALCULATED RADIAL CORE POWER DISTRIBUTIONS AT STEADY STATE DISTRIBUTIONS AND STEADY STATE, EQUILIBRIUM XENON, 15% FP CONDITIONS.

CORE CONDITIONS

	CALCULATED	ACTUAL		CALCULATED	ACTUAL
GPS 1-4 at	100 %Wd	100 %Wd	CORE POWER LEVEL	15 %FP	16.4 %FP
5 at	100 %Wd	100 %Wd	BORON CONC.	1441 ppm	1340 ppm
6 at	75 %Wd	76 %Wd	CORE BURNUP	0 EFPD	0.45 EFPD
7 at	0 %Wd	0 %Wd	AXIAL IMBALANCE	-5.04 %FP	-2.0 %FP
8 at	37.5 %Wd	34 %Wd	MAX. QUADRANT TILT.	0 %	0.5 %

1.04 1.11	1.37 1.32	1.36 1.32	1.53 1.41	1.19 1.19	1.30 1.31	1.33 1.44	0.90 0.94
	1.30 1.27	1.57 1.42	1.29 1.26	1.31 1.25	1.00 1.04	0.93 1.07	0.73 0.78
		1.32 1.29	1.38 1.30	0.96 0.99	0.92 0.97	0.60 0.69	0.42 0.44
			1.07 1.06	1.09 1.08	0.77 0.84	0.63 0.64	
				0.86 0.89	0.83 0.80	0.50 0.44	
					0.59 0.52		

X.XX X.XX	Calculated Results Measured Results, Group 36

1414 156

FIGURE 5.9-4

COMPARISON OF MEASURED AND CALCULATED RADIAL CORE POWER DISTRIBUTIONS AT STEADY STATE DISTRIBUTIONS AND STEADY STATE, EQUILIBRIUM XENON, 40% FP CONDITIONS.

CORE CONDITIONS

	CALCULATED	ACTUAL
GPS 1-4 at	100 %Wd	100 %Wd
5 at	100 %Wd	100 %Wd
6 at	75 %Wd	75 %Wd
7 at	0 %Wd	0 %Wd
8 at	30.8 %Wd	33 %Wd

	CALCULATED	ACTUAL
CORE POWER LEVEL	40 %FP	40.47 %FP
BORON CONC.	1430 ppm	1207 ppm
CORE BURNUP	1.6 EFPD	1.8 EFPD
AXIAL IMBALANCE	-5.7 %FP	-5.11 %FP
MAX. QUADRANT TILT.	0 %	0.41 %

0.98 1.12	1.29 1.29	1.26 1.32	1.44 1.38	1.12 1.17	1.28 1.29	1.35 1.38	0.96 0.93
	1.21 1.26	1.47 1.40	1.21 1.25	1.27 1.23	1.00 1.04	0.97 1.04	0.78 0.79
		1.24 1.26	1.32 1.29	0.95 1.00	0.94 0.97	0.64 0.72	0.48 0.45
			1.02 1.04	1.11 1.09	0.81 0.85	0.70 0.69	
				0.90 0.92	0.90 0.84	0.57 0.47	
					0.66 0.55		

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X.XX X.XX	Calculated Results Measured Results, Group 36

1414 157

FIGURE 5.9-5

COMPARISON OF MEASURED AND CALCULATED RADIAL CORE POWER DISTRIBUTIONS AT STEADY STATE DISTRIBUTIONS AND STEADY STATE, EQUILIBRIUM XENON, 76% FP CONDITIONS.

CORE CONDITIONS

	CALCULATED	ACTUAL		CALCULATED	ACTUAL
GPS 1-4 at	100%Wd	100%Wd	CORE POWER LEVEL	75 %FP	77.1 %FP
5 at	100%Wd	100%Wd	BORON CONC.	1134 ppm	1090 ppm
6 at	75%Wd	77.3%Wd	CORE BURNUP	15.2 EFPD	9.87 EFPD
7 at	0%Wd	5.3%Wd	AXIAL IMBALANCE	-13.0 %FP	-12.43 %FP
8 at	30.8%Wd	27.3%Wd	MAX. QUADRANT TILT.	0 %	1.18 %

1.00 1.14	1.32 1.29	1.30 1.33	1.46 1.38	1.13 1.21	1.25 1.26	1.28 1.35	0.90 0.91
	1.25 1.26	1.51 1.39	1.24 1.28	1.27 1.24	0.99 1.05	0.92 0.99	0.74 0.75
		1.27 1.26	1.34 1.30	0.96 0.99	0.94 0.97	0.62 0.74	0.46 0.46
			1.05 1.05	1.12 1.10	0.82 0.85	0.69 0.71	
				0.91 0.92	0.91 0.84	0.58 0.48	
					0.68 0.54		

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X.XX X.XX	Calculated Results Measured Results, Group 36

1414 158

FIGURE 5.9-6

COMPARISON OF MEASURED AND CALCULATED RADIAL CORE POWER DISTRIBUTIONS AT STEADY STATE DISTRIBUTIONS AND STEADY STATE, EQUILIBRIUM XENON, 100% FP CONDITIONS.

CORE CONDITIONS

	CALCULATED	ACTUAL
GPS 1-4 at	100 %Wd	100 %Wd
5 at	100 %Wd	100 %Wd
6 at	87.5 %Wd	93 %Wd
7 at	12.5 %Wd	14.5 %Wd
8 at	22.5 %Wd	21.2 %Wd

	CALCULATED	ACTUAL
CORE POWER LEVEL	100 %FP	99 %FP
BORON CONC.	968 ppm	1101 ppm
CORE BURNUP	25 EFPD	21.2 EFPD
AXIAL IMBALANCE	-2.5 %FP	-2.3 %FP
MAX. QUADRANT TILT.	0 %	1.39 %

1.02 1.20	1.28 1.29	1.27 1.28	1.43 1.37	1.16 1.23	1.24 1.25	1.24 1.33	0.88 0.89
	1.22 1.23	1.45 1.35	1.23 1.26	1.27 1.24	1.00 1.05	0.92 1.00	0.74 0.75
		1.25 1.23	1.33 1.30	0.98 1.00	0.95 1.01	0.66 0.81	0.48 0.48
			1.08 1.08	1.13 1.12	0.83 0.86	0.72 0.72	
				0.92 0.92	0.90 0.83	0.58 0.47	
					0.67 0.52		

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X.XX Calculated Results
X.XX Measured Results, Group 36

FIGURE 5.9-7

HOT CHANNEL MINIMUM DNBR VS REACTOR POWER

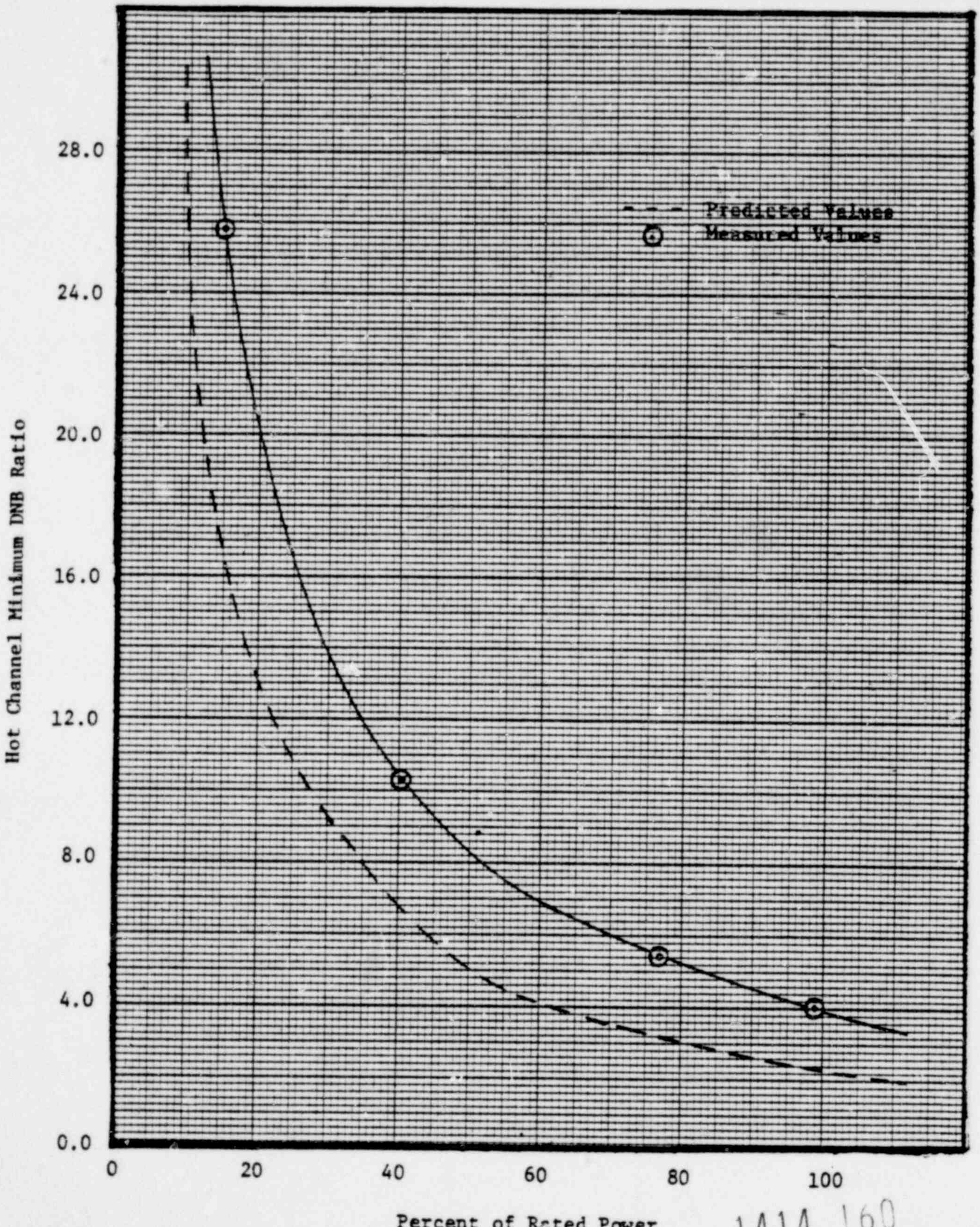

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FIGURE 5.9-8

5.10 NUCLEAR STEAM SYSTEM HEAT BALANCE

5.10.1 PURPOSE

The NSS Heat Balance test was performed as required during power escalation with the following objectives:

(a) To determine the core thermal power between 0% and 15% power using the core differential temperature method.

(b) To determine the core thermal power based upon a primary side calorimetric measurement.

(c) To determine core thermal power based upon a secondary side calorimetric measurement.

(d) To provide a value for "Core Thermal Power" to be used in the calibration of the power range nuclear instrumentation.

(e) To demonstrate the accuracy of the computer calculated heat balance.

5.10.2 TEST METHOD

Steady state conditions were established, as indicated below, at each of the specified power plateaus prior to recording test data.

(a) All four reactor coolant pumps operating.

(b) The plant computer operating.

(c) Reactor coolant pressure at 2155 $\pm$25 psig.

(d) RCS average temperature at 579 $\pm 2^{\circ}$F (except for 0% to 15% FP)

(e) Feedwater temperature stable with less than $\pm 2^{\circ}$F change in 15 min.

(f) Feedwater average flow rate stable with less than $\pm$3% change in 15 min.

When the required equilibrium conditions were verified, the following data was collected over a 30 minute interval.

(a) Selected computer data points.

(b) Computer group 32, Reactor Coolant Heat Balance.

(c) Selected reactimeter-patch panel points of primary and secondary parameters.

(d) Computer group 41, Steam Generator Performance.

Calculation of core thermal power by primary and secondary heat balance was then done by hand and compared to the computer results. After the computer calculated values were verified at each major power plateau, the computer values for Core Thermal Power were used for core power determination. Since the primary and secondary heat balance calculations are inaccurate below 15% FP, core power was

determined by core differential temperature during the initial escalation from 0% to 15% FP. Determination of reactor coolant flow rate from heat balance data was performed by setting the primary heat balance equal to the secondary heat balance on each loop and solving for the loop primary flow rate. Total coolant flow was then obtained from the sum of the loop flows.

5.10.3 TEST RESULTS

Initial data analysis at low power levels indicated a possible source of error in the data taking method for the hand calculations. The computer used time averaged values for the calculations, wheras the hand calculations were performed using single data sets. Normal plant oscillations introduced sizeable errors in the hand calculated values based upon single data sets when compared to the time averaged computer results. Average parameter values over a 15-20 minute period were then used for the hand calculations to obtain more representative data.

Core thermal power was determined from core differential temperature during the initial escalation to 15% FP. Figure 5.10-1 presents the normalized reactor coolant ΔT versus core power relationship used as a primary side heat balance.

The results of primary and secondary heat balance measurements performed from 30% to 100% FP are presented and compared in Table 5.10-1. Good agreement was found between the hand and computer calculated heat balances. As can be seen from Table 5.10-1, the primary heat balances were consistently lower than the secondary balances. This difference is due mostly to the indicated reactor coolant flow being slightly less than the actual flow, as determined by setting primary and secondary heat balances equal.

Reactor coolant flows calculated by setting the primary side heat balance equal to the secondary side balance are listed in Table 5.10-2. Preliminary calculations show that the indicated flow is an average of 2% less than the calculated flow. The best estimate for total reactor coolant flow at 100% FP is 108.6% of design, based on a design flow rate of 131.32×10^6 lbm/hr.

5.10.4 CONCLUSIONS

Primary and secondary heat balance calculations were performed during power escalation. Good agreement was found between the hand and computer calculated values for core thermal power. Primary side heat balances were consistently 2% lower than secondary side heat balances; therefore, actual or calculated RC flow is about 2% higher than indicated RC flow. The best estimate for total reactor coolant flow at 100% FP using normal plant temperature, pressure and flow indications is 108.6% of the design flow rate.

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TABLE 5.10-1

HEAT BALANCE CALCULATION SUMMARY

Nominal Power (% FP)	Pc (MWt)	Ph (MWt)	Diff Ph-Pc (% FP)	Sc (MWt)	Sh (MWt)	Diff Sc-Sh (% FP)	Diff Pc-Sc (% FP)	Diff Ph-Sh (% FP)	Bc (MWt)	Bh (MWt)	Diff Bh-Bc (% FP)
30	774	781	+.28	814	794	-.78	-1.6	-.51	781	770	-.43
35	934	894	-1.6	971	955	-.63	-1.5	-2.4	944	908	-1.4
40	1054	1013	-1.6	1083	1066	-.67	-1.1	-2.1	1063	1028	-1.4
50	1257	1202	-2.2	1285	1265	-.79	-1.1	-2.5	1269	1226	-1.7
65	1677	1599	-3.1	1695	1691	-.16	-.71	-3.6	1688	1655	-1.3
75	1907	1880	-1.1	1941	1919	-.87	-1.3	-1.5	1931	1908	-.91
85	2120	2100	-.79	2139	2167	-1.1	-.75	-2.6	2136	2155	+.75
95	2374	2347	-1.1	2409	2388	-.83	-1.4	-1.6	2407	2385	-.87
100	2454	2426	-1.1	2504	2486	-.71	-2.0	-2.4	2503	2484	-.75
100	2438	2422	-.63	2488	2466	-.87	-2.0	-1.7	2487	2464	-.91
Average Deviation:			-1.3			-.74	-1.3	-2.1			-.89

Pc: Primary computer heat balance

Ph: Primary hand heat balance

Sc: Secondary computer heat balance

Sh: Secondary hand heat balance

Bc: Best estimate, computer

Bh: Best estimate, hand

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TABLE 5.10-2

REACTOR COOLANT FLOW CALCULATION FROM PRIMARY - SECONDARY HEAT BALANCE

Date	Time	Nominal Power (% FP)	Indicated Flow ($x10^6$ lbm/hr)	Calculated Flow ($x10^6$ lbm/hr)	Difference Indicated-Actual (%)
7/14/74	0320	75	139.40	142.21	-1.98
8/03/74	0500	85	139.73	143.73	-2.78
8/03/74	1445	95	139.95	141.83	-1.33
8/06/74	0320	100	139.59	142.64	-2.14

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PRIMARY SIDE HEAT BALANCE
FROM NORMALIZED CORE ΔT

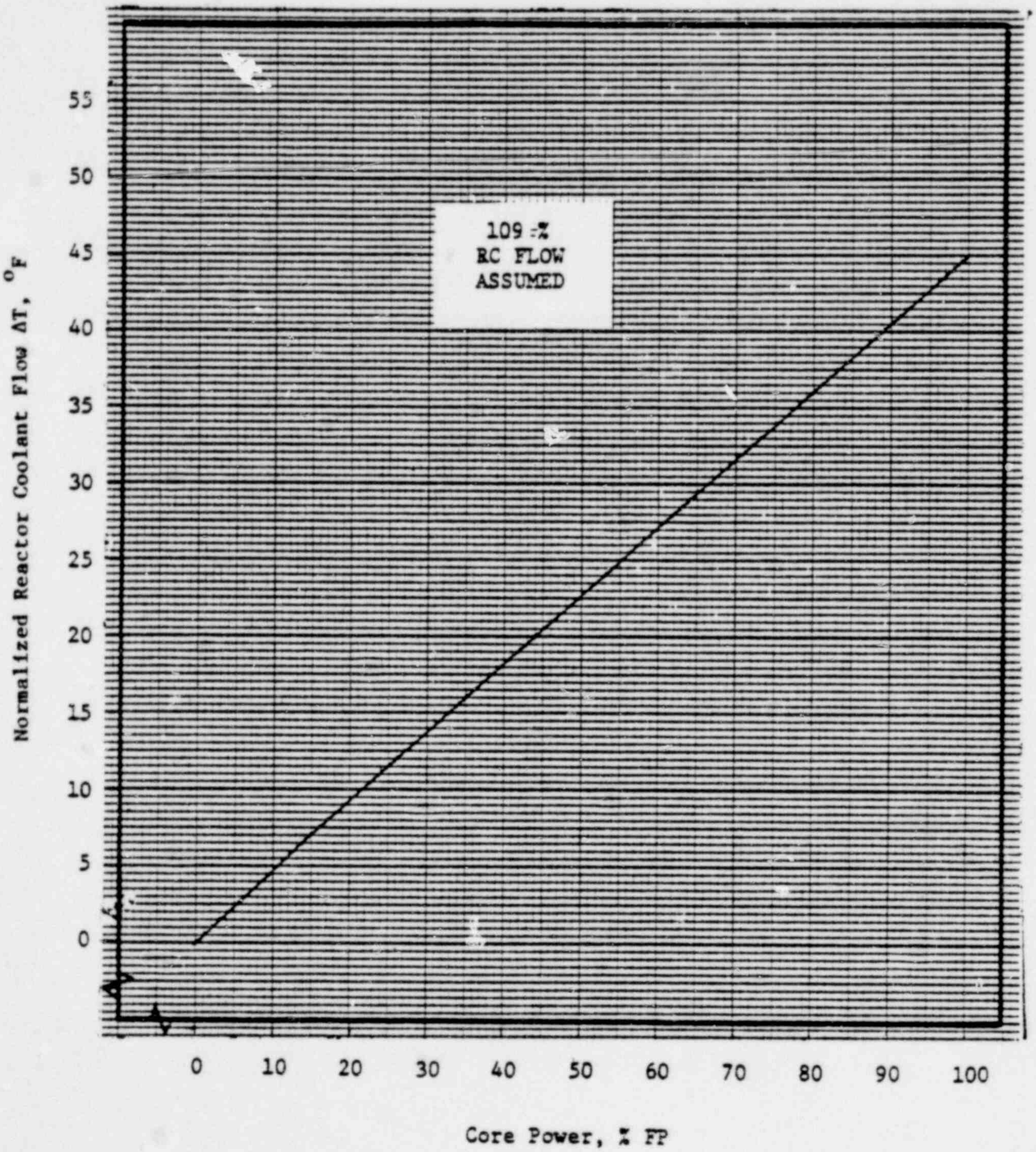

FIGURE 5.10-1

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5.11 REACTIVITY DEPLETION VERSUS BURNUP

5.11.1 PURPOSE

The purpose of the Reactivity Depletion vs. Burnup test was to determine the core excess reactivity based upon measured critical boron concentrations at various times in core life. Once the core excess reactivity is known, it can be used as the basis in a reactivity anomaly calculation.

5.11.2 TEST METHOD

The depletion test was performed with the reactor at 100% FP with 2-D equilibrium xenon established. The normal operating control rod configuration was used with group 6 at 87.5% and group 7 at 12.5% withdrawn. The pressurizer was sprayed previous to conducting the test to assure that RCS, pressurizer and make-up tank boron concentrations were at equilibrium.

Three separate make-up tank, pressurizer and RCS boron samples were taken when all required steady state conditions were satisfied. The test was scheduled to be run at 20, 30, 40 and 50 EFPD during the startup test program.

5.11.3 TEST RESULTS

The result of the measurements made at 22.0 EFPD are plotted in Figure 5.11-1. The average measured RCS boron concentration was 1091.67 ppm. This value was adjusted by 4.51 ppm to obtain 1087.16 ppm as the final result. The adjustment was made to account for minor deviations in control rod position and core power from the values used in the calculations. The measured critical boron concentration was in good agreement with the predicted value of 1060 ppm.

The scheduled measurements at 30, 40 and 50 EFPD were not performed as part of the test program but will be conducted as part of the normal plant surveillance testing.

5.11.4 CONCLUSIONS

The measured critical boron concentration at 22.0 EFPD and 100% FP conditions was within 30 ppm of the predicted result and well within the acceptance criteria value of 86 ppm.

