STARTUP REPORT

DOCKET NO. 50-423
LICENSE NO. NPF-49

MILLSTONE UNIT 3
NORTHEAST NUCLEAR ENERGY COMPANY

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FORWARD

This report addresses the conduct and results of the startup test program for Millstone Unit 3 and spans the period from Initial Fuel Loading through Commercial Operation and Warranty Run. It is submitted in accordance with the requirements of USNRC Regulatory Guides 1.16, Revision 4, and 1.68, Revision 2, and Millstone Unit 3 Technical Specification 6.9.1.1.

1.0 INTRODUCTION

Millstone Unit 3 consists of a Westinghouse 4 loop pressurized water nuclear steam supply system rated at 3411 MWT and a General Electric turbine-generator rated at 1204 MWE. The overall net electrical output of the unit is 1150 MWE. Millstone Unit 3 is located adjacent to Millstone Unit 1 (a 660 MWE General Electric BWR) and Millstone Unit 2 (a 870 MWE Combustion Engineering PWR) on an approximately 500 acre site on the north shore of Long Island Sound in the town of Waterford, Connecticut. The unit utilizes a subatmospheric containment design with a supplemental leak collection and release system (secondary containment) to further limit offsite releases in the event of a design basis accident.

The ownership of Millstone Unit 3 is divided among 15 joint owners. The majority owners are the Northeast Utilities subsidiaries, Connecticut Light and Power Company and Western Massachusetts Electric Company. The remaining portion is divided among 13 New England public and private utilities.

The joint owners have designated Northeast Nuclear Energy Company (NNECo), a wholly owned subsidiary of Northeast Utilities, to act as their agent and representative in matters relating to the design, construction, testing, licensing, operation and maintenance of Millstone Unit 3. NNECo presently performs a similar function for Millstone Units 1 and 2. The unit was designed and constructed by Stone & Webster Engineering Corporation.

The unit was constructed under Construction Permit CPPR-113 and currently operates under Operating License NPF-49. Operating License NPF-44 was issued on November 11, 1985 to permit initial fuel load and low power operation (not to exceed 5 percent of rated thermal power). Operating License NPF-49 was subsequently issued on January 31, 1986 to permit full power operation.

2.0 PROJECT SUMMARY CHRONOLOGY

The following is provided as an overview of the major milestones in the chronology of Millstone Unit 3.

DATE	EVENT
08-09-74	Construction Permit CPPR-113 issued by the then Atomic Energy Commission (AEC).
09-74	First structural concrete (turbine building) is placed.
04-75	Rebar placement for the containment mat begins.
09-78	First containment wall concrete is placed.
07-79	The turbine generator stator is set in place.
10-80	Reactor vessel and containment polar crane are set in place.
06-81	Steam generator erection is begun.
11-82	Emergency diesel generators are installed.
01-17-83	The system turnover process and preoperational test program are begun.
07-18-83	The reserve station service transformers (RSST) are energized.
12-09-83	Energization of 4160 volt switchgear is begun.
12-03-84 to 12-22-84	Perform steam generator secondary side hydrostatic testing.

DATE	EVENT
04-16-85	Receive Special Nuclear Material (SNM) license SNM-1950.
04-19-85 to 04-25-85	Perform RCS cold hydrostatic testing.
04-24-85	The first shipment of reactor fuel is received.
05-15-85	Unit 3 emergency drill is conducted.
06-10-85 to 10-19-85	Perform turbine building hot functional testing.
07-10-85 to 07-24-85	Perform the containment structural integrity test (SIT) and integrated leak rate test (ILRT).
07-24-85	The last shipment of reactor fuel is received.
08-16-85 to 09-06-85	Perform the engineered safeguard features (ESF) test
09-27-85 to 11-02-85	Perform precore hot functional test.
09-17-85	Perform initial turbine roll utilizing RCP heat as the heat source.
11-25-85	Receive Operating License NPF-44 permiting fuel load and operation up to 5% reactor power.
11-26-85 to 12-03-85	Perform initial fuel loading. Startup test program begins.

DATE	EVENT
01-11-86 to 01-23-86	Perform post core hot functional testing.
01-23-86	Initial criticality achieved at 2200 hours.
01-24-86 to 01-31-86	Perform low power physics testing (LPPT).
01-31-86	Receive Full Power Operating License NPF-49.
02-01-86 to 04-21-86	Perform the power ascension test program.
02-15-86	Achieve 30% power.
02-16-86	Initial synchronization to the grid.
03-17-86	Achieve 50% power.
03-26-86	Achieve 75% power.
04-15-86	Achieve 90% power.
04-17-86	Achieve 100% power.
04-23-86	Commercial operation is declared.
04-25-86 04-29-86	Perform the unit warranty run. Complete the startup test program.