CRITICAL BORON CONCENTRATION VS BURNUP

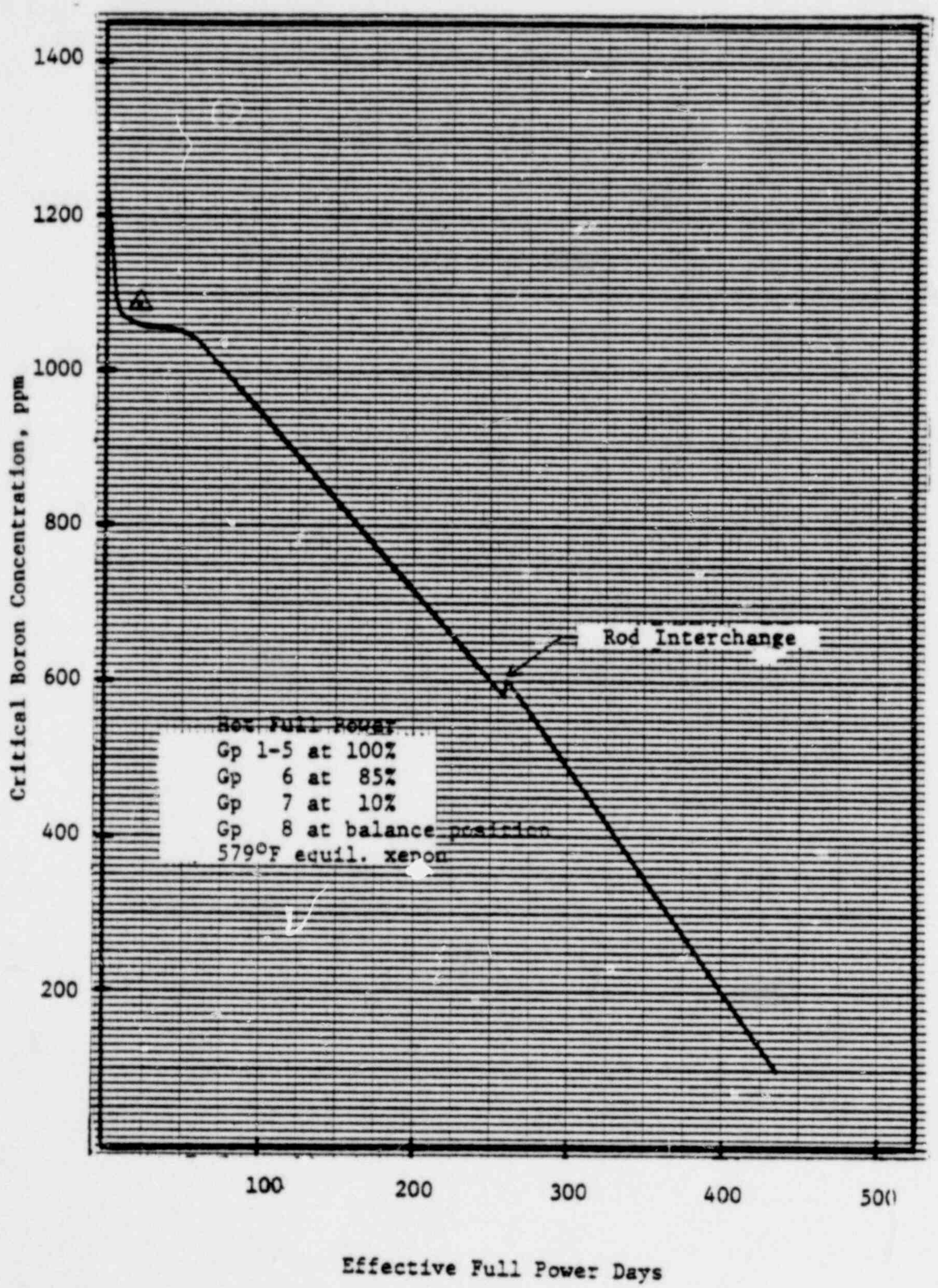

FIGURE 5.11-1

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5.12 NEUTRON NOISE MEASUREMENTS

Neutron noise data was recorded on the TMI Unit I core during the startup test program to serve as baseline data for future periodic measurements. Signals from the four power range detectors and eight other primary and secondary plant parameters were recorded on a Honeywell model 3600 FM tape recorder after establishing steady state three-dimensional equilibrium xenon conditions at 40%, 76% and 100% of full power. A brief summary of the measurement conditions is given below. Initial analysis of the data indicates no major differences from the expected neutron noise signatures.

Power Level	Date	EFP Days	Boron (ppm)	Rod Positions, % wd Gp 6	Gp 7	Gp 8
40%	7/1/74	1.9	1202	76	4	32
76%	7/19/74	8.5	1120	74	0	30
100%	8/8/74	20.7	1104	94	20	12

6.0 NUCLEAR STEAM SYSTEM PERFORMANCE

Several tests were performed during power escalation and operation at full power to monitor the performance of the nuclear steam system. The test results presented in this section provide a discussion of reactor coolant system performance under steady state and transient conditions, reactor coolant pump performance and a summary of radioactive waste management and primary and secondary system water chemistry.

Steady state and transient operation of the reactor coolant system and steam generators was monitored at various power levels during the escalation to 100% FP. The response of reactor coolant inlet, outlet and average temperature; steam generator pressure, temperature and level; and feedwater flow and temperature versus reactor power was determined. Reactor coolant pump vibration and reactor coolant system leakage were maintained within the specified operational and Technical Specification limits.

Radioactive wastes generated during power operation were adequately processed, stored and/or disposed using plant and off-site facilities. Primary and secondary water chemistry have been maintained within limits allowable for operation at power. Radiochemistry analysis of reactor coolant activity indicate that no fission product releases occurred during the startup test program.

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6.1 REACTOR COOLANT SYSTEM PERFORMANCE

6.1.1 PURPOSE

A number of tests were performed prior to and during the escalation of the unit to rated power to monitor the performance of the Reactor Coolant System (RCS) and to verify the capability of the individual components to support operation at power. Various RCS parameters were monitored to identify and eliminate any oscillatory or unstable characteristics during the escalation to power. This section presents the results of those tests and an evaluation of:

(a) Steady State Operation of the reactor and steam generators at several reactor power levels.

(b) The Reactor Coolant System response to major unit transients.

(c) The performance of the reactor coolant pumps.

(d) Reactor Coolant System leakage.

6.1.2 TEST METHOD

During escalation of the unit to rated power, selected reactor coolant, steam generator and reactor coolant pump parameters were recorded by the plant computer, reactimeter and Brush recorders at the various power levels specified in the power escalation test sequence. The data collected for steady state testing was recorded over a thirty minute interval, after steady state steaming conditions were established at each plateau. Data was collected at a high recording frequency during transient tests with several minutes of steady state data before and after the transients. The recorded values were plotted and/or tabulated, where applicable, and then compared to predicted results.

6.1.3 TEST RESULTS

6.1.3.1 Steady State Operation

Reactor Coolant System and steam generator parameters were monitored after establishing steady state conditions at 0%, 15%, 25%, 35%, 40%, 50%, 65%, 76%, 85%, 95% and 100% of full power with four reactor coolant pumps operating. The average values for reactor coolant inlet and outlet temperature; steam generator pressure, temperature and level; and feedwater flow and temperature were computed, plotted and compared to the predicted response of these parameters versus reactor power. Results are presented in Figures 6.1-1 through 6.1-7. The plotted data at each plateau was extrapolated to the next power plateau to determine whether any operating limit would be exceeded during the escalation.

Figure 6.1-1 shows the reactor coolant outlet, inlet and average temperatures versus power. During the escalation to 15% FP, the reactor coolant average temperature did not follow the predicted response and was not maintained at 579°F at 15% FP. The steam generator level control setpoints were lowered from 30 in. on the startup range indication to 28 in. to obtain a 579°F average RC temperature at 15% FP.

As can be seen from Figures 6.1-2 through 6.1-7, the response of the steam generator major parameters versus NSS power was as expected with the exception of the OTSG A and B outlet steam pressures. Although the OTSG outlet pressures increased as expected with increasing power, the magnitude of the pressure reported by the plant computer was less than expected. This is attributed to a ±2% instrument string error band, which is greater than the allowable parameter deviation specification of ±1%. A more precise string calibration would yield values for steam pressure that lie within the allowable pressure band. This was verified by data recorded on the reactimeter, which did fall within the allowable deviations. The plot of OTSG A startup level on Figure 6.1-5 shows a step increase in level of about 52 in. of water at 85% FP. There was no corresponding increase in operating level at this time. The change in startup level was attributed to a shift in the transmitter calibration. Table 6.1-1 lists the average values of the major steam generator parameters recorded during escalation of the unit to 100% FP.

Steam generator upper and lower downcomer temperatures were recorded at each power level listed above and are given in Table 6.1-2. These results show that the OTSG downcomers were always in an unflooded condition above 5% reactor power, since the upper downcomer temperatures were always within 10°F of the lower downcomer temperatures.

Minor oscillations were observed in the Reactor Coolant System and steam generator parameters during the escalation to full power. Table 6.1-3 lists the period and amplitude of the oscillations observed in reactor coolant average temperature and turbine header pressure for various power levels. The amplitude of the oscillation in these parameters reached a maximum of ±0.81°F and ±14.0 psi, respectively, at approximately 70% FP. The period of the oscillation is approximately 4 seconds. At 100% power, the 4 second oscillation completely disappeared and only a low frequency oscillation in turbine header pressure was observed, with an amplitude of ±3 psi. These oscillations stem from two sources. The 4 second oscillation was due to an oscillation in the steam generator boiling regions at partial flow-rates. The oscillation is not present under full load conditions. The cause of the low amplitude oscillation at 100% FP is the slight drift of the average reactor coolant temperature by about ±0.3°F and the corresponding insertion or withdrawal of the control rods. The drift in reactor coolant average temperature is due to instrumentation drift within its deadband range. Overall, no major oscillatory problems were encountered during the startup and the reactor coolant system flows, temperatures and pressures matched their design values.

6.1.3.2 Reactor Coolant System Transients

The Reactor Coolant System was subjected to a number of varied transients during the startup program at TMI Unit I. Some of these transient operations were anticipated in the individual testing phases of the startup program but others resulted due to unpredicted plant startup problems. The nature of such transients and the corresponding behavior of the Reactor Coolant System parameters is important in regard to the number of transient cycles allowable for each component over the 40 year lifetime of the plant. Two different Reactor Coolant System transients have been selected for presentation in this section. Each represents a different transient operation for the Reactor Coolant System.

The first example of system transient behavior is the variation in Reactor Coolant System parameters due to a turbine-generator load rejection transient. The reactor was operating at 75% FP when the turbine-generator was inadvertently tripped during routine testing. In this type of transient, the turbine main stop valves shut which decreased the heat transferred from the reactor coolant to the secondary side of each steam generator. A rapid buildup in the steam header pressure opened the main steam safety relief valves to transfer some heat across the steam generators. The reduction in heat transfer out of the reactor coolant increased the reactor coolant temperature and the resultant expansion of the volume of the reactor coolant caused an increase in RCS pressure and pressurizer level. The reactor power was reduced to 15% FP by the integrated control system, thus reducing reactor coolant temperature and pressure, which were subsequently controlled at their respective setpoints. Table 6.1-4 presents the maximum and minimum values measured for reactor coolant pressure, temperature and pressurizer level during this transient.

TABLE 6.1-4

MEASURED VARIATION IN R.C. SYSTEM
PARAMETERS FOR AT TURBINE-GENERATOR TRIP

Parameter	Maximum Value	Minimum Value
RCS Pressure	2320 psig	1970 psig
RCS Average Temperature	590°F	570°F
Pressurizer Level	296 in. water	200 in. water

A second example of system transient behavior is a reactor trip from 100% FP. Tripping the reactor will immediately trip the turbine-generator and cause the turbine stop valves to close and the bypass valves and main steam safety relief valves to open. As in the first example, heat is transferred from the reactor coolant through the steam generators and the reactor hot leg and cold leg temperatures decrease uniformly. The reduction in reactor coolant temperature and the corresponding shrinkage in coolant volume will cause a decrease in reactor coolant pressure and pressurizer level. The primary difference between this type of transient and the first is the smaller increase in reactor coolant pressure as a result of tripping the reactor. Table 6.1-5 summarizes the maximum variation in reactor coolant parameters as a result of this transient.

TABLE 6.1-5

MEASURED VARIATION IN R.C. SYSTEM
PARAMETERS FOR A REACTOR TRIP FROM 100% FP

Parameter	Maximum Value	Minimum Value
RCS Average Temperature	579°F	545°F
RCS Pressure	2210 psig	1799 psig
Pressurizer Level	251 in. water	60 in. water

Based upon the low pressurizer level reached during the unit shutdown test conducted at 15% FP, the normal operating level of the pressurizer was raised from 220 in. to 240 in. The RCS temperature decrease and the corresponding shrinkage in RCS volume during a reactor trip result in pressurizer levels below the 80 in. low-low level setpoint. The pressurizer heaters shut off below 80 in., and thus limit operator control of RC pressure. Although the higher operating level has increased

the low level reached during RCS transients, the reactor trip from 100% FP resulted in a pressurizer level of 60 in. Studies are now in progress to improve RCS response in this area.

6.1.3.3 Reactor Coolant Pump Performance

The performance of the Reactor Coolant Pumps of TMI Unit I has been satisfactory. These Westinghouse controlled seal leakage pumps have produced flows of 108.6% of the 88,000 gpm design rate per pump. With the exception of minor oil leaks discovered in the pump oil coolers and upper bearing inspection plates during precritical testing and an improperly installed seal on the B RCP during hot functional testing, the overall mechanical performance of the pumps has been good. The measured shaft vibrations have been relatively low at 5 to 12 thousandth of an inch (mils). The measured bearing temperatures, component cooling and seal leakoff outlet temperatures have been within acceptable operating limits.

6.1.3.4 Reactor Coolant System Leakage

Reactor Coolant System leakage was monitored during the startup test program as part of a periodic surveillance procedure. The method used to measure reactor coolant leakage was discussed in section 3.5. The results of measurements made while the reactor was critical are summarized below. The values listed are the maximum unidentified reactor coolant leakage measured during the specified time period. The data shows that the maximum unidentified leakage limit specified in the Technical Specifications of 1 gpm was not exceeded.

SUMMARY OF RCS LEAKAGE DURING TEST PROGRAM

Period	Maximum Unidentified Leakage (gpm)
6/8 - 6/14	+0.87
6/15 - 6/21	+0.61
6/22 - 6/28	+0.68
6/29 - 7/5	+0.11
7/6 - 7/12	+0.89
7/13 - 7/19	+0.43
7/20 - 7/26	+0.58
7/27 - 8/2	+0.74
8/3 - 8/9	+0.89
8/10 - 8/13	+0.43
8/22 - 8/31	+0.74

6.1.4 CONCLUSIONS

Steady state operation of the reactor coolant system and the steam generators was monitored at various power levels during the escalation to 100% FP. The average values for reactor coolant inlet, outlet and average temperature; steam generator pressure, temperature and level; and feedwater flow and temperature followed the expected response with power. The response of the reactor coolant system to major unit transients has been satisfactory. One area that is under study is the low pressurizer level reached during a reactor trip. The reactor coolant pumps have performed well and produce flows in excess of their design values. Reactor coolant system leakage was maintained within the limits specified in the Technical Specifications.

TABLE 6.1-1

REACTOR COOLANT SYSTEM PARAMETERS - AVERAGE VALUE VS. POWER

NSS NOM PWR %	GEN MW MW(e)	OTSG A PRESS psig	OTSG B PRESS psig	TURB HDR PRESS psig	OTSG A STM TEMP °F	OTSG B STM TEMP °F	FDW TEMP °F	OTSG A SU LEVEL inches	OTSG B SU LEVEL inches	OTSG A OP LEVEL %	OTSG B OP LEVEL %	FDW FLOW $x10^{6}$ lb/hr	TAVE °F	RX PWR %
0	0	859	858	847	531	531	---	---	---	98.6	---	0.65	531	0.97
15	102	872	881	881	581	582	292	26	35	7	3	1.5	578	17.0
25	164	886	882	882	584	585	285	38	36	8	7	2.3	579	26.4
35	240	884	888	879	586	587	287	51	48	13	11	3.3	579	35.9
40	278	869	880	881	588	588	376	55	53	15	13	4.1	579	39.4
50	422	875	883	884	589	589	396	78	65	19	16	5.0	579	48.1
65	557	871	879	875	591	592	420	103	91	28	27	6.7	579	63.6
76	648	877	885	884	592	593	394	123	107	35	34	7.5	578	74.3
85	731	880	894	886	592	592	447	193	130	45	45	8.9	578	83.1
95	808	895	909	887	591	591	458	213	148	55	55	10.2	578	96.4
100	837	889	902	890	590	592	461	220	152	57	57	10.5	579	98.5
100	829	887	898	875	592	593	428	220	152	55	56	9.9	579	98.1

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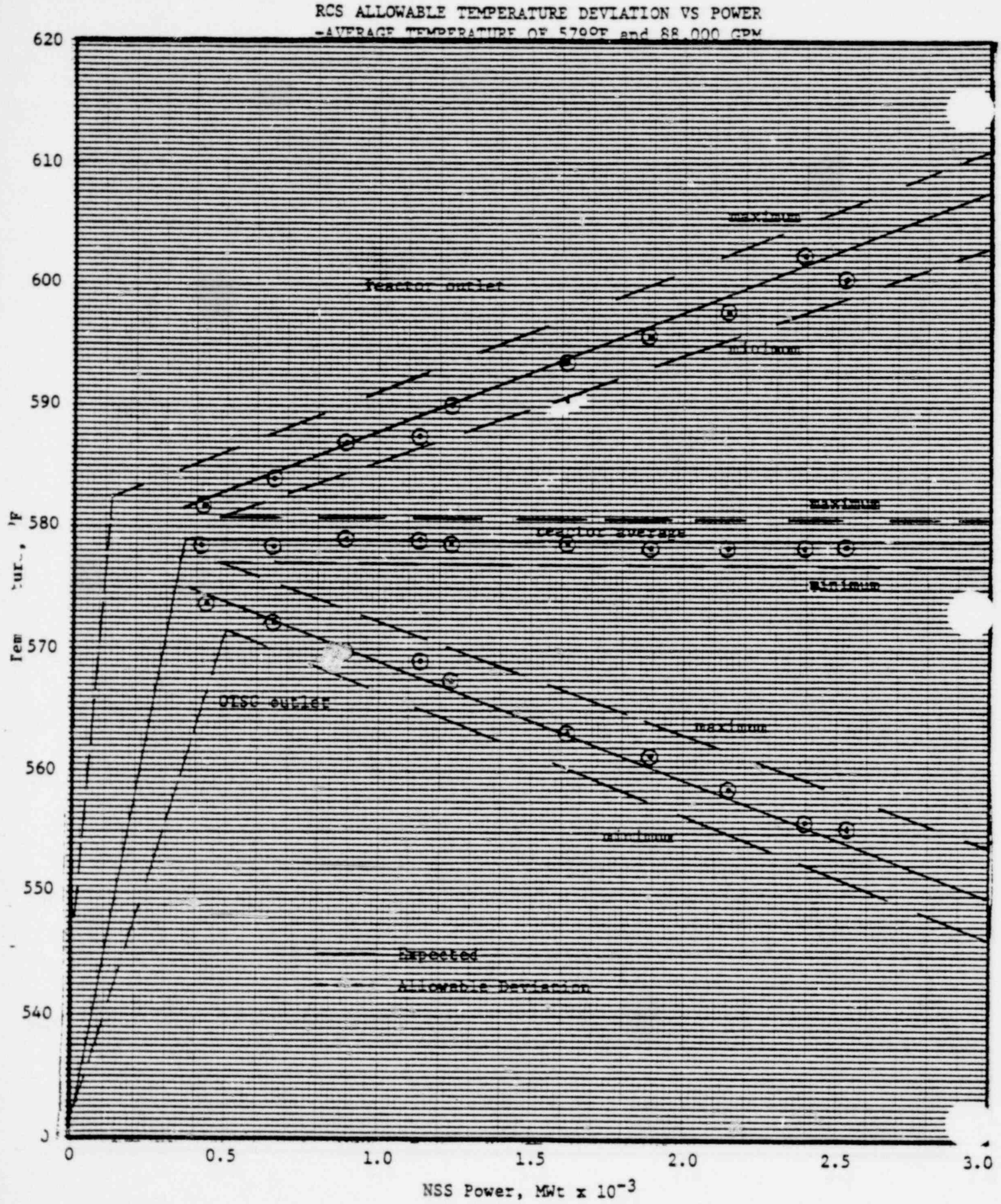

FIGURE 6.1-1

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TABLE 6.1-2

DOWNCOMER TEMPERATURES ^{o}F

NSS Nominal Power (%)	OTSG A Lower	OTSG A Upper	Difference (Lower-Upper)	OTSG B Lower	OTSG B Upper	Difference (Lower-Upper)
0	527.6	527.5	0.1	524.2	-----	---
15	534.2	532.0	2.2	532.9	529.7	3.2
25	534.1	532.0	2.1	533.0	529.6	3.4
35	534.1	532.0	2.1	533.0	529.7	3.3
40	534.2	532.1	2.1	533.1	529.8	3.3
50	535.2	532.5	2.7	533.7	530.3	3.4
65	534.4	532.3	2.1	533.5	530.0	3.5
76	535.7	533.6	2.1	534.8	531.4	3.4
85	537.3	534.9	2.4	536.3	532.6	3.7
95	537.8	535.3	2.5	536.9	533.3	3.6
100	537.4	534.8	2.6	536.5	532.7	3.8

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PLANT STEADY STATE OSCILLATIONS

Power Level (% FP)	Oscillation in T AVE Period (seconds)	Oscillation in T AVE Amplitude (°F)	Oscillation in THP Period (seconds)	Oscillation in THP Amplitude (psi)
44.0	4.0	±0.41	4.0	±5.5
50.0	4.0	±0.44	4.0	±7.9
58.6	4.2	±0.38	4.1	±8.6
62.5	4.3	±0.56	4.3	±14.0
71.3	4.4	±0.81	4.4	±10.5
76.0	4.6	±0.50	4.6	±8.0
97.5	none	±0.32	none	±3.0

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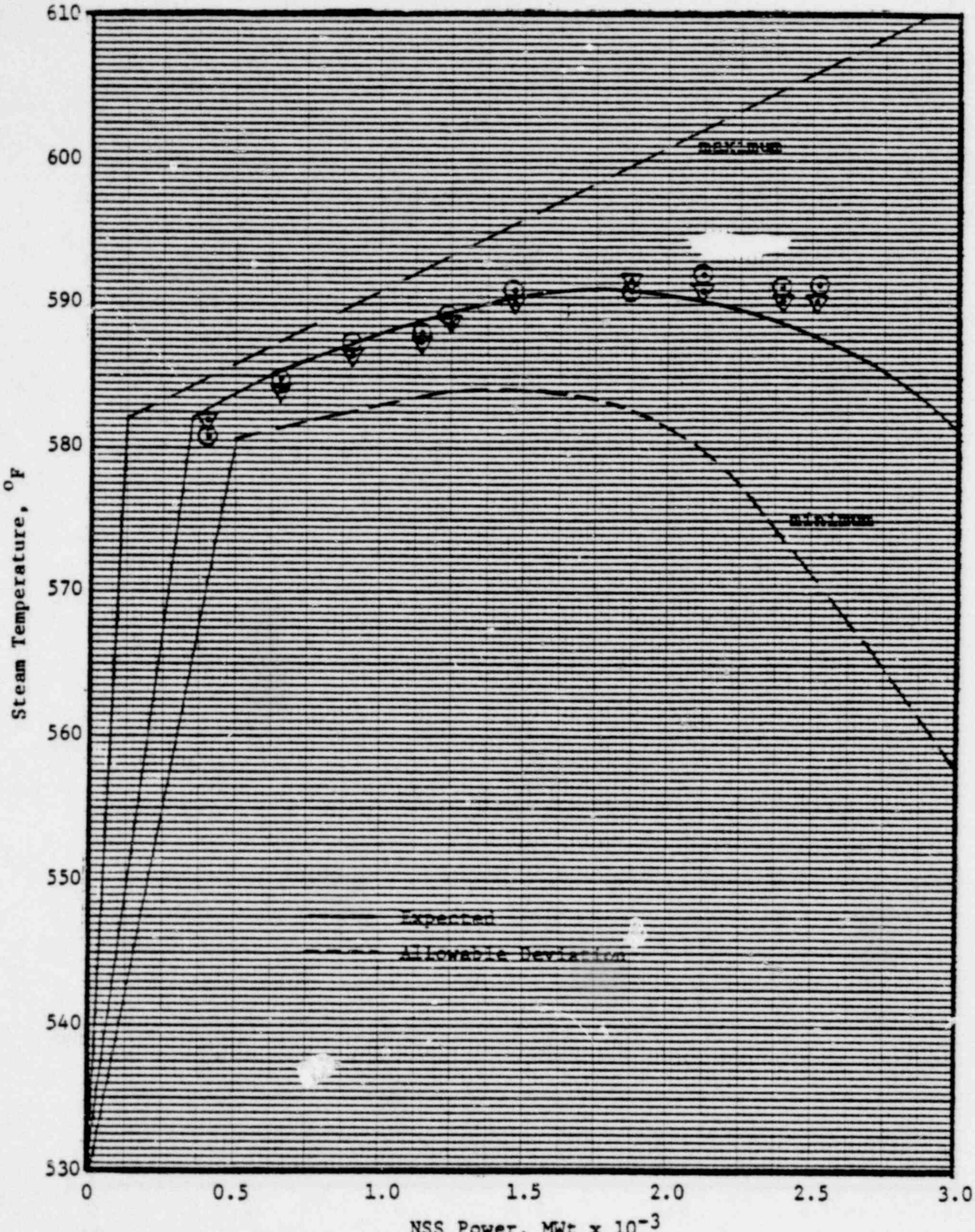
TABLE 6.1-3

STEAM TEMPERATURE VS POWER



▽ OTSG A

⊙ OTSG B

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FIGURE 6.1-2

STEAM GENERATOR OUTLET PRESSURE
DEVIATION VS POWER

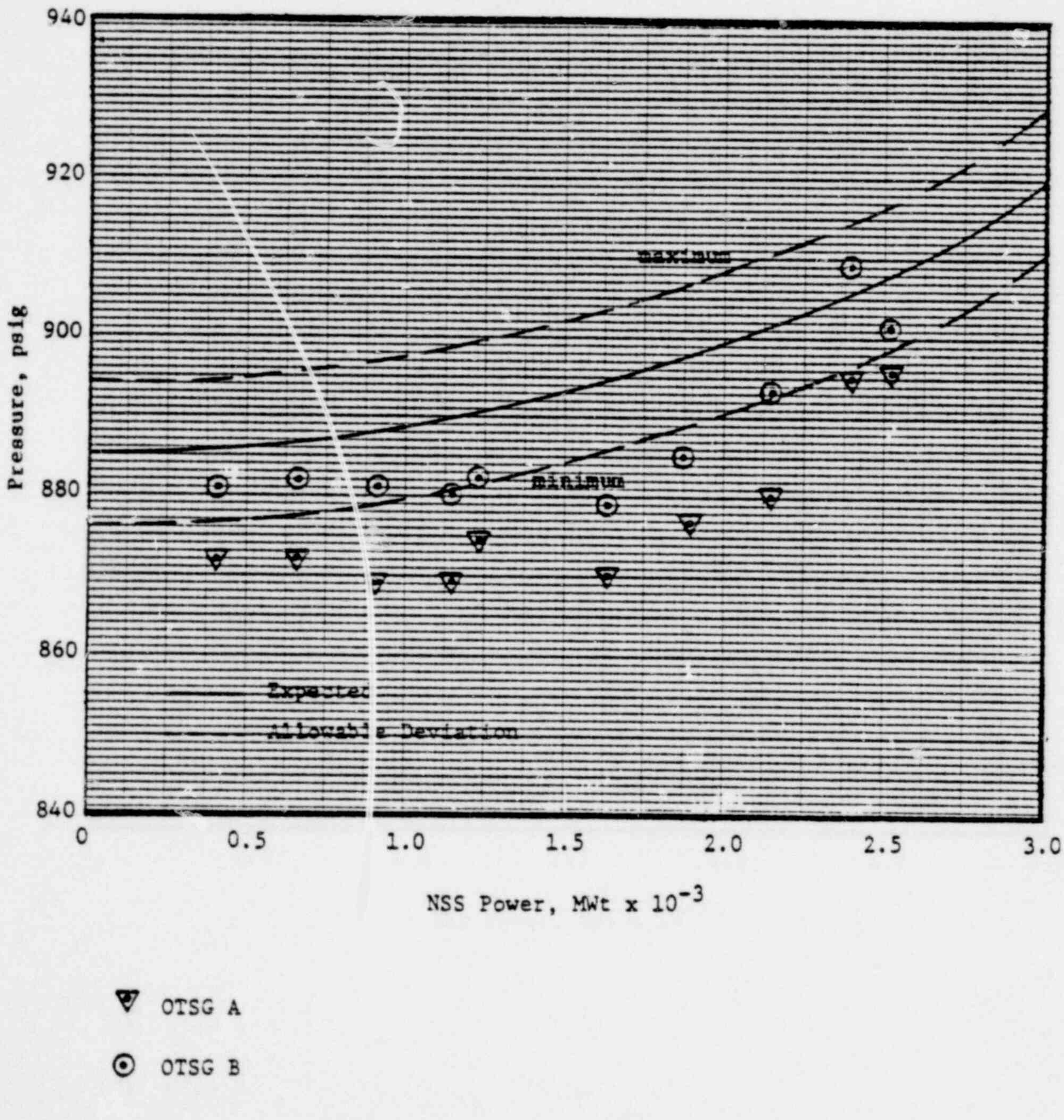

FIGURE 6.1-3

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OTSG OPERATE RANGE LEVEL VS POWER

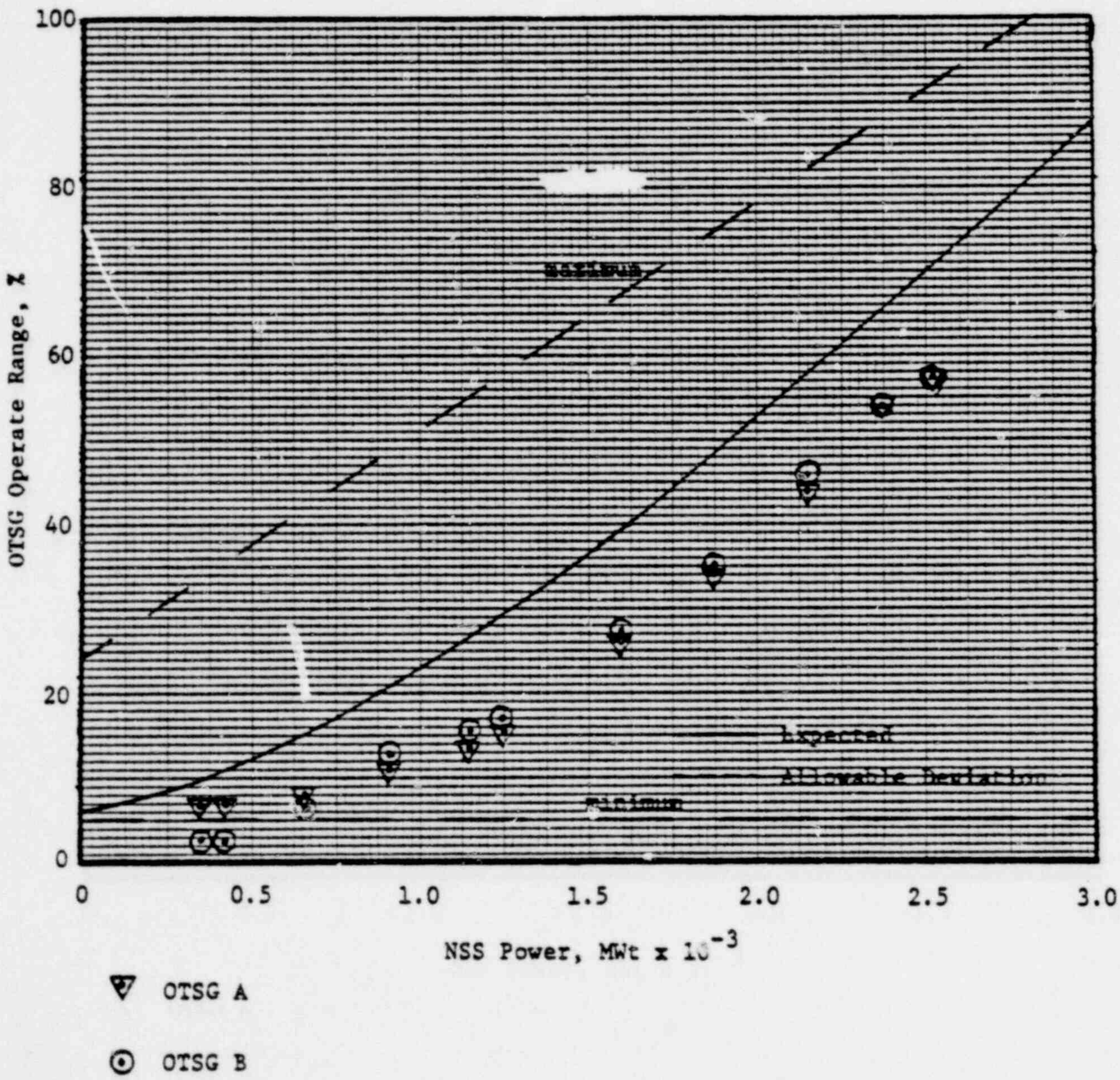

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FIGURE 6.1-4

OTSG STARTUP RANGE LEVEL ΔP VS POWER

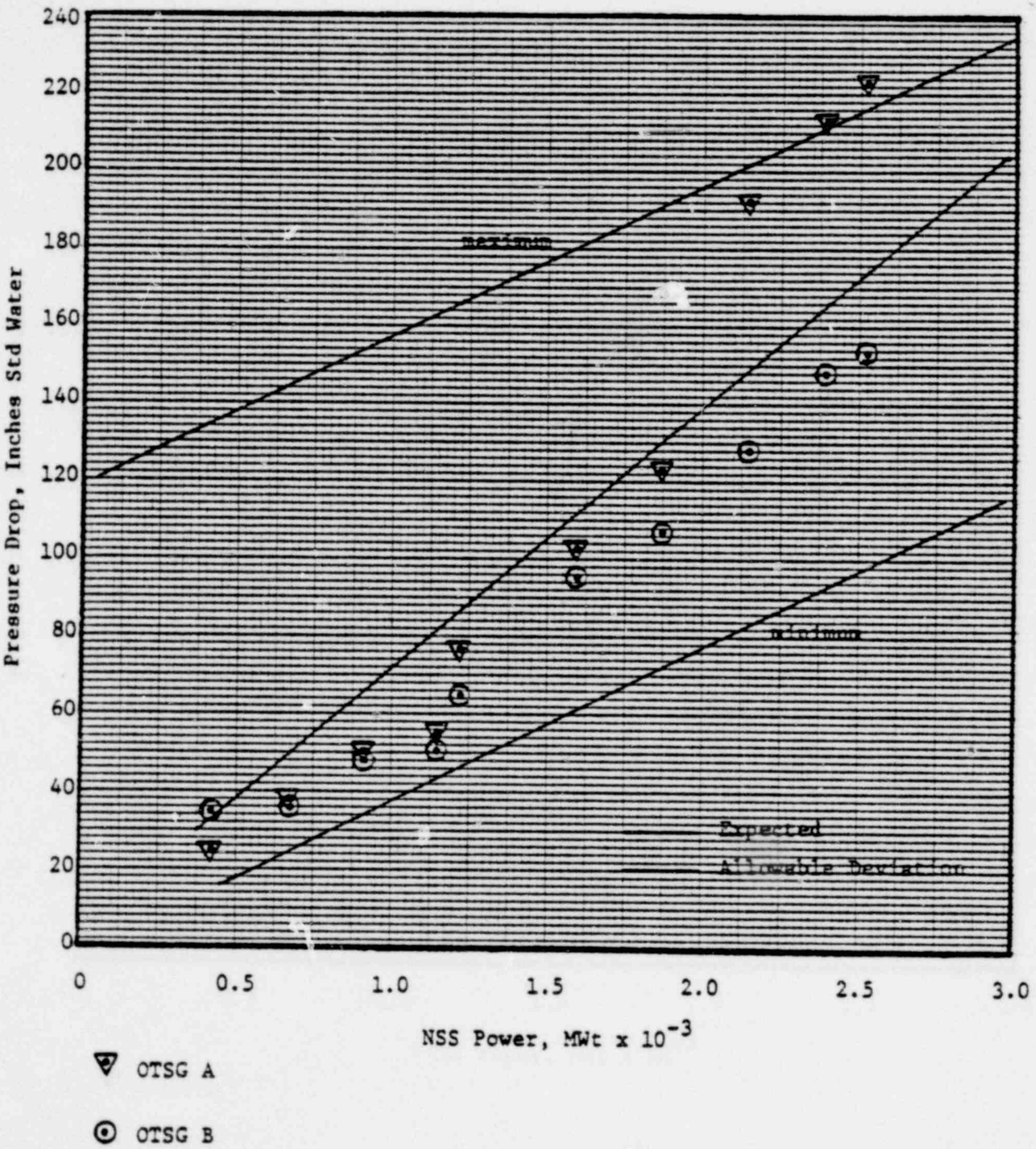

FIGURE 6.1-5

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TOTAL FEEDWATER FLOW VS POWER

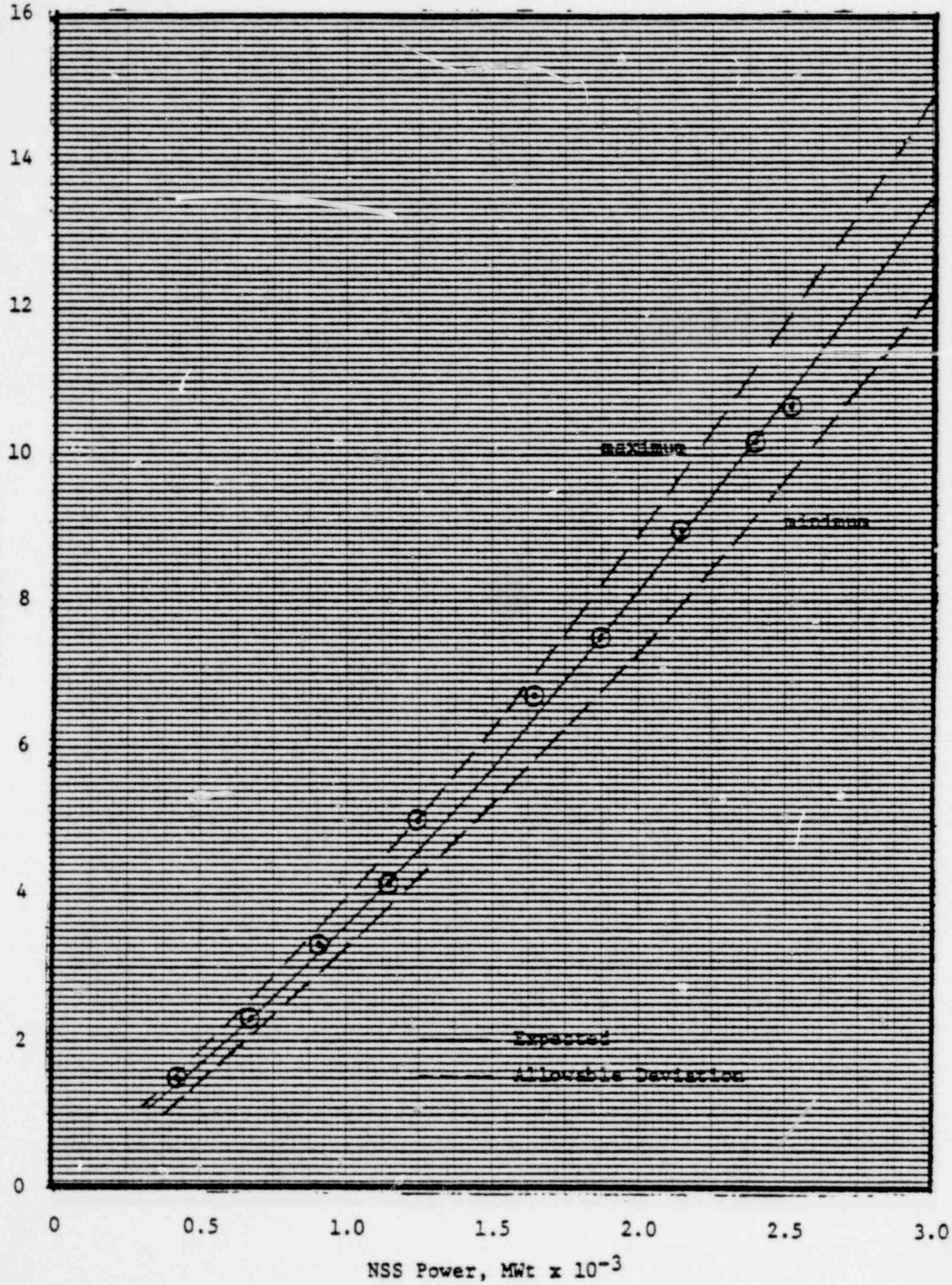

FIGURE 6.1-6

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FEEDWATER TEMPERATURE VS POWER LEVEL

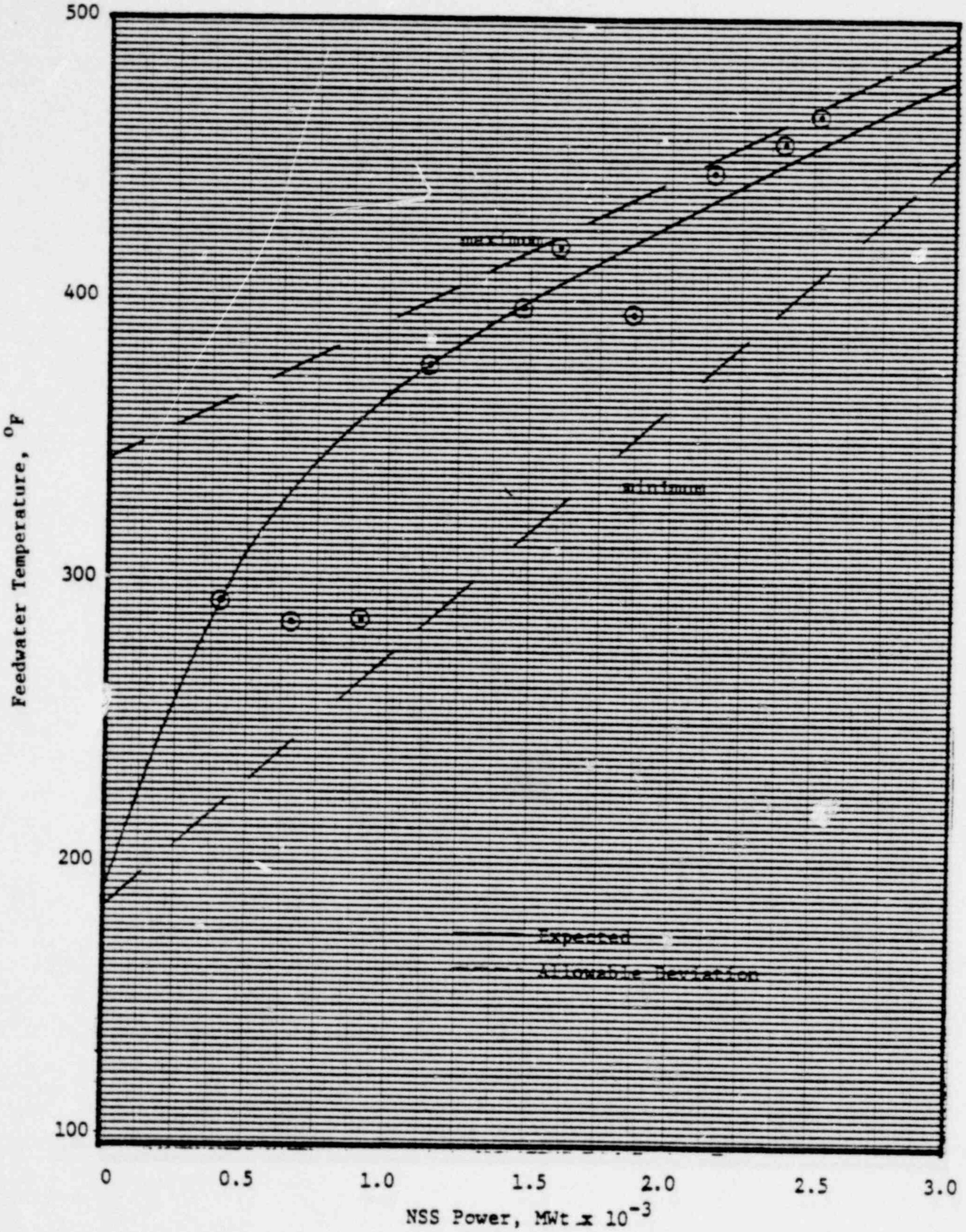

FIGURE 6.1-7

6.2 AUXILIARY SYSTEM PERFORMANCE

6.2.1 RADIOACTIVE WASTE MANAGEMENT

Radioactive wastes generated during power operation consist of liquid, gaseous, and solid wastes. Liquid radioactive wastes are basically generated from RCS letdown and RCS and auxiliary system leakage. Those wastes that are of reactor coolant grade chemistry are stored in the RC Bleed tanks (3 tanks at 88,000 gallons capacity, each) RC Bleed tank wastes can be filtered and/or demineralized, as required, to make minor adjustments in chemistry or to remove radioactive fission products and then evaporated at rates of up to 12½ gpm in the RC Bleed evaporator. The distillate from the evaporation process is demineralized and stored in one of two 7,000 gallon capacity evaporator condensate tanks. This distillate is then either recycled back into the plant or dumped into the mechanical draft cooling tower effluent to the river at flow rates of up to 30 gpm. The evaporator concentrate (high in boric acid concentration) is stored in either of the reclaimed boric acid storage tanks for reuse in the plant or in either of the two concentrated waste storage tanks for drumming.