3.0 PREOPERATIONAL TEST PROGRAM OVERVIEW

The Preoperational Test Program officially began with the first system turnover from Construction to Startup, on 01-10-83, of the motor control centers to support the water treatment system. This turnover process continued for both systems and buildings through to completion of system turnovers, on 06-05-85, of the yard security system and the completion of building turnovers, on 11-04-85, of the yard area. This was the last of 234 turnover packages.

The Preoperational Test Program included component testing and system flushing which, in most cases, preceded the preoperational testing of systems. System pressure testing (except steam generator and RCS hydros) was performed prior to system turnover. Preoperational testing continued through 1983 and 1984, leading up to the transition to milestone testing. Major milestones that were established are listed below along with the start and completion dates for each milestone.

Milestone	Date Start/Date Complete
Plant on Permanent Power	07-18-83
Steam Generator Hydro	12-04-84/12-20-84
RCS Cold Hydro	04-14-85/04-24-85
Fuel Receipt	04-24-85/07-24-85
Emergency Drill	05-15-85
Containment ILRT	07-12-85/07-15-85
Engineered Safety Features Test	08-16-85/09-06-85
Turbine Building Hot Functional Test	06-10-85/10-19-85
Precore Hot Functional Test	09-27-85/11-02-85

A summary description of each milestone follows.

Steam Generator Hydrostatic Test

The Steam Generator Hydrostatic Test involved the hydrostatic testing of the secondary side of the four steam generators and their associated piping. This milestone was subdivided into one test for each generator. The boundaries for each test included the attached piping systems out to the nearest isolation points. For main steam piping, the main steam isolation valves provided isolation and the main feedwater piping was isolated at the steam generator feedwater stop valves. The remaining piping systems were isolated inside containment by installation of blank flanges or valve positioning.

The generators were filled for test with water from the condensate storage tank after being preheated to 180°F. A temporary transport system was utilized from the discharge side of the condensate system makeup pumps through the containment equipment hatch to each generator. A recirculation skid was provided to assist in chemical addition and temperature maintenance prior to start of the test within the 120°F to 180°F test range.

The hydrostatic testing to 1570 psig began with the "A" generator, which completed its test on 12-04-84, and concluded with the last generator test completed on 12-12-84. Tube to tubesheet leaks were detected on generators A, B and C. Subsequent to repair of the detected tube sheet leaks, retesting was performed. This activity incorporated six separate tests with a maximum test pressure of 840 psig. This testing commenced 12-12-84 and was completed on 12-20-84.

Following completion of the test, the Steam Generators were placed in a wet lay-up condition with a nitrogen overpressure.

Reactor Coolant System Cold Hydrostatic Test

The Reactor Coolant System (RCS) Cold Hydrostatic Test involved the pressure testing of the reactor vessel and associated piping/components to 3107 psig. In addition, the test involved the initial fill and venting of the RCS as well as the initial operation of the reactor coolant pumps (RCPs). Prior to assembling the reactor vessel to close the RCS pressure boundary, the reactor vessel internals were installed. During the test, the RCPs were utilized to heat the inventory of the RCS above the 150°F lower limit based on brittle fracture concerns.

The assembly sequence for the reactor vessel began on 04-03-85 when preparations were started for reactor vessel internals installation. On 04-04-85 the internals were installed and preparations began for installation of the vessel head. The head was installed on 04-05-85. The RCS fill sequence began on 04-13-85 and was complete on 04-15-85. During this sequence, the tensioning of the reactor vessel head was completed on 04-14-85. The RCPs were bumped on 04-19-85. The vibration testing runs of the RCPs were completed on 04-20-85 and the heatup of the RCS was begun. During the period of 04-20-85 to 04-24-85, the pressure boundary was groomed and minor leakage paths repaired. Final pressurization to test pressure began on 04-24-85 and was completed that day.

Fuel Receipt

The Fuel Receipt milestone was established to provide a framework to accomplish fuel receipt on site with subsequent fuel assembly transfer to a safe storage facility. Significant prerequisites to this milestone included testing of the following systems: fuel pool cooling and purification, radiation monitoring, fuel building HVAC, fuel building fire protection and detection, and fuel handling equipment. Additional prerequisites included fuel building turnover, establishment of a physical security plan for the fuel building and surrounding areas, operator fuel handling training, and establishment of radiation and fire protection programs for the fuel building, all of which would lead to receipt of a license from the NRC to receive and store special nuclear material. Upon completion of all prerequisites, the NRC issued license SNM-1950 on 04-16-85. Specific fuel shipment scheduling and receipt concerns were resolved with Westinghouse representatives over the next few days, and the initial receipt of 14 fuel assemblies occurred on 04-24-85. The final fuel shipment was received 07-24-85.