Those wates that cannot be practically cleaned up to reactor coolant grade chemistry are stored in the 25,000 gallon capacity miscellaneous waste storage tank. These wastes are then evaporated at rates of up to 12½ gpm in the miscellaneous waste evaporator. Evaporator distillate is demineralized and stored in one of the evaporator condensate tanks. Again, this distillate is either recycled back into the plant or dumped into the mechanical draft cooling tower effluent to the river at flow rates of up to 30 gpm. The evaporator concentrate is stored in the concentrated waste storage tanks for drumming.

During the period from initial criticality on June 5, 1974 through commencement of commercial operation on September 2, 1974, a total of 6.99×10^{5} liters or 185,000 gallons of evaporator distillate was released from the evaporator condensate tanks to the river. Total radioactivity associated with these releases was 20.9 curies of tritium and 5.07×10^{-3} curies of other isotopes (mostly Co^{60} and Cs^{137}).

These values compare with the <u>maximum allowable limit</u> of 10 curies/quarter for isotopes other than tritium and noble gases and a <u>target limit</u> of 1.25 curies/quarter for isotopes other than tritium and noble gases.

Gaseous radioactive releases are basically associated with one of three sources:

1) Purging of the reactor building
2) Normal ventillation of the auxiliary and fuel handling buildings
3) Release of a waste gas decay tank

During the period from initial criticality through commencement of commercial operation, the gaseous activity released from all three sources was $.048 \times 10^{-7}$ μCi/sec of I^{131} and particulate with half lives >8 days and $3.3 m^3$/sec of gross gaseous activity. These values compare with the quarterly average limits of .05μCi/sec for I^{131} and particulate with half lives >8 days and $1.9 \times 10^{4} m^3$/sec for gross gaseous activity; and the target average limits of .024μCi/sec for I^{131} and particulate with half lives >8 days and $4.8 \times 10^{3} m^3$/sec for gross gaseous activity.

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Solid radioactive waste consists of:

1) Solidified evaporator concentrates
2) Spent filter sulka floc and demineralizer resins
3) Miscellaneous solid wastes, such as spent filter cartridges

During the power escalation test program, a total of 145 55-gallon drums of solidified evaporator bottoms were shipped off-site in two separate shipments. Total radioactivity associated with these 145 drums was 2.286×10^{-3} curies. No other solid radioactive waste was disposed of during the power escalation test program.

A major portion of the radioactive liquid waste generated during the power escalation test program was the result of RCS leakage into the Reactor Coolant Drain Tank, (still considered a portion of the RCS for leakage calculations covered in section 3.5). Starting from initial criticality on June 5, 1974, this leakage was about 2.4 gpm. On June 19, the leakage had increased to 3.7 gpm and caused RCDT temperature to increase to ≈190°F. In order to keep RCDT temperature from exceeding 190°F, cool demineralized water was added to the drain tank in large volumes and the tank was pumped out to the miscellaneous waste storage tank or an RC bleed tank (depending upon which tank had enough reserve capacity to receive the contents of the RCDT). The high leakage was traced to the pressurizer spray valve stem leakoff and was repaired on June 21. Leakage after the repair continued at the original value of 2.4 gpm. On July 5, leakage rose to 5.5 gpm and a significant number of RCDT "mixings and dumpings" were required to keep RCDT less than 190°F. The leak was located and isolated within 24 hours. Again, leakage returned to 2.4 gpm. As of September 2, the leakage had risen to ≈4 gpm and constituted the major portion of liquid and solid (solidified evaporator bottoms) radioactive waste generated.

6.2.2 PRIMARY AND SECONDARY SYSTEM WATER CHEMISTRY

Water chemistry specifications were established for the primary and secondary systems to reduce the amount of corrosion that would occur over the lifetime of the plant. Periodic samples of the primary and secondary fluids were taken during the startup test program to identify and eliminate any adverse conditions and to detect the presence of failed fuel assemblies.

Primary system water chemistry was monitored through periodic samples of the reactor coolant letdown, purification demineralizers, pressurizer, make-up tank, boric acid mix tank, core flood tanks, spent fuel pool, borated water storage tank and reclaimed boric acid mix tanks. Secondary system water chemistry was monitored through periodic samples of the steam generator feedwater and condensate trains. Radiochemistry analysis of the reactor coolant was performed during the startup test program to:

(a) Monitor activity buildup in the reactor coolant during initial fuel loading, reactor startup and initial power operation.

(b) Establish base activity levels to determine the presence of failed fuel and primary-secondary leakage.

(c) Monitor for radionuclide leakage from fuel pins to coolant, from coolant to steam generators or coolant to closed cooling water systems during startup and initial operation.

Overall, primary and secondary water chemistry specifications have been maintained during the startup test program. However, 'out-of-spec' conditions have occurred from time to time due to operational or mechanical problems. Water chemistry was returned to specification when the problems were corrected. No significant time delays were encountered during the test program due to chemistry conditions. Radiochemistry analysis of the reactor coolant system yielded results which were reasonable and consistent with expected activities and identifiable isotopes. The data indicates that no failed fuel is present and iodine activities indicate less than the predicted activity for "tramp" uranium.

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7.0 BALANCE OF PLANT TESTING

This section presents the results of balance of plant testing, adjustments and operation at power. Balance of plant systems consist mainly of the turbine generator, main steam, turbine bypass, atmospheric dump, condensate, feedwater, moisture separator, steam extraction and feedwater heating, heater drain, emergency feedwater, and cooling water systems. The cooling water systems consist of the circulating water, natural draft cooling tower, intermediate cooling water, nuclear service closed cooling water, nuclear service river water, secondary service closed cooling water, secondary service river water and mechanical draft cooling tower systems.

Turbine generator initial synchronizing, loading, monitoring, overspeed trip testing, and adjustments were performed in accordance with TP 800/9 - Turbine Generator Operational Testing. The turbine was tripped under load at 30% full power as part of TP 800/14 - Turbine/Reactor Trip Test; the generator was tripped under load at 100% full power as part of TP 800/34 - Generator Trip Test; several inadvertant turbine trips occurred during the power escalation test program. TP 800/9 will be discussed in this section whereas results of the major trips will be discussed in Section 8 of this report.

The ability of the turbine bypass and atmospheric dump systems to control steam pressure during steady state operation prior to loading the turbine, during transients, and during turbine trips was determined in TP 800/6 - Turbine Bypass System Test. A discussion of main steam safety valve operation is also included in this section.

Monitoring of initial power operation of the condensate, feedwater, moisture separator, heater drain, steam extraction and feedwater heating systems, comparison of operating flows, temperatures and pressures against design values at several power plateaus; and initial flushing of sections of the moisture separator and heater drain system were performed in accordance with TP 800/7 - Feedwater System Operation and Testing.

A discussion of the ability of the emergency feedwater system to maintain a supply of feedwater to the OTSG's during loss of the main feedwater pumps is presented in this section.

Verification of the ability of the plant cooling water systems to adequately cool their service components plus one set of performance testing (summer conditions) of the natural and mechanical draft cooling towers were conducted per TP 800/30 - Power Escalation Test Checkpoints. Cooling tower performance testing under several other seasonal atmospheric conditions still need to be performed; Met Ed will perform these tests in their respective seasons.

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7.1 TURBINE GENERATOR OPERATIONAL TESTING

7.1.1 PURPOSE

The purposes of turbine generator operational testing were to:

1) Monitor bearing vibration levels; stator, bearing, and valve chest temperatures; turbine shell, rotor and differential expansion; exhaust hood spray operation; generator field current and voltage; generator hydrogen gas temperature and pressure; and turbine cycle performance during steady state and transient testing at the various power plateaus.

2) Verify the TG operating procedure for bringing the turbine up to rated speed (1800 RPM), synchronizing it, and loading it.

3) Provide instructions for performing TG checkouts required prior to and during initial synchronization and loading; checkouts include voltage regulator adjustment, overspeed trip testing, oil trip lockout testing, and protective relaying checks.

4) Monitor EHC (electro-hydraulic control) operation during steady state and transient testing to verify acceptable control characteristics and stability.

5) Plot percent megawatts electrical VS percent megawatts thermal at 40%, 76% and 100% reactor power and compare with design values.

6) Perform the turbine generator acceptance test at 100% reactor power.

7.1.2 TEST METHOD

The turbine generator was brought up to rated speed (1800 RPM), initially synchronized, loaded and tested in accordance with the steps in the test procedure (TP 800/9) and the operating procedure (OP 1106/1) in the following sequence, as directed by the Controlling Procedure for PET (TP 800/21):

1) With 15% full power steam flow through the turbine bypass valves, warm the turbine steam chest, then accelerate it to 1800 RPM. Monitor bearing vibration levels; stator, bearing and valve chest temperatures; turbine shell, rotor, and differential expansion; and exhaust hood spray operation during this time.

2) Test oil trip and trip lockout systems.

3) Synchronize and load the turbine to 5 MWe. Remain at this load until bearing temperatures equalize and exhaust hood temperatures decrease below 125°F.

4) Pick up generator load until the bypass valves close and perform generator voltage regulator adjustments at 15% reactor power.

5) Unload the generator and perform overspeed trip tests.

Synchronize and load the turbine generator until the bypass valves close.

7) Print out computer groups 18 and 19 - turbine cycle performance and generator conditions and continue to monitor the parameters listed in 1).

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8) Perform step 7) at reactor power levels of 40, 76 and 100%.

9) Plot percent megawatts electric (corrected for differences in condenser vacuum between actual and design) VS percent megawatts thermal at reactor power levels of 40, 76 and 100%.

10) Perform inservice protective relaying phase angle, current, and voltage checks at 40% reactor power.

11) Perform turbine generator acceptance test at 100% reactor power by calculating the TG gross heat rate.

12) Monitor the parameters of 8) during the design rate ramp unit load changes performed per the Unit Load Transient Test Procedure - TP 800/23.

7.1.3 TEST RESULTS

During initial acceleration, synchronization and loading of the turbine generator at 15% reactor power, all TG parameters monitored were within limits of acceptance criteria. Several bearings exhibited high vibration (>3 mils), but not high enough to require a shutdown for balancing. One of the generator output breakers (GB1-02) could not be closed due to mechanical interference; it was repaired later on in the PET program and functioned satisfactorily. Also, a control problem developed with the turbine bypass valves whenever the turbine stop valves were open. This problem and its resolution are discussed in detail in Section 7.2.

The oil trip lockout test was satisfactory; however, the mechanical trip finger could not be reset. The trip finger was replaced and the oil trip lockout test and reset were repeated with satisfactory results.

The overspeed trip should be set at ≤1980 RPM (110% of 1800) and the backup overspeed set at ≤2016 RPM (112% of 1800). Values determined during the overspeed trip test were 1965 and 1990 RPM, respectively.

At 40% power, all TG parameters monitored were within limits of acceptance criteria. A very high frequency, low amplitude oscillation exhibited in turbine header pressure and in control valve servo currents was traced to turbine header pressure transmitter rack vibration; it was resolved by providing additional rack supports. Generator output was 311 MWe vs a design prediction of 304.6 MWe.

At 76% power, all TG parameters monitored were well within limits of acceptance criteria except one - differential expansion approached its limit of 530 mils during a power runback from 76% to 55% power. Evaluation by General Electric resulted in a new limit of 560 mils. Several EHC system oil leaks at the main Control valves resulted in forced manual trips; however, replacement of the leaking fittings resolved the problem.

A reactor/turbine trip form 76% power combined with tube leakage in the 4B high pressure heater and failure of the motor operated heater extraction steam isolation valve to close, resulted in water ingestion into the turbine; however, subsequent turbine operation appears normal and indicates minimal, if any, damage was done. Investigation indicated that the tube leak (≈1200 gpm) had existed for some time and the failure was not caused by normal plant operation nor forces

resulting from the trip transient. The heater float level control which should have closed the motor operated isolation valve was repaired and installation of a faster responding air operator in place of the motor operator on a number of heaters is under consideration.

Following heater tube repair, measured generator output was 668 MWe comapred with a design value of 661.5 MWe.

A 30 HZ control valve oscillation which produced a 12 psi peak to peak turbine header pressure oscillation was traced to the turbine speed sensor gear and noise pickup of associated cabling. The gear and cables were replaced at 100% power and the oscillation disappeared.

At 100% power, all TG parameters were within limits of acceptance criteria except vibration on #3 bearing and generated megawatts. Bearing #3 had a vibration level of 3.5 mils, which is .5 mils greater than the acceptance criteria for long term operation. Short to intermediate term operation (up to a year or more) at this level is acceptable, however, with balancing to be performed at the first convenient opportunity.

Generated megawatts at 100% power were 854 MWe at poorer vacuum conditions than design. When corrected to design vacuum by a very conservative calculation, generated megawatts were 861 MWe. This is lower than the 870.4 on the reactor power vs generated megawatts design curve; however, GE only guarantees 837 MWe. Commercial plant operation has shown the vacuum correction calculations to be conservative because generation levels of 864 MWe have been obtained with vacuum poorer than design values. The gross heat rate at 100% reactor power was 9993 Btu/KWhr compared with the acceptance criteria of ≤10,002 Btu/KWhr. Auxiliary plant load averaged 49 MWe. Currently, the clean steam generators produce steam with 33 degrees more superheat than minimum design; therefore, less steam is required to deliver a given amount of energy to the turbine and the turbine runs at something less than "valves wide open" conditions. Steam conditions currently are 10,621,000#/hr at 592°F compared with turbine VWO design of 11,158,286 #/hr at 559°F.

7.1.4 CONCLUSIONS

Turbine generator performance was very satisfactory throughout the startup test program. Approximately 7 days of testing time were lost due to unscheduled turbine trips and turbine related problems. Water ingestion into the turbine through the 4B heater extraction line due to the isolation valve failure to close on high shell side level caused by ruptured tubes was the only unanticipated startup problem which could have led to major damage and delays; however, subsequent turbine operation indicates that the turbine suffered no damage. #3 bearing vibration is approximately 0.5 mils higher than acceptable for long term operation; balancing operations will be performed at the first convenient outage. TG output at 2535 MWt is 861 MWe, when conservatively corrected to design vacuum conditions, which compares well with a guaranteed value of 837 MWe. Steam conditions are 10,621,000 #/hr at 592°F compared with design of 11,158,286 #/hr at 559°F. Due to the increased amount of superheat over design, the turbine operates at less than "valves wide open" conditions. Gross heat rate is 9993 Btu/KWhr compared with design of 10,002 Btu/KWhr.

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7.2 TURBINE BYPASS SYSTEM TEST AND MAIN STEAM SAFETY VALVE OPERATION

7.2.1 PURPOSE

The purposes of the turbine bypass system testing were to:

1) Determine the capability of the bypass valves to control main steam pressure at turbine header pressure setpoint during steady state operation and following pressure perturbations at 10 to 15% full power.

2) Determine bypass valve opening times and peak main steam pressure following turbine trips from 30% and 100% full power.

3) Determine atmospheric dump valve opening times and peak main steam pressure following loss of offsite power testing at 15% full power.

In addition to the above tests delineated in TP 800/6, the main steam safety valves were monitored for lift and reset pressures during the turbine, reactor, and generator trip tests to verify initial setpoints and determine magnitude of setpoint shift as a result of repeated actuation. Results are recorded in TP 800/14 and 800/34.

7.2.2 TEST METHOD

At ≈10% power with the turbine stop valves closed, the ability of the turbine bypass valves to maintain constant turbine header pressure was determined and ICS adjustments made as required. Then a 10 psi step change in header pressure setpoint was made and the ability of the bypass valves to control header pressure at the new setpoint determined. All ICS adjustments were recorded in TP 800/8 - ICS Tuning at Power.

Bypass valve opening times and ability of the bypass valves to control peak steam pressure following a 30% turbine trip and a 100% generator/turbine trip were to be determined by wiring up the valve limit switches to a Brush Recorder and monitoring main steam pressure. Atmospheric dump valve opening times and ability of the bypass valves to control peak steam pressure following a 15% loss of offsite power were to be determined by wiring up the valve limit switches to a Brush Recorder and monitoring main steam pressure.

Main steam safety valve lift pressures and shift in lift pressures during turbine, reactor and generator trips were determined by analyzing pressure recordings, monitoring thermocouples attached to each valve, and visual inspection.

7.2.3 TEST RESULTS

Ability of the bypass valves to control turbine header pressure at setpoint with the turbine stop valves closed was satisfactory at 10% power. Peak to peak amplitude of oscillations was an acceptable ± 6 psi. Response to changes in header pressure setpoint was also satisfactory after it was discovered that the A header pressure controlled the B valves and B header pressure controlled the A valves and the wiring error was corrected.

When the turbine stop valves were opened at 15% power in preparation for rolling and loading the turbine, it was observed that the two steam generators began steaming unevenly and the disparity increased with time. Analysis indicated that proximity of A and B header pressure sensors to the steam chest resulted in a common pressure sensed by both when the stop valves were opened thus joining the A and B headers. Under these circumstances, small errors between actual and indicated pressures well within the range of calibration tolerances combined with the interaction of reverse flow stop check valves in the main steam headers led to highly uneven steaming rates between the steam generators. In order to resolve this problem, the control signal to each set of bypass valves was modified to include the difference in pressures between the two steam generators. This modification serves to converge the steaming rates. Since TP 800/6 did not cover pressure control by the bypass valves at 15% power with the stop valves open, a special operating procedure was written to cover it (SOP #46).

Peak main steam pressure reached following the turbine trip from 30% power was 1025 psia; following the generator/turbine trip from 100% power, it was 1082 psia. This compares with a design peak of 1100 psia and a code allowable peak of 1170 psia. Valve opening time was not recorded during the transient due to limit switch misalignment caused by vibration, however, static times indicated a maximum opening time of 2.1 seconds compared with the acceptance criteria of 3 seconds. During the test program, it was discovered that design called for only four of the six bypass valves to be in operation at any given time; therefore, two valves were isolated and the 100% turbine/generator trip test was done with only four bypass valves in service.

Peak main steam pressure reached following the loss of offsite power was 1008 psig in the A header and 1032 psig in the B; this was far less than the design peak of 1100 psia and the code allowable peak of 1170 psia. In fact, pressure reached such a low peak and decayed so rapidly that the atmospheric dump valves did not need to open to control pressure; therefore, opening times were not measured.

The main steam safety valves were originally set during factory testing. Their setpoints were checked and adjusted in place using a hydro set assist during hot functional testing. Subsequent hot functional testing indicated several valves were lifting at lower than setpoint so the valves were set a second time. During the 30% turbine trip test, one safety valve lifted ≈25 psi below setpoint and reseated at ≈25 psi below reseat pressure. The valves were reset for a third time; during resetting, it was determined that an incorrect curve provided by the valve manufacturer for conversion of hydro set pressure to steam pressure had been used on the previous field sets. Traces of the 100% generator/turbine trip indicate that all safety valves lifted and reseated at acceptable limits; however, safety valve operation will be the subject of a continuing study on plant transient response optimization.

7.2.4 CONCLUSIONS

Acceptable response of the turbine bypass valves in maintaining turbine header pressure setpoint and response to small changes in setpoint at reactor powers $\leq 15\%$ was attained after the difference between steam generator pressures was included in the control system and a wiring reversal error was corrected. Peak to peak oscillations are ± 6 psi.

The turbine bypass valves, along with the main steam safety valves, function adequately to limit main steam pressure during turbine trips to <1100 psia. The longest valve opening time was 2.1 seconds; peak steam pressure following the 100% generator/turbine trip was 1082 psia.

Operation of the atmospheric dump valves was not even required to limit main steam pressure to <1100 psia during the loss of offsite power test.

Final settings of the main steam safety valves appear adequate for continued plant operation; however, safety valve operation is one of several areas under study in an attempt to optimize plant response to major transients.

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7.3 FEEDWATER SYSTEM OPERATION AND TESTING

7.3.1 PURPOSE

The purposes of feedwater system testing were to:

1) Provide an organized approach to placing secondary cycle components into operation as a function of power level.

2) Verify that feedwater system and associated equipment perform without excessive oscillations during transients and steady state operations at 15, 40, 76 and 100% power.

3) Verify that feedwater system flow, temperature and cycle performance meet design specifications at 40, 80 and 100% steam flow.

7.3.2 TEST METHOD

The secondary cycle systems and components (condensate, feedwater, moisture separator, heater drain, feedwater heating and steam extraction) were placed in operation as follows:

1) Prior to power operation, the level controllers on the heater drain tank, moisture separator drain tanks and feedwater heaters were set to provide anticipated stable control. Also, correct float cage centerline location with respect to tank or heater bottom was verified. Initial values of proportional band and reset were recorded for each controller.

2) As power level was increased (around the 40% power range), the moisture separator drain tanks, 6th stage heater drain tank, and feedwater heaters were flushed until their chemistry conditions were acceptable to permit normal valve lineup operation.

3) At major power plateaus of 40, 76 and 100%, each controller was tuned for stability at steady state, if required, and then optimized by disturbing the controller flapper and monitoring the controller's ability to dampen the resultant perturbation. If response was unacceptable, the proportional band and/or reset were tuned and the new values recorded along with the power level. The final setpoints of each controller were recorded for each of the above power levels.

4) At 40, 80 and 100% full steam flow, feedwater cycle performance was compared with design. This included a comparison of feedwater flow, final feedwater temperature, terminal temperature difference for each FW heater and approach temperature for each FW heater.

7.3.3 TEST RESULTS

The following problems were uncovered and resolved to the point of supporting continued acceptable plant operation:

1) Three of the six moisture separator drain tank high level dumps were blocked and three of the six MSDT pump discharge valves were either blocked or bound up. This prevented moisture separator drain tank level controller tuning until after the turbine generator screen outage shutdown and repair. Blocking of these lines may have been prevented by flushing them to a drainage ditch for

initial cleanup prior to flushing to the condenser. Subsequent MSDT operation has proven satisfactory.

2) Feedwater flow oscillations were caused by oscillations of the heater drain pump discharge valves. Initially the two valves in the heater drain pump discharge header operated in series; in this mode, the first valve was greater than 70% open at 76% power and severe oscillations existed. Oscillations were significantly reduced by changing valve cams to somewhat linearize the equal percent valve plug, installing snubbers in the air supply lines to the diaphragm operators, and overlapping the control signals to permit parallel operation. Currently, the first valve is 60% open and the second is 40% open at 100% full power (both are <70%). Even though oscillations are currently of low amplitude, a further reduction will be attempted by installing snubbers between the valve diaphragm and plug.

3) Heater drain pump recirculation valve actuation also caused feedwater oscillations. The recirc valves were actuated at too low a flow rate by the 0-4000 gpm flow switches to permit accurate operation. An improvement in response was affected by changing the actuation setpoint from 400 to 800 gpm. Recirculation valve operation and its interaction with the integrated control system is still under investigation in an attempt to further optimize ICS performance.

4) Feedwater pump minimum speed for ICS control was lowered from 3300 RPM to 2800 RPM. This resulted in a lower flow at minimum speed, which caused the differential pressure across the feedwater valves to drop to its control point of 35 psid at a lower power level. This also resulted in the main feedwater flow block valve opening at a lower power level, which resulted in a smoother transition from startup feedwater valve control to main feedwater valve control.

5) Severe transients on the control system, such as turbine trips, feedwater pump trips, and dropped control rods from high power levels revealed the fact that feedwater flow to the steam generators during these transients decreased faster than the feedwater demand signal from the ICS. This phenomonon is shown in Figure 8..-10 of the Section 8.1 discussion of plant transient tests and occurred because the high steam generator pressure caused by the transient prevented all of the demanded feedwater from entering the steam generators. This problem was resolved to the point of permitting continued plant operation by providing an error signal to the feedwater speed controller proportional to the difference between actual turbine header pressure and its setpoint. This is another area that is still under investigation in an attempt to further optimize ICS performance.

6) Another phenomenon shown in Figure 8.1-8 of Section 8.1 is the increase in feedwater flow substantially above a decreasing feedwater demand about 1-1/2 minutes after a feedwater pump trip from 100% power. This increase in flow also causes reactor power to increase for about 1-1/2 minutes; then both power and flow decrease to levels corresponding to feedwater demand. Preliminary investigation indicate that sluggish response of feedwater control valve position to valve differential pressure is responsible for this action and, even though response is acceptable to permit continued plant operation, minor control system modifications will probably be made to further optimize ICS performance.

7) In section 7.1, the effects of ruptured tubes in the 4B heater on turbine generator operation were discussed.

Review of information prior to the reactor trip at 76% power indicates that the ruptured tubes had existed for some time. For example, the "C" heater drain pump current was greater than full load current at 76% power with the normal number of drain pumps running (2) as are required for 100% power operation. Also, the condensate booster pumps were running close to their full load current rating and condensate flow was about 1200 gpm higher than expected. Heater performance data required by TP 800/7 indicated less than acceptable heater performance at 40% and at 76% power; however, this was blamed on bad computer input data rather than feedwater heater leakage and the 40% and 76% heater performance data was not recorded. Hindsight reveals that we should have paid more attention to these discrepancies. Heater performance data met acceptance criteria after the computer inputs were calibrated and condensate booster pump and heater drain pump currents were acceptable after repair of the 4B heater.

7.3.4 CONCLUSIONS

The condensate, feedwater, moisture separator, heater drain, feedwater heating, and steam extraction systems function acceptably to support steady state and transient operation at 100% power. Oscillations and transient response associated with these systems are acceptable; however, investigations are continuing in several areas in an effort to further optimize plant response. These areas are:

1) Heater drain pump discharge valve control.

2) Heater drain pump recirculation valve control.

3) Ability of the feedwater pumps to supply feedwater to the steam generators when turbine header pressure increases rapidly.

4) Ability of the feedwater control valves to respond to changes in valve differential pressure.

Smoother initial operation of the moisture separator drain system may have been possible if we had flushed the high level dumps and pump discharge lines overboard prior to putting them into normal operation. This may have prevented several cases of valve blockage with debris.

A review of heater drain pump and condensate booster pump current data and heater cycle performance data indicates that the 1200 gpm tube leak in the 4B heater did not occur during the reactor trip at 76% and could possibly have been detected prior to the trip. Heater cycle performance meets design acceptance criteria at 100% power.

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7.4 EMERGENCY FEEDWATER SYSTEM OPERATION AND TESTING

7.4.1 PURPOSE

The purpose of emergency feedwater system testing during the power escalation program was to verify that the steam turbine driven emergency feedwater pump supplies adequate cooling water to the steam generators during the loss of off-site power test (TP 800/32) to remove decay heat generated in the core and limit the energy buildup in the reactor coolant system.

During the hot functional test program, all automatic interlocks of the turbine driven emergency FW pump were verified; also, head-flow curves of both 1/2 capacity motor driven and the full capacity turbine driven pumps were verified.

7.4.2 TEST METHOD

During performance of the loss of offsite power test (TP 800/32), the steam turbine driven emergency feedwater pump should start automatically. It should pump water into the steam generators at a rate great enough to increase their levels up to 95% on the operating range level instrumentation and maintain them at 95%, while removing all of the decay heat generated by the reactor.

7.4.3 TEST RESULTS

The turbine driven emergency feedwater pump started feeding the steam generators automatically, as expected. Emergency feedwater flow was so great and decay heat generation so low that RCS pressure and temperature began decreasing as soon as the EFW pump started. Initial pressure was 2164 psig and temperature was 582°F. In order to avoid exceeding a 100°F/hr cooldown rate, the emergency feedwater valve were switched to manual as soon as both steam generator levels were verified as increasing (the 95% level setpoint control had been verified during hot functional testing) and the feed rate slowed down by manually throttling the valves. Uneven feeding of the steam generators resulted from operator error, but final RCS pressure was 2004 psig and final temperatures were 574°F out of the reactor, 522°F out of the A steam generator and 545°F out of the B steam generator.

As a result of this test data, B&W has proposed a natural circulation level setpoint of 50% instead of 95% in order to keep from exceeding cooldown limits. The 50% level is more than adequate for decay heat removal by the EFW system.

7.4.4 CONCLUSIONS

With the amount of decay heat present during performance of the loss of offsite power test, the turbine driven emergency feedwater pump provided more than adequate flow to control RCS temperature and pressure. In fact, the EFW valves had to be throttled to keep from exceeding RCS cooldown limits as steam generator levels began increasing to 95% on their operating range level indication. A setpoint of 50% instead of 95% will be used to adequately remove decay heat without exceeding cooldown rate limitations.

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7.5 POWER ESCALATION CHECKPOINTS

7.5.1 PURPOSE

The purposes of performing power escalation checkpoints were to verify that:

1) The secondary service closed cooling water system adequately cools its service components at power levels of 15, 40, 76 and 100%.

2) The mechanical draft cooling tower performs as designed at 15, 40, 76 and 100%.

3) The natural draft cooling tower performs as designed at 100% power under three different sets of climate and weather conditions.

4) The Powdex system performs as designed at 100% power.

7.5.2 TEST METHOD

At each major power plateau (15, 40, 76 and 100%), the common secondary service closed cooling water inlet temperature to all SSCCW heat exchangers and the outlet temperature of each SSCCW heat exchanger were recorded and valve adjustments were made, if required, to limit outlet temperatures to less than 95°F. Service component cooler flows were adjusted as required to maintain outlet temperatures within design limits.

At each major power plateau, the mechanical draft cooling tower effluent temperature and differential temperature between river water and effluent were recorded and compared with summer acceptance criteria of:

1) Differential temperature between effluent and inlet shall be no greater than +7F° or less than -3°F during steady state operation for inlet temperatures less than 87°F.

2) If inlet temperature is 87°F or greater, then effluent temperature shall be maintained at or below inlet during steady state operation.

During RCS cooldown, the mechanical draft cooling tower effluent temperature and differential temperature were recorded and the rate of change of differential temperature was monitored and all were compared with the summer acceptance criteria of:

1) Differential temperature between effluent and inlet shall be no greater than +12F° and this differential shall not decrease at a rate exceeding 2F°/hr.

Performance testing of the MDCT was to be performed at 100% power operation and during cooldown by Marley Company (the supplier) in accordance with Gilbert Associates Specification 5572. The acceptance criteria are as follows:

1) During cooldown with 3 fans running, the MDCT shall cool 33,000 gpm from 108°F to 85°F at an ambient wet bulb temperature of 78°F.

2) During cooldown with 2 fans running, the MDCT shall cool 33,000 gpm from 108°F to 87.5°F at an ambient wet bulb temperature of 78°F.

3) During 100% power operation, the MDCT shall cool 15,000 gpm from 110°F to 85°F at an ambient wet bulb temperature of 78°F.

Performance testing of the natural draft cooling towers at 100% power was performed by Marley Company at one of the three conditions required. In order to determine performance over the full range of climate and weather conditions normally experienced, performance measurements are to be made under atmospheric conditions prevailing around April, August, and December - the August type performance evaluation was conducted soon after commencement of commercial operation in September and preliminary results are included herein. Acceptance criteria are in the form of performance curves supplied by Marley.

Powdex (condensate cleanup system) effluent chemical analysis was performed at 100% power and compared with acceptance criteria.

7.5.3 TEST RESULTS

The secondary service closed cooling water system provides adequate flow to cool its component heat loads at 100% power. With a SSCCW heat exchanger common inlet temperature of 87.6°F, outlet temperatures at each of the four heat exchangers were 78°F, 74°F, 78°F and 79°F, respectively, compared with the acceptance criteria of 95°F.

The differential temperature controller on the mechanical draft cooling tower was inoperable during the power escalation program; therefore, effluent temperature had to be manually controlled. In the manual mode, both differential temperature and effluent temperature were within the limits of acceptance criteria at each power plateau and during cooldown when the readings were taken; however, changes in ambient conditions and failure, at times, of the control room operator to provide continuous monitoring resulted in exceeding these limits several times.

Performance testing of the MDCT at 100% power was not performed during the power escalation program due to the inoperability of the differential temperature controller. It will be performed at a later date.

Performance testing of the natural draft cooling towers resulted in an equivalent flow of 241,603 gpm per tower from the performance curves for August type conditions compared with the design value of 232,000 gpm. This is a capacity of 104.1% of design. April and December type performance tests will be conducted by Marley at a later date.

Powdex effluent chemical analysis demonstrated satisfactory ability to control condensate chemistry. Analysis results are compared with acceptance criteria below:

	Analysis	Acceptance Criteria
1) total dissolved solids	2ppb	$\leq$25ppb
2) total suspended solids	0	$\leq$10ppb
3) dissolved SiO_2	2ppb	$\leq$ 5ppb
4) iron	0	$\leq$ 5ppb
5) copper	0	$\leq$ 2ppb
6) pH	9.41	9.3-9.5
7) conductivity	.1 on highest vessel of 6	$\leq$.1MMHO/CM^3

7.5.4 CONCLUSIONS

The secondary service closed cooling water adequately cooled its heat loads; SSCCW heat exchanger discharge temperatures were well below their design limit of 95°F at 100% power.

The mechanical draft cooling tower effluent temperature and differential temperature between inlet and effluent had to be controlled manually because the automatic controller was inoperative. Acceptable operation could be obtained with continuous surveillance; however, until operators gained familiarity with tower operation, differential temperature limits were exceeded several times. Final performance testing will be conducted at a later date.

The natural draft cooling towers were performance tested under August type conditions. Capacity was determined as 104.1% of design. April and December performance tests will be conducted at a later date.

Powdex effluent chemistry analysis demonstrates acceptable capability to clean up the condensate system at 100% power operation.

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8.0 UNIT PERFORMANCE

The tests presented in this section were performed during and after the escalation to 100% FP and measured the overall performance of the unit under normal operating, transient and emergency conditions. A description of unit response to planned and unplanned major load changes is presented in the section on Unit Transient Response. The Loss Of Offsite Power and Shutdown From Outside the Control Room Tests demonstrated the ability to safely control the unit under emergency conditions. The Unit Acceptance Test verified that the nuclear steam system can operate in accordance with the warranted design specifications.

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8.1 UNIT TRANSIENT RESPONSE

8.1.1 PURPOSE

The capability of the Integrated Control System (ICS) to maintain control of the turbine, steam generators and reactor under non-steady state conditions was demonstrated through various types of scheduled and unscheduled transients that occurred during the startup test program. Six different types of transient operation have been selected for presentation in this section to demonstrate ICS response to major load changes. The material presented includes a discussion of the following:

(a) Transient testing performed at 40%, 76% and 100% FP that required load changes at rates equal to the design ramp rates while in the fully integrated, turbine following and reactor/steam generator following modes of control.

(b) ICS response during steady state and transient operations with three reactor coolant pumps operating at 25% FP.

(c) ICS response to tripping a main feedwater pump from 100% FP.

(d) ICS response to a dropped control rod at 76% FP.

(e) ICS response to a turbine trip from 76% FP.

(f) ICS response to a load rejection at 100% FP.

8.1.2 TEST METHOD

Tests conducted on the ICS at specified power plateaus during the escalation to full power were designed to optimize the performance of the sub-loop and feedforward controllers. The capability of the ICS to control the reactor, turbine, turbine bypass valves, feedwater pump speed, steam generator level and feed flow, and average reactor coolant temperature was monitored under steady state and transient conditions and sub-loop adjustments were made as required. Unit load changes were performed prior to conducting the ma; . transient tests to determine the actual relationship between reactor power, feedwater flow and generated megawatts and to set these functions into the ICS feedforward control.

The Unit Load Transient Test conducted at 40%, 76% and 100% full power tested ICS response in the three modes of control and at rates up to and including the design ramp rates. In each case, steady state conditions were first established and data recording commenced using Brush recorders and the reactimeter. Unit load was then reduced to a specified power level and steady state conditions were again established while turbine header pressure, reactor coolant average temperature and pressure, and other plant parameters were monitored. Unit Load was then increased to the original level. This procedure was repeated as required until the transient was performed at the design ramp rates and optimum control in each mode was achieved.

A similar method was used to perform the transient tests described in sections 8.1.1 (b) through 8.1.1 (f). Steady state conditions were established with the ICS in the fully integrated mode of control and data recording commenced. The respective component (feedwater pump, reactor coolant pump, generator, etc.) was then tripped to start the transient; data recording continued until steady state conditions were once again attained. In those cases where unplanned turbine and reactor trips occurred, data was obtained from the reactimeter and/or the Brush recorders, which were used as plant monitors during the startup program.

8.1.3 TEST RESULTS

Initial testing of the Integrated Control System in preparation for initial plant operation began prior to escalation into the power range. The first set of adjustments to the system was to set all the modules to the calculated parameters and to perform a basic wiring check on the cabinets. Simulated plant signals were then used to monitor the response of the control system to changes in the plant parameters. This provided a check of ICS response to abnormal plant condition without affecting the reactor or secondary system.

The response of the ICS to all scheduled and unscheduled plant transients and trips was analyzed. The transient results discussed in this section are summarized in Table 8.1-1. Based upon the results of the transient tests, minor adjustments were made to the ICS during the escalation to full power to provide the best possible response in all modes of operation. The following adjustments were made to the ICS as a result of the startup testing:

(a) Adjustments were made to the sub-loop controllers to replace the theoretical relationship between generated megawatts, neutron power and feedwater to the measured one.

(b) The range in which the ICS controls feedwater pump speed was changed to allow lower speeds and thus better control of feed valve differential pressure at lower flow rates.

(c) An input from turbine header pressure was added to the feed pump control circuit to increase pump speed rapidly when steam generator pressure rises due to a turbine or reactor trip.

(d) The turbine bypass valve control was changed to provide balanced steam generator pressures. This was required because of the stop-check valve feature of the TMI Unit I main steam lines.

(e) Several capacitors were added to the system as required to filter process and ac noise.

8.1.3.1 Unit Load Transient Tests During Escalation

The results of the Unit Load Transient Test performed at 40%, 76% and 100% FP are presented in Tables 8.1-2 through 8.1-5 and Figures 8.1-1 through 8.1-6. Table 8.1-2 gives a summary of the tests performed at each power level. Tables 8.1-3 and 8.1-4 show the maximum and minimum values reached for turbine header pressure, RC average temperature and RC pressure during the 40% and 76% FP test, respectively. The results of the 100% FP transients, are presented in Table 8.1-5 and plotted in the figures.

The testing at 40% FP maneuvered the unit in a 55%-45%-55% transient in the fully integrated, turbine following and reactor/steam generator following modes of control at the design ramp rate of 5% FP/min. The testing at 76% FP was conducted in the same three modes of control for a 76%-66%-76% ramp, again at 5% FP/min. As can be seen from Tables 8.1-2 and 8.1-3, the ICS was able to maneuver the plant through a 10% load change without exceeding acceptable limits on primary and secondary plant parameters. The maximum rate that could be achieved in the reactor/steam generator following mode without exceeding of 50 psi turbine header pressure error was approximately 5.0% FP/min.

After completion of the 10% load changes at 76% FP, steady state conditions were established again and one main feedwater pump was taken off line. Reactor power was then increased with only one feedpump supplying both steam generators. The results of this test showed that the flow from a single feedwater pump would support operation up to 78% FP.

The results of the Unit Load Transient Test performed at 100% FP are shown in Table 8.1-5 and Figures 8.1-1 through 8.1-6. The first part of the test maneuvered the plant in a 100%-90%-100% transient at 5% FP/min. These ramps were successfully completed in all three modes of control. Ramp rates up to 10% FP/min were used in the second part of the testing at 100% FP. The load change rate limiter on the turbine EHC control was raised from 6%/min to 10%/min as a result of the 100% FP testing.

The 100%-50%-100% ramp was completed without exceeding ±40 psi turbine header pressure error and ±2°F T AVE error in the fully integrated mode. Maneuvering in the turbine following mode was accomplished within acceptable limits at 8% FP/min. The maximum rate obtained in the reactor/steam generator following mode without exceeding a 50 psi THP error was 6% FP/min decreasing power and 4% FP/min on the increase.

8.1.3.2 Reactor Coolant Pump Trip at 25% FP

Steady state conditons were established at 26% FP with four reactor coolant pumps (RCP) operating. The ICS was in the fully integrated mode of control. The A RCP was tripped from this condition and the results are shown in Figure 8.1-7. The unbalanced reactor coolant loop flows resulted in a decrease in heat transferred to the A steam generator and an increase in core average temperature. The ICS corrected this condition by redistributing the total feedwater flow to increase the flow to the A loop. The maximum change in turbine header pressure and T AVE were less than ±20 psi and ±5°F, respectively. These results show that the transition from four pumps to three pumps operating was accomplished smoothly by the steam generator ΔTc controller.

8.1.3.3 Main Feedwater Pump Trip At 100% FP

The effects of tripping one main feedwater pump while operating at 100% FP are shown in Figure 8.1-8. Once steady state conditions were established with the ICS in the fully integrated mode of control, the A main feedwater pump was taken off line. Unit load demand was run back at 48%/min and reactor power was reduced to 66% FP at 31.5% FP/min. Both runback rates agreed well with the design rates of 50%/min and 30% FP/min, respectively. Turbine header pressure error remained less than 20 psi. The final power level reached should have been approximately 60% FP. This power level was not reached until approximately five minutes after the trip, following a 2½ minute period in which reactor power increased slightly before stablizing at 62% FP.

As can be seen from Figure 8.1-8, feedwater flow decreased rapidly and then held nearly constant until the feedwater demand signal reached the same value during the first two minutes of the transient. Feedwater flow then increased and decreased over the next 2½ minutes while the demand for feedwater remained approximately constant. The increase in feedwater flow was attributed in part to the input to the feedwater pump speed circuit which increases pump speed whenever a positive turbine header pressure error develops. The feedwater cross limit signal, which requires reactor power to match feedwater flow within ±5%, caused reactor power to follow feedwater flow. The ICS corrected the various error signals that develope.

during the transient and feedwater flow matched the demand for feed at 5 minutes after the trip. Reactor power was then decreased to the proper value.

8.1.3.4 Asymmetric Rod Runback

The asymmetric rod runback feature of the ICS was tested during the Dropped Control Rod test at 76% FP (Section 5.7). Steady state conditions were established with the ICS in the fully integrated mode of control and control rod 7 in group 6 inserted 9 inches below the group 6 average position. The results of the transient are shown in Figure 8.1-9. The ICS ran unit load demand back to 570 MW(e) at 12%/min upon detection of the asymmetric rod by the rod drive control circuit. Reactor power was reduced to 55% FP at approximately 30% FP/min, which was within the acceptable limit of 30±2.6% FP/min. Feedwater cross limits developed at about 2 minutes into the transient which limited the overall runback rate to 11% FP/min.

8.1.3.5 Turbine Trip From 76% FP

An inadvertant turbine trip occurred during a simulated turbine overspeed test with the reactor at 76% FP. The effects of this transient on selected system parameters are shown in Figure 8.1-10. Tripping the turbine caused the generator to trip and initiated a reactor power runback to 14% FP at 18.7% FP/min. The rapid closing of the turbine stop valves opened the turbine bypass and main steam safety valves to relieve secondary side pressure. Main steam pressure reached a maximum value of 1062 psig and was subsequently controlled at 895 psig, as required. The maximum values for reactor coolant average temperature and pressure were 590°F and 2320 psig, respectively. Pressurizer level reached a low value of 200 inches and a high of 296 inches. The ICS established level control in the steam generators, as required, in a power reduction to 15% FP.

At the time of the turbine trip, the sudden increase in steam generator pressure caused a rapid decrease in feedwater flow, as is seen in Figure 8.1-10. The demand for feed flow remained high due to Btu limits on the generators. Feedwater flow recovered and remained approximately constant from ½ to 2 minutes into the transient, but was higher than the demand for feed flow. This caused a hold in power runback at 45% FP to correct the negative T AVE error which developed. Feed flow then followed feedwater demand and power was then reduced to the proper value.

8.1.3.6 Generator-Reactor Trip From 100% FP

A generator trip from 100% FP was conducted as part of the test program. The transient was initiated by opening the main generator breakers after establishing steady state conditions with the ICS in fully automatic control. The ICS started to run back unit load demand at 20%/min and the turbine began to overspeed due to loss of load. The turbine governing valves started to close and turbine header pressure increased to 1040 psi. The turbine bypass and main steam safety valves opened. Reactor coolant pressure increased to the high pressure limit and the reactor tripped at approximately 4 seconds after the generator breakers opened. The turbine tripped due to the reactor trip. Figure 8.1-11 shows the behavior of selected primary and secondary system parameters for this transient.

After the reactor trip, turbine header pressure was controlled at 10[illegible] psi, as required, and level control was established in the steam generators. Reactor coolant average temperature increased to 581°F and then decreased after the reactor tripped. The behavior of feedwater flow after the trip is due to several error signals received by the ICS. The increase in turbine header pressure caused Btu limits to override the feedwater demand signal and the demand signal followed the Btu limits for several seconds after the trip. Feedwater pump speed decreased to minimum speed due to the drop in feedwater demand. After neutron power dropped to zero, the neutron error modification to the feedwater demand was removed and feedwater demand followed the Btu limits in a reduction of feedwater flow.

8.1.4 CONCLUSIONS

An analysis of scheduled and unscheduled transient results have led to several minor modifications to the ICS during the startup test program to optimize ICS performance. Transient testing of the unit was conducted at the design ramp rates in the turbine following, reactor/steam generator following and fully integrated mode of control at 40%, 76% and 100% full power. The ICS successfully maneuvered the plant in all three modes of control at 40% and 76% FP and in the fully integrated mode at 100% FP. The 100%-50%-100% load swing was completed at 8%/min in the turbine following mode and at 6%/min on the decrease and 4%/min on the increase in the reactor/steam generator following mode. The ability of the ICS to control the plant upon a loss of reactor coolant pump, a loss of main feedwater pump, a dropped control rod and a turbine trip at 76% FP was demonstrated. The reactor tripped on high RC pressure after a load rejection at 100% full power. Studies are in progress to determine the necessary corrective actions required to keep the reactor on line following a load rejection.

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TABLE 8.1-1

SUMMARY OF TRANSIENTS REPORTED IN SECTION 8.1

Transient Number	Description	Data Presented in Table	Data Presented in Figure
1-3	Unit Load Transient Test at 40% FP	8.1-3	X
4-6	Unit Load Transient Test at 76% FP	8.1-4	X
7-12	Unit Load Transient Test at 100% FP	8.1-5	8.1-1 to 8.1-6
13	Reactor Coolant Pump Trip at 25% FP	X	8.1-7
14	Main Feedeater Pump Trip at 100% FP	X	8.1-8
15	Asymmetric Rod Runback at 76% FP	X	8.1-9
16	Turbine Trip at 76% FP	X	8.1-10
17	Generator-Reactor Trip at 100% FP	X	8.1-11

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TABLE 8.1-2

SUMMARY OF
UNIT LOAD TRANSIENT TESTS AT POWER

Transient Number	ICS Mode of Operation	Load Change Decrease (% FP)	Load Change Increase (% FP)	Test Ramp Rate (%/min)	Accept Rate (%/min)
A. Unit Load Transient Test at 40% Full Power					
1	Fully Integrated	55-45	45-55	1 to 5	5
2	Turbine Following	55-45	45-55	1 to 5	5
3	Rx/SG Following	55-45	45-55	1 to 5	5
B. Unit Load Transient Test at 76% Full Power					
4	Fully Integrated	76-66	66-76	1 to 5	5
5	Turbine Following	76-66	66-76	1 to 5	5
6	Rx/SG Following	76-66	66-76	1 to 5	5
C. Unit Load Transient Test at 100% Full Power					
7	Fully Integrated	100-90	90-100	1 to 5	5
8	Turbine Following	100-90	90-100	1 to 5	5
9	Rx/SG Following	100-90	90-100	1 to 5	5
10	Fully Integrated	100-50	50-100	1 to 10	10
11	Rx/SG Following	100-50	50-100	1 to 10	10
12	Turbine Following	100-50	50-100	1 to 10	10

TABLE 8.1-3

SUMMARY OF UNIT LOAD
TRANSIENT TEST RESULTS AT 40% FP

Transient Number	ICS Mode of Operation	Power Change (% FP)	Rate, %/min Ave	Rate, %/min Max	ΔT AVE, °F Min	ΔT AVE, °F Max	ΔTc, °F Min	ΔTc, °F Max	ΔTHP, psi Min	ΔTHP, psi Max
1	Fully Integrated	54-44	4.3	4.3	-1	0	-0.2	+0.2	0	+24
		45-55	4.2	4.2	0	+0.5	-0.2	+0.2	-30	0
2	Turbine Following	54-45	4.1	4.1	-1.5	+0.5	-0.2	+0.2	-6	+6
		45-55	4.2	5.4	-0.5	+0.5	-0.2	+0.2	-8	+8
3	Rx/SG Following	53-43	5.0	5.0	-1.0	0	-0.2	+0.3	0	+50
		41-60	4.5	6.9	-2.0	0	-0.2	+0.1	-48	+60

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TABLE 8.1-4

SUMMARY OF UNIT LOAD TRANSIENT TEST RESULTS AT 76% FP

Transient Number	ICS Mode of Operation	Power Change (% FP)	Rate, %/min		ΔT AVE, °F		ΔTHP, psi	
			Ave	Max	Min	Max	Min	Max
4	Fully Integrated	70-58	5.5	5.5	-0.5	0	0	+24
		58-71	5.9	5.9	0	+1.0	-24	0
5	Turbine Following	68-58	3.8	3.8	-2.0	0	0	0
		56-74	4.1	4.4	-0.5	+1.0	-6	0
6	Rx/SG Following	74-63	1.8	3.8	-0.5	+1.0	-18	+36
		63-74	2.4	6.3	-2.0	+2.0	-36	+36

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SUMMARY OF UNIT LOAD
TRANSIENT TEST RESULTS AT 100% FP

Transient Number	ICS Mode of Operation	Power Change (% FP)	Rate, %/min Ave	Rate, %/min Max	ΔT AVE, °F Min	ΔT AVE, °F Max	ARC Press, psi Min	ARC Press, psi Max	ΔTHP, psi Min	ΔTHP, psi Max
A. Ramp Rates at 5%/min, 100%-90%-100%										
7	Fully Integrated	100-88	6.0	6.7	-1.1	+0.6	-28	+20	-2	+17
		88-100	6.5	6.5	-0.4	+0.6	-4	+16	-16	+2
8	Turbine Following	98-85	3.9	7.4	-1.6	+0.4	-20	+48	-3	+2
		85-100	5.0	8.0	-1.0	+1.0	-4	+76	-1	+2
9	Rx/SG Following	100-87	5.6	12.0	-1.2	+0.8	-8	+48	0	+40
		87-98	5.7	15.0	-0.6	+1.2	-4	+16	-20	+12
B. Ramp Rates at 10%/min, 100%-50%-100%										
10	Fully Integrated	99-56	9.1	15.4	-2.0	+0.6	-40	+48	-12	+16
		55-91	10.8	10.8	-0.8	+1.0	0	+56	-36	+16
11	Rx/SG Following	94-62	5.2	12	-1.6	+0.1	-16	+40	0	+76
		62-91	4.9	14.4	-0.8	+0.4	-8	+20	-72	0
12	Turbine Following	98-58	8.3	11.4	-1.6	+0.6	-40	+48	-12	+12
		58-91	10.4	10.4	-3.8	+1.4	-20	+96	-48	+24

TABLE 8.1-5

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TRANSIENT NO. 7, 100% FP
FULLY INTEGRATED ICS MODE
5%/min

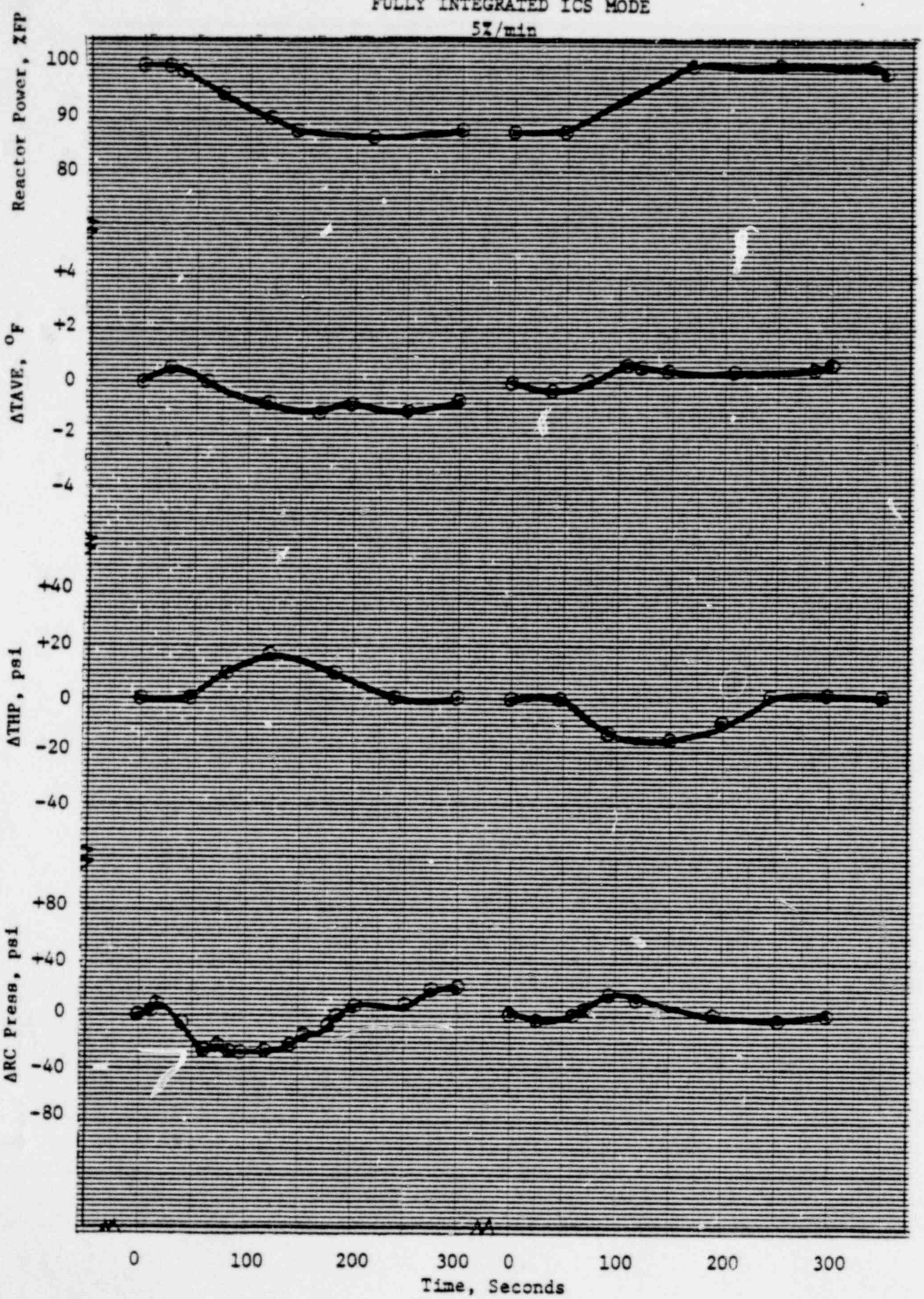


FIGURE 8.1-1

TRANSIENT NO. 8, 100% FP
TURBINE FOLLOWING MODE
5%/min

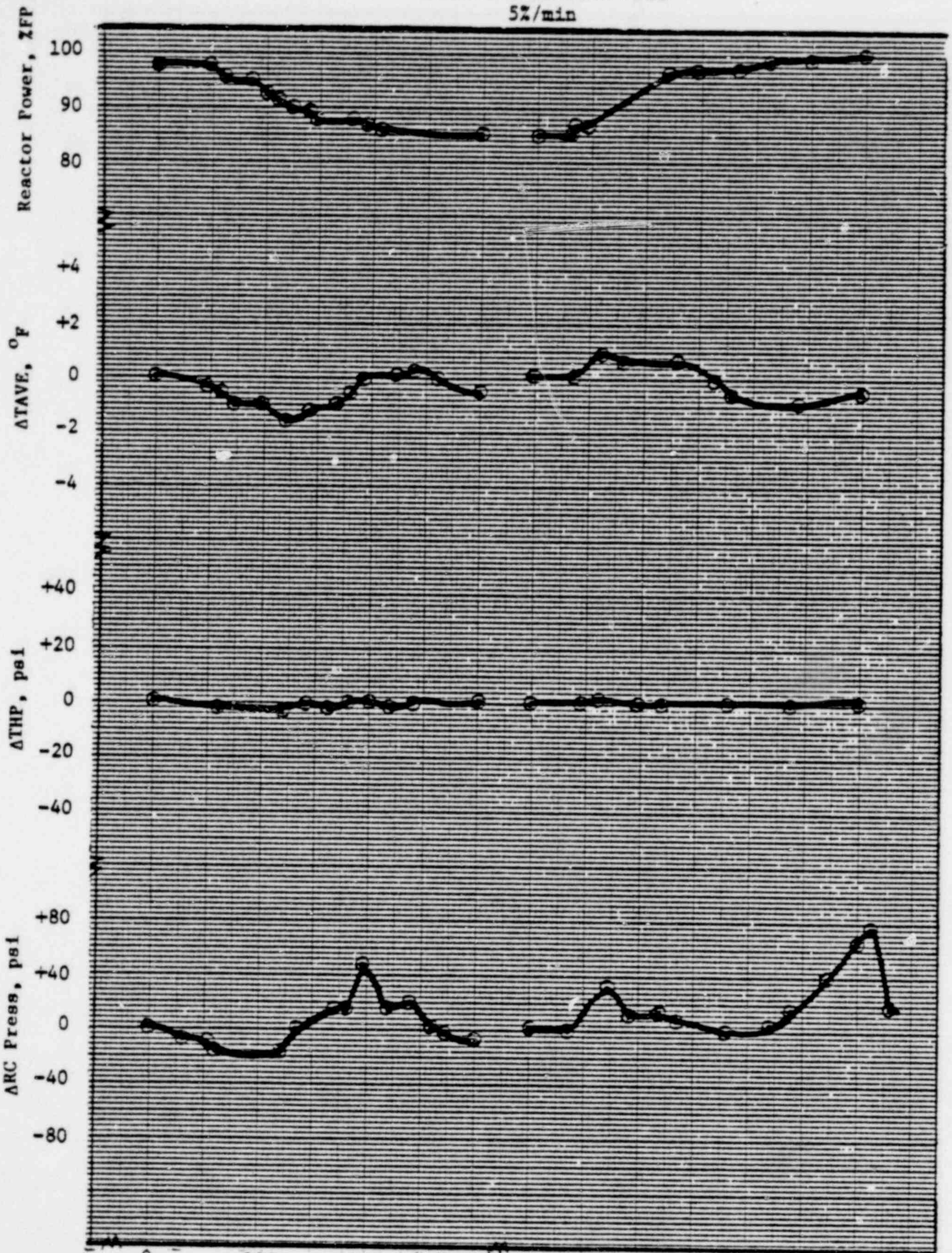


FIGURE 8.1-2

TRANSIENT NO. 9, 100% FP
Rx/SG FOLLOWING MODE
5%/min

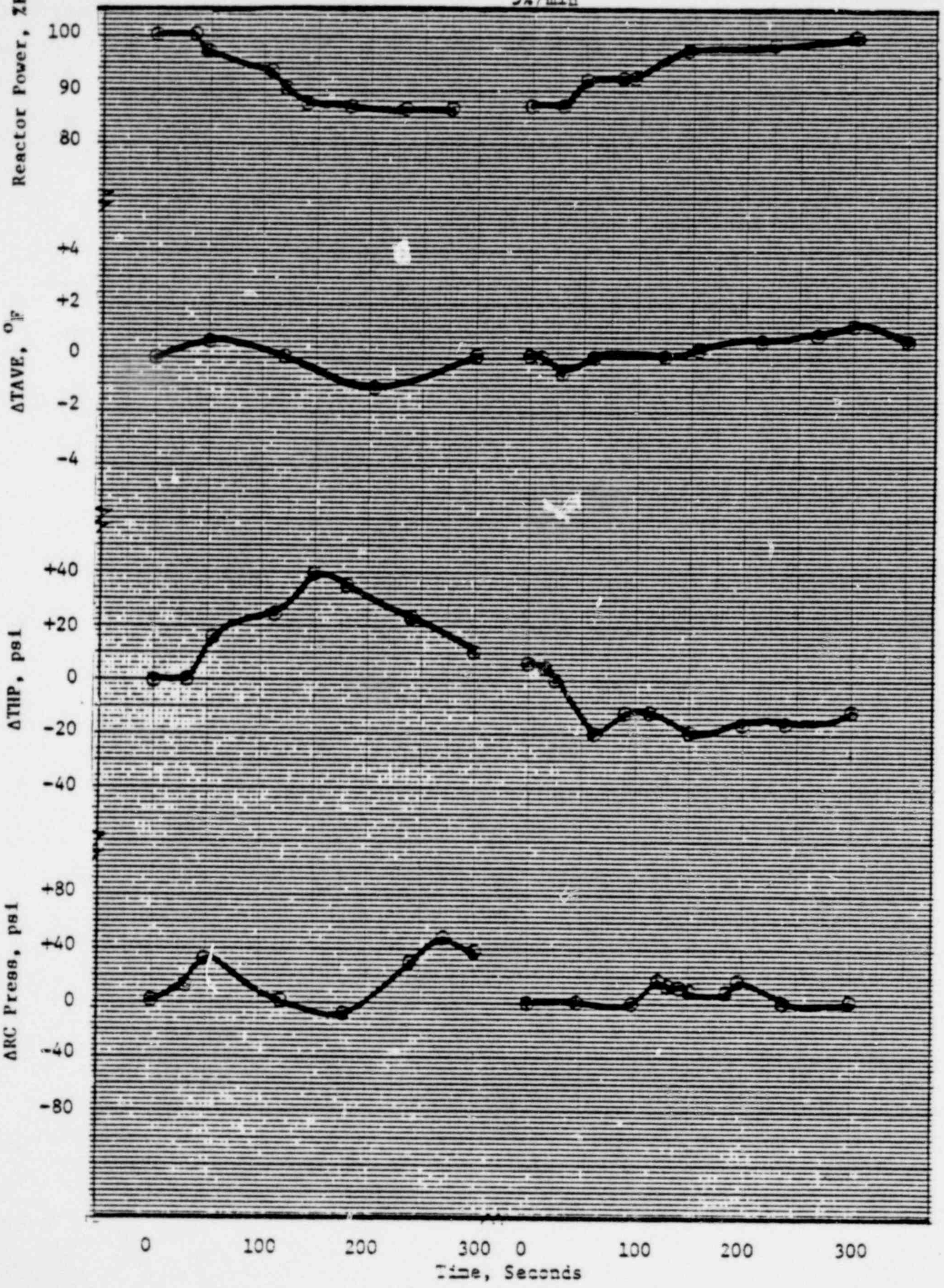


FIGURE 8.1-3

1414 214

TRANSIENT NO. 10, 100% FP
FULLY INTEGRATED ICS MODE
10%/min

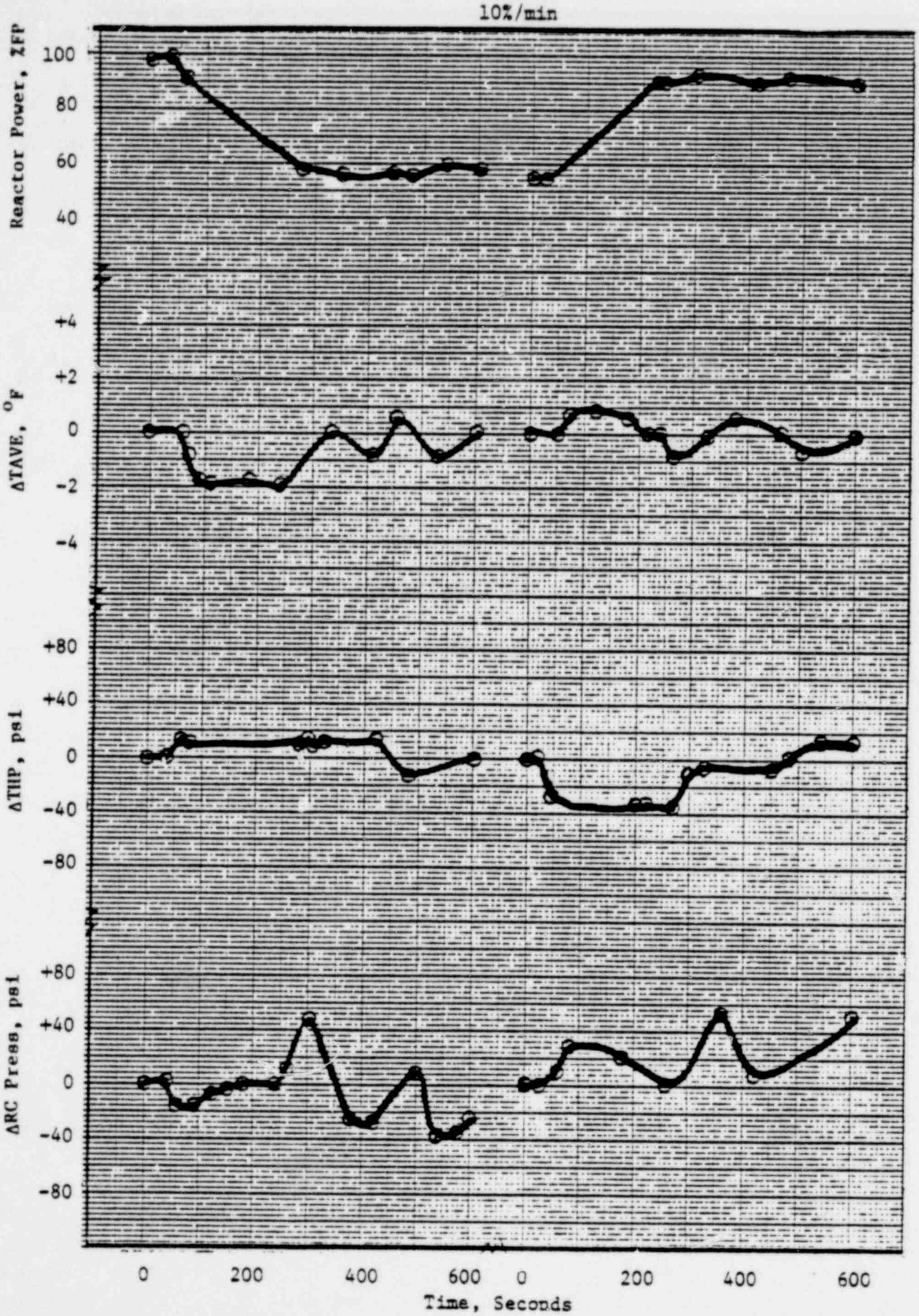


FIGURE 8.1-4

1414 215

TRANSIENT NO. 11, 100% FP
Rx/SG FOLLOWING MODE
10%/min

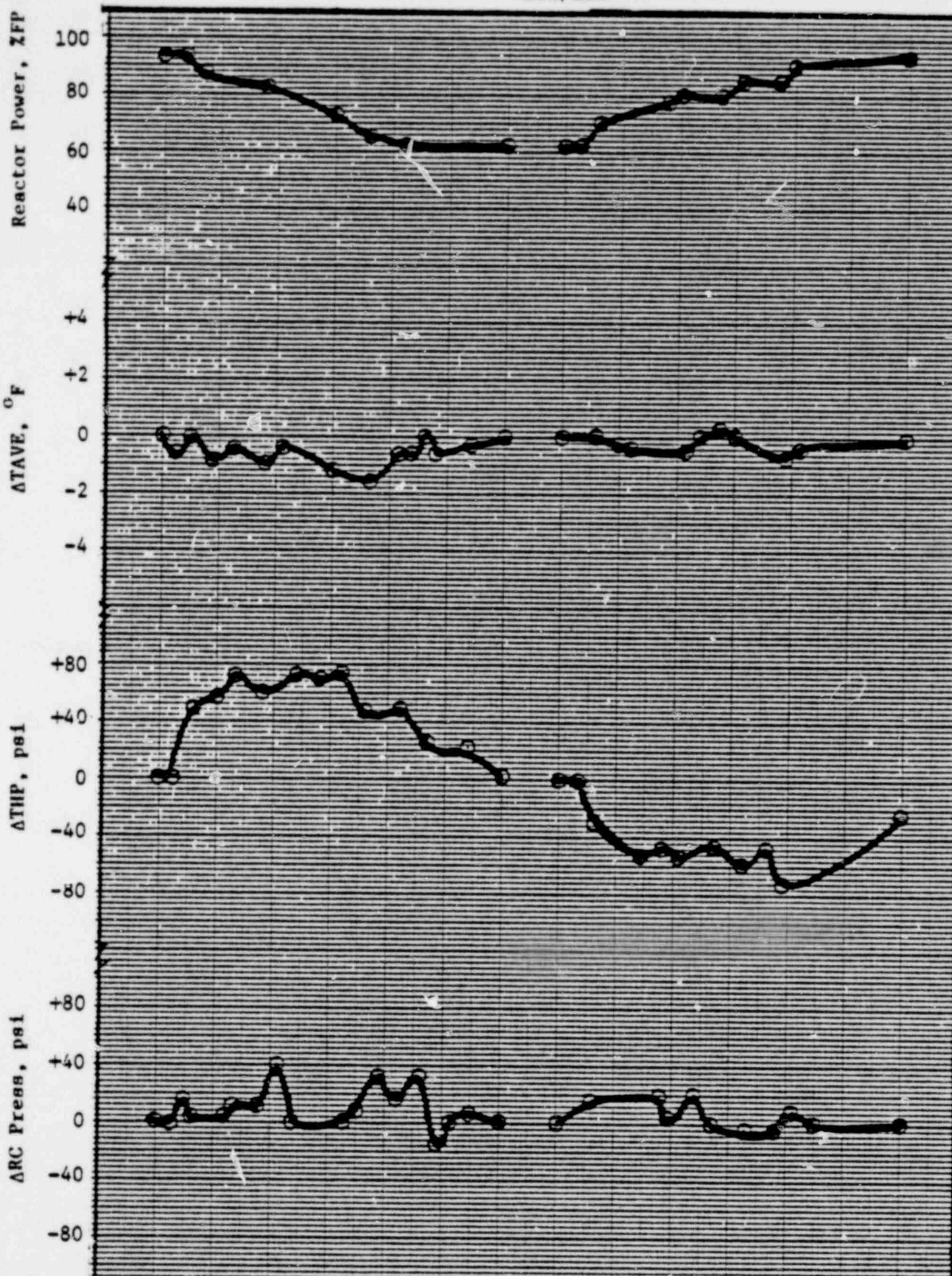


FIGURE 8.1-5

1414 216

TRANSIENT NO. 12, 100% FP
TURBINE FOLLOWING MODE
10%/min

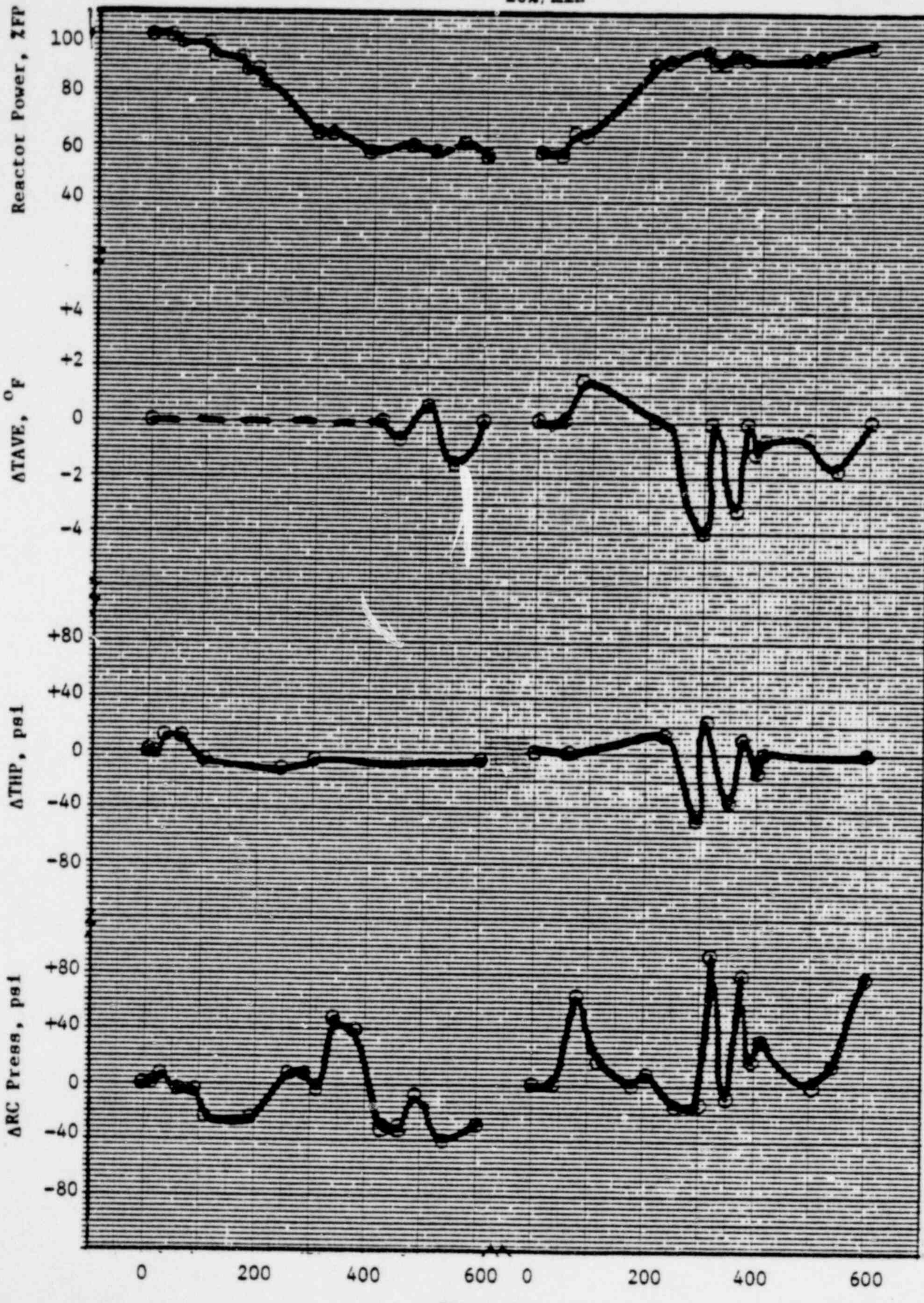
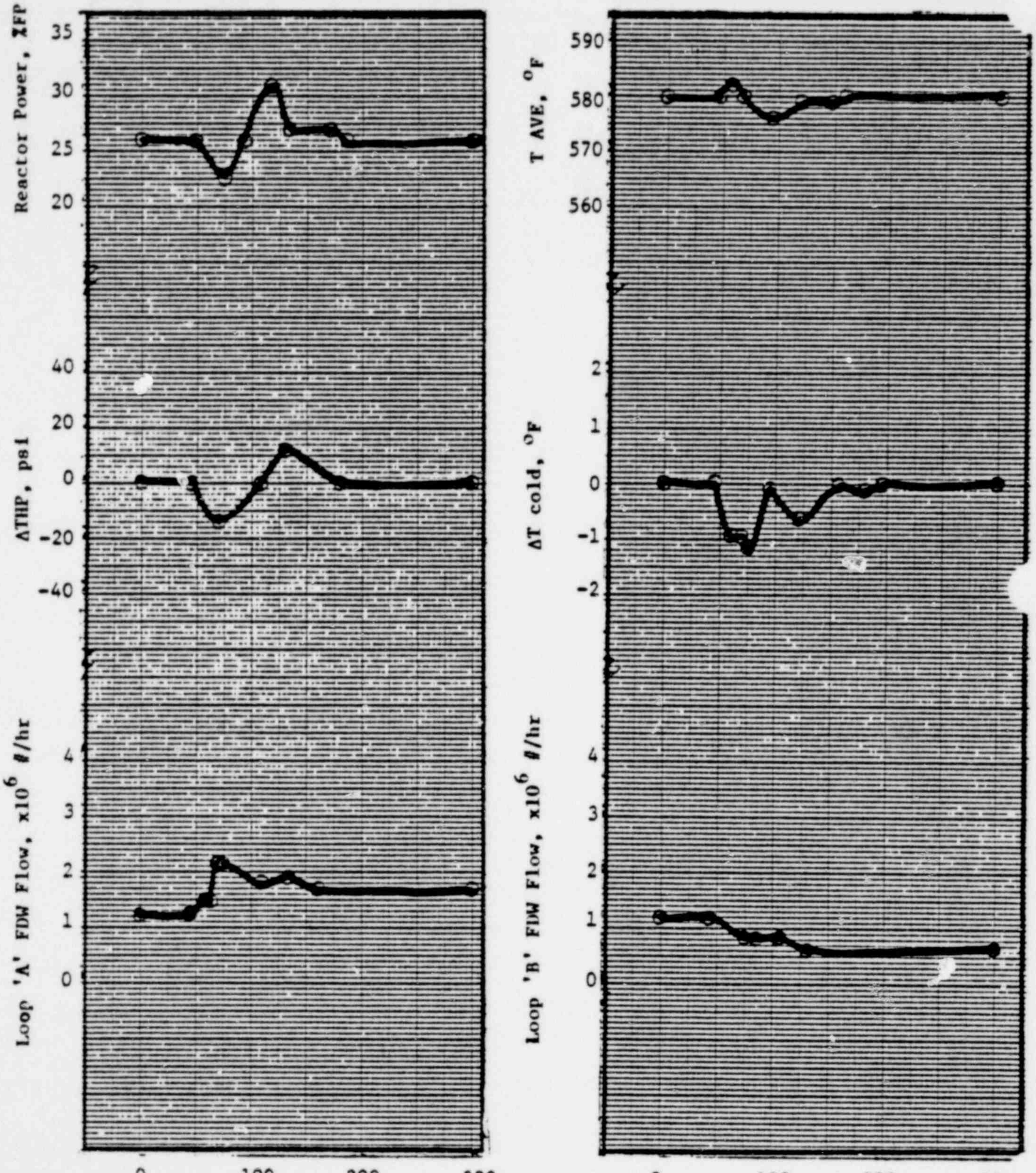


FIGURE 8.1-6

1414 217

TRANSIENT NO. 13, 25% FP
REACTOR COOLANT PUMP TRIP



Time, Seconds

FIGURE 8.1-7

TRANSIENT NO. 14, 100%FP
MAIN FEEDWATER PUMP TRIP

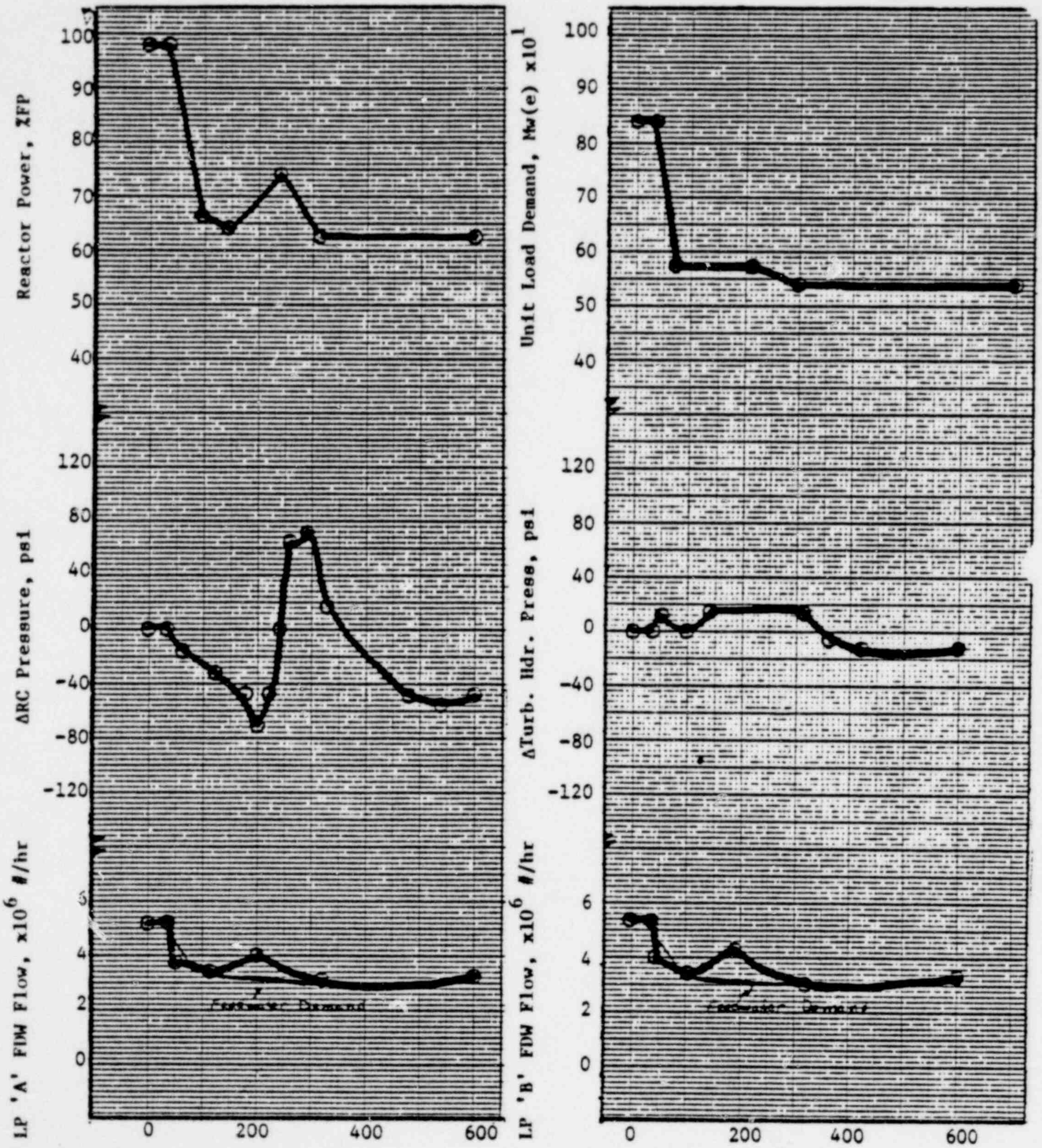


FIGURE 8.1-8

1414 219

TRANSIENT NO. 15, 76%FP
ASYMMETRIC ROD RUNBACK

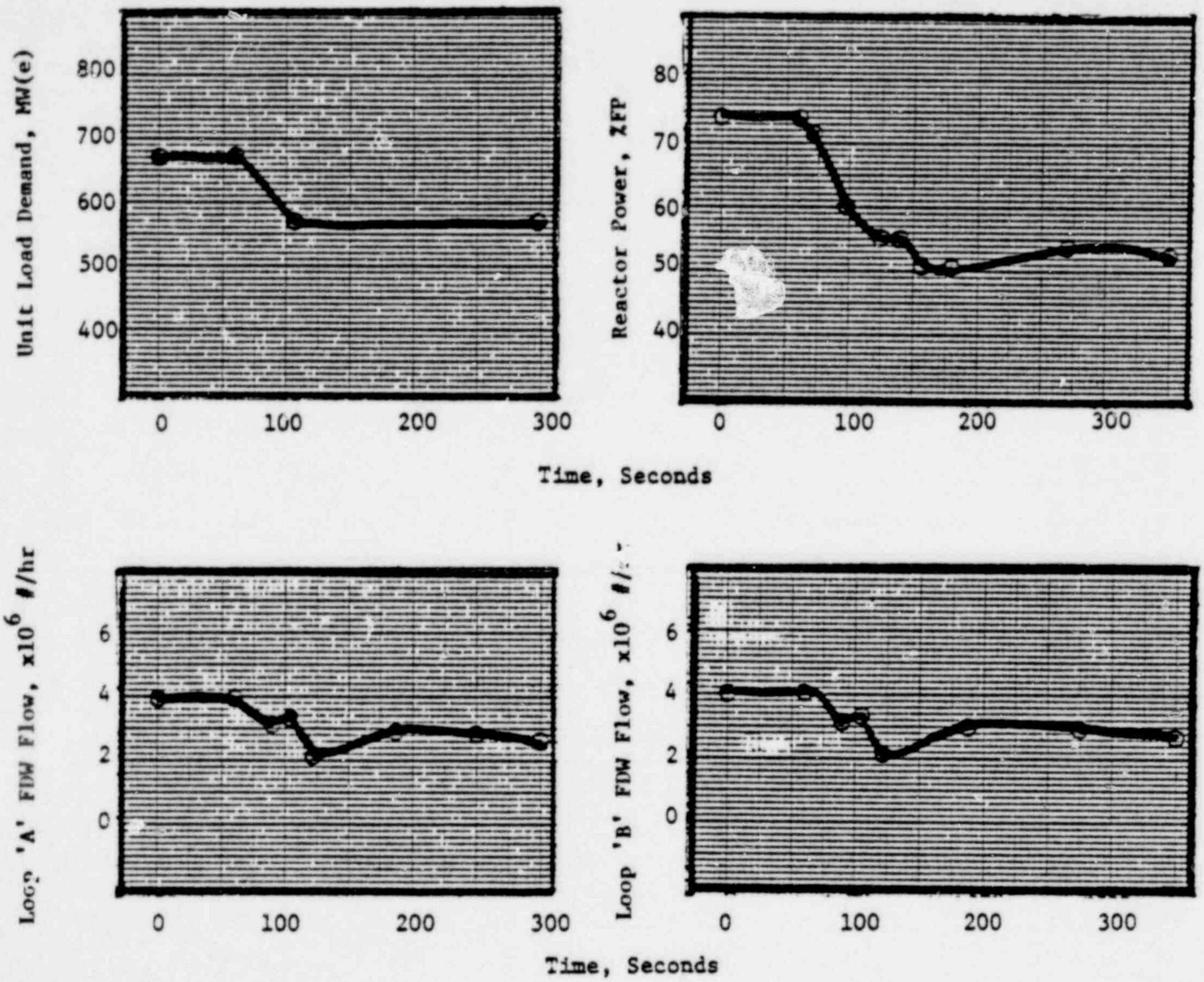


FIGURE 8.1-9

1414 220

TRANSIENT NO. 16, 76% FP
TURBINE TRIP

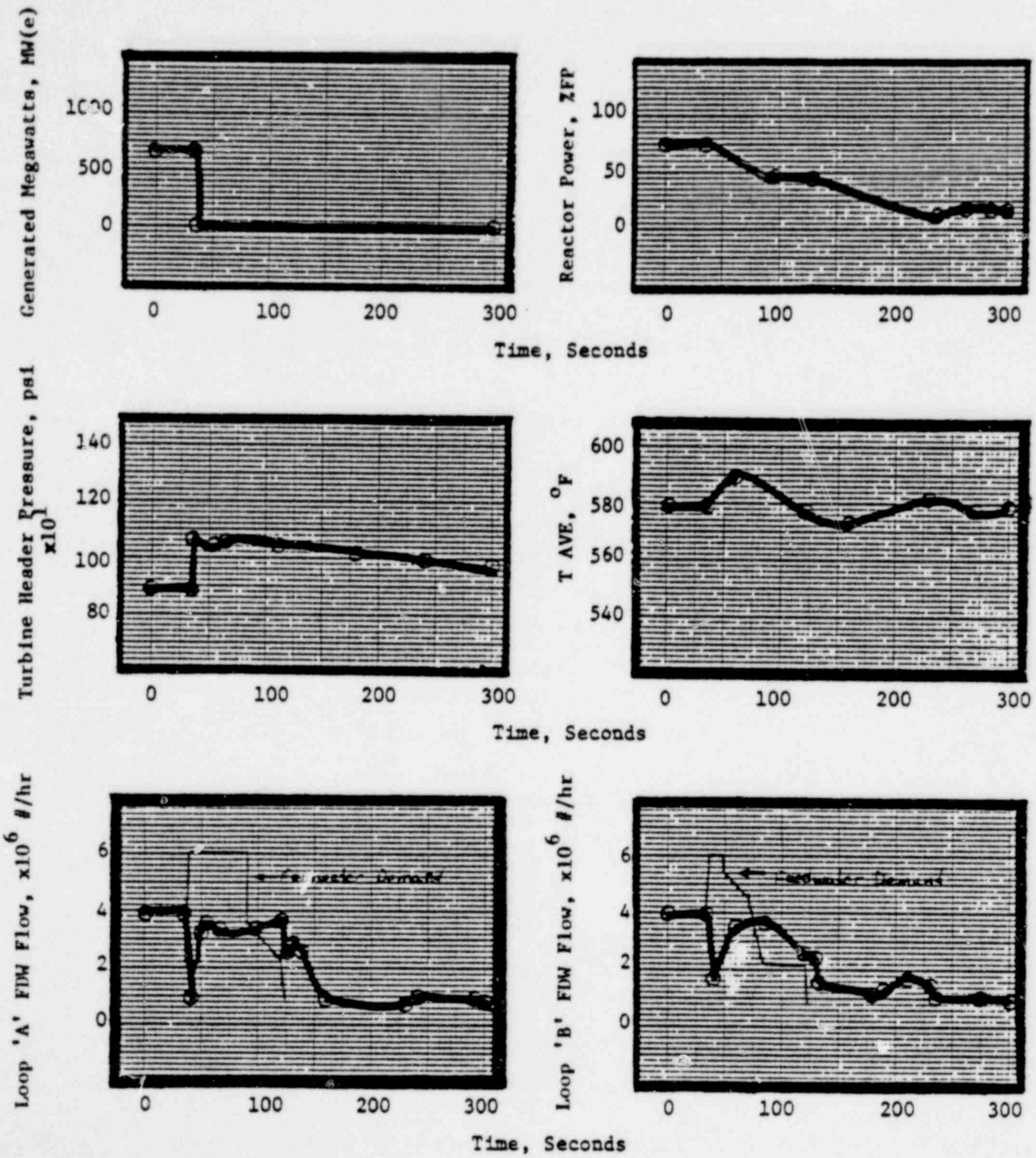
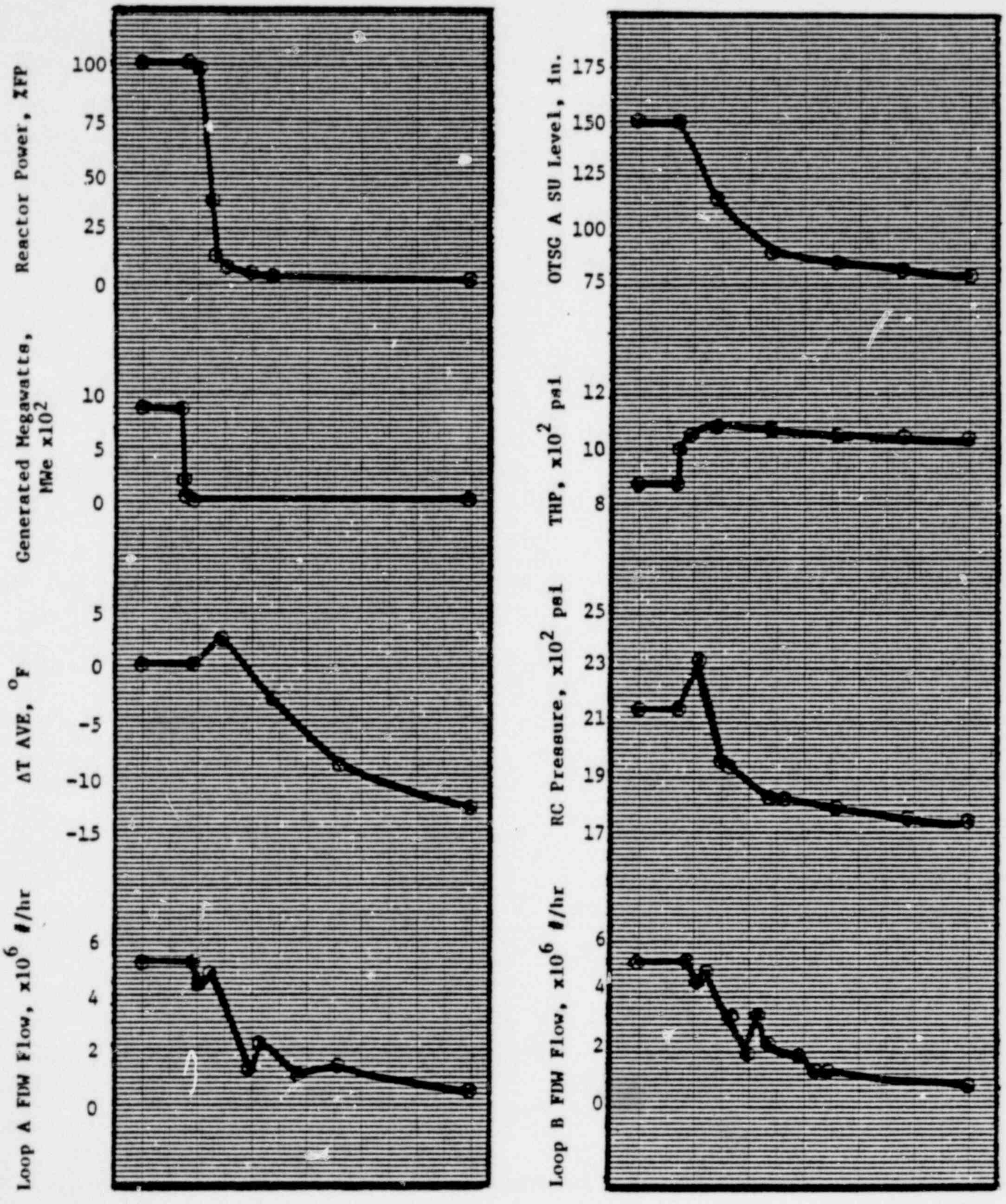


FIGURE 8.1-10

1414 221

TRANSIENT NO. 17, 100% FP
GENERATOR - REACTOR TRIP



Time, Seconds

FIGURE 8.1-11

8.2 LOSS OF OFFSITE POWER TEST

8.2.1 PURPOSE

The purposes of the loss of offsite power test were to verify that:

1) The automatic response of the reactor and auxiliary systems results in the plant being controlled in such a manner as to prevent fuel damage and excessive pressure in the reactor coolant system.

2) The plant can be shutdown with power supplied from the station batteries and emergency diesel generators.

3) The emergency procedure to be followed during a Station Blackout Emergency (EP 1202/02) is a satisfactory document for verifying automatic actions and performing manual actions to maintain the plant in a safe condition and prevent equipment damage.

8.2.2 TEST METHOD

The normal plant electrical lineup consists of the turbine generator output current passing through voltage step up transformers and then through the generator breakers to the substation. The substation is of the "breaker and a half" design. Plant house load is supplied from the substation through breakers to two auxiliary transformers, either one of which is designed to supply 100% of house load.

Prior to the test, all house load was placed on the B auxiliary transformer. The breaker between the substation and the A auxiliary transformer was opened and placed in the "pull-to-lock" position. Reactor power was brought to 15%. With this alignment, all incoming power to the plant could be cut off by opening the breaker between the substation and the B auxiliary transformer: if trouble developed, (for example, if the diesel generators failed to start up and assume load), the A auxiliary transformer could be immediately energized to supply required load by taking its feeder breaker out of "pull to lock".

Brush recorders and the reactimeter were set up to monitor primary and secondary plant parameter behavior and an equipment status log was kept to record the status (running or off) of equipment subject to automatic action following the loss of power. RCS pressure was monitored by a 0-3000 psi, 0.1% accuracy Heise gage and compared with the acceptance criteria of $\leq$2750 psig. The trend of plant fission product activity was monitored before and after test performance to determine if any fuel damage occurred during the test. Emergency Procedure 1202/02 (Station Blackout) was followed to verify all automatic actions and to perform all required manual actions to maintain the plant in a safe shutdown condition and keep from damaging equipment.

8.2.3 TEST RESULTS

All required automatic actions following the loss of power at 15% occurred as expected and listed in EP1202/02. They are:

1) reactor trips
2) turbine generator trips
3) control room DC lighting comes on
4) steam driven emergency feedwater pump starts and raises steam generator levels to 95% on operating range

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5) all four reactor coolant pumps trip
6) condensate, condensate booster and feedwater pumps trip
7) generator DC oil pump, RC pump DC oil pumps and DC lube oil pumps start
8) A and B diesel generators start and pick up block 1 (non safeguard) loads

EP1202/02 was satisfactorily verified for use during occurrence of a Station Blackout.

Both RCS temperature and pressure decreased after the blackout; therefore, peak RCS temperature was the initial hot leg temperature of 582°F and peak RCS pressure was the initial value of 2164 psig. Minimum RCS pressure was 2004 psig just prior to normal power restoration. Pressurizer level also decreased after the blackout from an initial value of 231 inches to a low point of 120 inches just prior to normal power restoration. Main steam peaked at 1008 psig in the A header and 1032 in the B header. As a result of this pressure differential, emergency feedwater preferentially went to the A steam generator; therefore the A steam generator level increased rapidly while the B steam generator level slowly decreased to 20 inches on the startup range level instrument and held there. In order to keep from exceeding the cooldown rate limit of 100F°/hr, both emergency FW valves were placed in manual control and closed when the A steam generator level reached 28% on the operating range. At this point, the valve to the B steam generator should have been opened to bring its level up even to the A, but this was not done. Had the valves been left in auto, the valve feeding the A steam generator would have throttled down at 95% and the valve feeding the B steam generator would have remained open until its level rose to 95% also.

Following this test, B&W evaluated the total amount of natural circulation flow through the core to remove decay heat as a function of steam generator level and reduced the post blackout control level setpoint from 95% to 50% on the operating range. Review of our test data justifies this change, since RCS pressure and temperature never increased above initial steady state values after the blackout, and final control levels were 28% in one steam generator and 20 inches in the other.

Radiochemical analyses before and after the blackout indicated no increase in fission product activity and therefore no failed fuel as a result of test performance.

8.2.4 CONCLUSIONS

The Station Blackout Emergency Procedure (EP1202/02) was verified for use during a blackout and all required automatic actions occurred as expected.

RCS pressure and temperature never increased above their initial steady state values. Peak pressure of 2164 psig was far less than the 2750 psig upper limit. There was no increase in RCS fission product activity as a result of test performance.

The emergency feedwater supply valves were placed in manual and closed before steam generator levels increased to 95% to keep from exceeding the 100F°/hr cooldown limit. B&W has since revised the 95% setpoint to 50% to avoid exceeding the cooldown limit. Prior to EFW valve closure, the steam generators were being fed unevenly due to the difference between A and B header pressure. Had the EFW valves been left in auto, the supply valve to the A OTSG would have throttled down when A level reached 95% and the B OTSG would have increased to 95%.

8.3 SHUTDOWN FROM OUTSIDE THE CONTROL ROOM

8.3.1 PURPOSE

The purposes of the shutdown from outside the control room test were to demonstrate the ability to remove decay heat from the reactor coolant system with control of all systems at locations remote from the control room and to verify portions of Emergency Procedure 1202/37 (Cooldown from Outside the Control Room).

8.3.2 TEST METHOD

Emergency Procedure 1202/37 (Cooldown from Outside the Control Room) provides instructions for shutting the plant down from power operation and cooling it down to 140°F in the event the control room must be evacuated. The procedure provides steps for the entire shutdown to be performed from outside the control room, but lists those actions which should be performed in the control room prior to evacuation as time permits. Also, considerations regarding nuclear safety override considerations regarding potential equipment damage in shutting down and cooling down the plant. A remote control center has been established where sufficient instrumentation and communications channels exist for monitoring and directing a safe plant shutdown and cooldown. Equipment actuation is performed at various locations throughout the plant. In light of the above remarks, we established the following bases for performing the shutdown from outside the control room test:

1) Up to two minutes could be spent in the control room to perform initial steps of the emergency procedure prior to evacuation.

2) The emergency procedure should be modified slightly to minimize the potential for equipment damage during the test.

3) Demonstration of the capability to bring the plant to hot shutdown conditions (Tave $\geq$525°F and at least 1% $\Delta k/k$ shutdown) is sufficient to satisfy the requirements of USAEC Regulatory Guide 1.68 (Pre-op and Initial Startup Test Programs for Water Cooled Reactors), so only those steps of the emergency procedure would be performed.

4) The test would be performed utilizing only the number of personnel scheduled on a normal shift to conduct operations; however, there would be additional personnel to act as observers and to take over operation from the control room in the event assistance was required.

Using these bases, the test was performed as follows:

1) Prior to evacuating the control room

   a) the reactor was manually tripped
   b) both motor driven emergency feedwater pumps were started
   c) both main feedwater pumps were tripped
   d) the letdown block valve was closed to terminate letdown flow
   e) the nuclear and turbine plant communications channels were cross tied
   f) the concentrated boric acid emergency injection valve was opened and both injection pumps were started.

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The emergency procedure was modified such that the reactor coolant pumps were not tripped (to keep from exceeding the 100F°/hr cooldown limit) and the operating makeup pump was not tripped (to continue the supply of RC pump seal injection water to prevent damaging the seals).

2) Following the above actions, the Shift Foreman announced the control room evacuation over the PA system. Then he and a control room operator proceeded to the remote control center to monitor and direct plant shutdown. One of these men was available to turn pressurizer heaters on and off as required to maintain RCS pressure. Parametric indication available at the remote control center was:

   a) pressurizer level
   b) makeup tank level
   c) RCS pressure
   d) RCS temperature
   e) both OTSG levels

3) Another operator proceeded to the chemical addition control panel where he shutdown the concentrated boric acid injection pumps and then to the makeup valve manifold in the auxiliary building where he controlled pressurizer level and makeup tank level as directed by the Shift Foreman.

4) Another operator proceeded to the intermediate building to the emergency feedwater valves to control steam generator levels between 5% and 95% on the operating range as directed by the Shift Foreman.

5) A second operator proceeded to the intermediate building to the atmospheric dump valves to control RCS temperature as directed by the Shift Foreman.

6) The Shift Foreman made calculations to verify that the reactor was 1% $\Delta k/k$ subcritical.

The test was terminated after stable conditions were maintained for twenty minutes.

8.3.3 TEST RESULTS

The actions performed in the control room prior to evacuation took one minute and ten seconds to perform compared with the two minutes allotted.

The remote control center instrumentation and communications proved adequate to monitor and direct plant operations to maintain the reactor plant at hot shutdown conditions. RCS pressure was maintained at ≈2155 psig without operation of the pressurizer heaters. Both pressurizer level and makeup tank level were adequately maintained manually from the makeup valve manifold. Steam generator levels were maintained between 5% and 95% by regulating the emergency feedwater valves in manual. Levels were kept near their initially low values to avoid excessive cooldown rates. RCS temperature was satisfactorily decreased from 579°F to 527°F by controlling the atmospheric dump valves in manual. The shutdown margin was calculated to be 2.43% even after xenon reactivity decayed to zero.

The only problem encountered during the test was the plugging of the concentrated boric acid pump suction strainer with boron crystals and small pieces of the paper bags the boron is shipped in. A field change is being made to include redundant parallel heat traced strainers to eliminate this problem.

The reactor plant was satisfactorily maintained in a hot shutdown condition for greater than the required twenty minutes by the minimum shift complement of five men.

8.3.4 CONCLUSIONS

The reactor plant can be maintained in a safe hot shutdown condition from locations outside the main control room by the minimum shift complement of five men. The alternate control center contains sufficient instrumentation and communications to permit satisfactory monitoring and direction of shutdown operations.

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8.4 UNIT ACCEPTANCE TEST

8.4.1 PURPOSE

The purpose of the unit acceptance test was to verify that the energy output from the nuclear steam supply system meets or exceeds the equivalent of 10,521,000 lbm/hr steam flow at the steam generator outlet nozzles at conditions of 925 psia and 569°F when supplied with feedwater at conditions of 455°F and 1030 psia at the inlet to the steam generators, with a primary coolant letdown flow of 55 gpm and makeup water supply temperature of 125°F. This energy output corresponds to a gross secondary side output of $\geq$2449 MWt.

8.4.2 TEST METHOD

This test was conducted in accordance with the provisions set forth in the ASME Power Test Code 4.1-1964, Steam Generating Units. It was performed as follows:

1) Prior to the test, an agreement was reached as to the specific instrumentation to be used for recording test parameters.

2) The above instruments were calibrated within two weeks of test performance.

3) The test consisted of a preliminary four hour run during which data was recorded every ten minutes. The data was then averaged over the four hour interval.

4) The four hour data averages were then used to determine the values for use in the following equation:

$$MW_t = \frac{FW_A\ (H_{AS} - H_{AF}) + FW_B\ (H_{BS} - H_{BF})}{3,412,142\ \text{Btu/MWhr}}$$

where MW_t = gross NSS megawatts thermal output

FW_A = loop A feedwater flow in lbm/hr

H_{AS} = loop A main steam enthalpy in Btu/lbm

H_{AF} = loop A feedwater enthalpy in Btu/lbm

FW_B = loop B feedwater flow in lbm/hr

H_{BS} = loop B main steam enthalpy in Btu/lbm

H_{BF} = loop B feedwater enthalpy in Btu/lbm

5) Since the MW_t calculated above far exceeded 2449 MW_t, the preliminary run was accepted and therefore qualified as the first official run.

6) A second four hour run was made in the same manner as 3) above and evaluated as in 4) above.

8.4.3 TEST RESULTS

Average gross NSS megawatts thermal output during the first four hour run was 2551.016 MW_t; average for the second four hour run was 2554.213 MW_t. The average for the entire unit acceptance test was 2552.615 MW_t as compared with the acceptance criteria of $\geq$2449 MW_t.

Average loop A main steam temperature was 591.4°F and average loop B main steam temperature was 591.8°F compared with the acceptance criteria of $\geq$569°F.

Average main steam flow was 10,770,000 lbm/hr at 591.6°F with feedwater at 461.1°F. Gross electrical megawatts generated was 848.2 MW_e throughout the unit acceptance test. Condenser vacuum was lower than design values, which accounts for the lower than expected (870 MW_e) electrical output.

The unit acceptance test was completed on August 26, 1974, which completed the TMI Unit I power escalation test program.

8.4.4 CONCLUSIONS

The Three Mile Island Unit I nuclear steam supply system produces 2552.615 MW_t gross energy output compared with the warranty value of $\geq$2449 MW_t. Main steam temperature is 591.6°F compared with the warranty of $\geq$569°F. These results indicate a substantial margin of NSS performance above warranty specifications.

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