SIT/ILRT

The Structural Integrity Test/Integrated Leak Rate Test was performed to demonstrate the structural integrity of containment at 1.15 times design pressure and to measure the leak rate from containment at peak accident pressure. Major test prerequisites included completion of Type B and C leakage tests on containment isolation valves and penetrations (including equipment and personnel hatches), installation of pressurization equipment, and containment turnover process. During the performance of the prerequisite activities, some delays were caused by Type C test failures, rework and subsequent retest of containment isolation valves.

Initial pressurization for the SIT commenced on 07-10-85, but this effort was stopped when an open containment leakage path was discovered. In this instance, misalignment of Leakage Monitoring System lines penetrating containment resulted in an open-ended pipe. This deficiency was corrected by installation of a jumper, and pressurization recommenced after an eight-hour delay. Peak pressure of 52 psig achieved within 24 hours, and the SIT was completed the morning of 07-12-85 with no deficiencies noted. Pressurization for the ILRT was commenced nine hours later; full pressure of 39.4 psig was achieved, and the test run commenced on 07-13-85. After a 24 hour hold, leakage was determined to be 52.57 scfm (10% of the acceptance criteria). Depressurization was completed 07-15-85.

Engineered Safety Features (ESF) Test

The ESF Test was started on 08-16-85, and completed on 09-06-85. The test was divided into two separate sections: ESF without loss of power and ESF with loss of power.

The ESF test without loss of off-site power was performed with the breakers of the major ESF-actuated equipment placed into the test position. This was done to verify safeguard logic before placing the plant under the dynamic transients of the operating equipment. The ESF test with loss of off-site power was then performed to verify emergency diesel performance, correct sequential loading of ESF equipment and proper train separation.

The performance of the ESF test without loss of power revealed some logic errors with HVAC equipment and inadequately sized slave relays in the Main Steam Isolation Valve control logic. These concerns were subsequently corrected and satisfactorily retested.

The ESF test with loss of off-site power revealed a deficiency in the diesel sequencer logic in that the diesel output breakers failed to close due to incorrect time delay settings on certain control relays. Also, several electrical busses were not stripped during the LOP, Orange Train test. These problems were resolved and successfully retested.

Turbine Building Hot Functional Test

The overall purpose of the Turbine Building Hot Functional Test (TBHFT) was to prepare, cleanup and test the secondary side of the plant utilizing Auxiliary Steam to ensure system operability, and to establish a level of reliability for integrated system operation. This was all in preparation to support the activities associated with Precore Hot Functional and subsequent Startup and Power Ascension Testing. The test procedure (3-INT-2006) was utilized as a controlling document which integrated and sequenced all the secondary plant activities, i.e., plant conditions, Phase II tests, condensate/feedwater train cleanup, operator training and validation of the plant's operating procedures. Major objectives for this test included:

1. Demonstrate the ability to steam seal the main turbine and feed pump turbines utilizing the gland seal steam system. Auxiliary boiler steam was utilized for this process.
2. Demonstrate the ability to draw vacuum and maintain a design pressure (1.5 in HgA) in the condenser. As required, condenser vacuum boundary leaks were to be located and corrected.
3. Demonstrate the ability to operate the condensate system in the short and long recycle modes.
4. Demonstrate the ability to clean the hotwell, condensate and feedwater systems prior to feeding forward through the use of the condensate mixed bed demineralizers. In conjunction with this process, the proper operation of the condensate chemical feed system and portions of the turbine plant sampling system was verified.
5. Perform the initial no-load uncoupled and coupled runs of the main turbine driven feedwater pumps utilizing auxiliary steam supplied from the auxiliary boilers.

During coupled runs, the feed pumps were operated in the recirculation mode only, due to limited steam supply from the auxiliary boilers.

6. Perform the initial coupled run of the motor driven feedwater pump.
7. Perform the Phase II tests for the following systems:
   - gland seal steam
   - condenser air removal
   - secondary plant sampling (partial)
   - condensate system (partial)
   - condensate chemical feed
   - feedwater and recirculation

The test was released for establishment of initial conditions and performance of system lineup on 06-10-85. Physical testing began on 06-14-85 when the main turbine was placed on turning gear. Testing and secondary side system grooming continued until 11-06-85 when the test procedure was officially completed. The procedure was kept open into the Precore Hot Functional Test so it could serve as a coordinating document for various balance of plant related Phase 3 tests.

Several major testing interruptions were experienced during the performance of 3-INT-2006. No impact on the precore hot functional testing or any other milestone event was caused by these interruptions.

- <u>07-08-85 to 08-11-85</u>

  A seawater leak into the hotwell was caused when the condenser air removal piping in the B condenser, D waterbox separated from the tubesheet face and allowed a seawater ingress into the hotwell. The separation was caused by corrosion of the bolts holding the penetration flange against the tubesheet face. During inspection of all waterboxes, corrosion of the inlet side tubesheets was observed. Engineering analysis determined the corrosion of both the bolt heads and tubesheets was the result of improper material compatibility which was accelerated by non-optimal performance of the waterbox cathodic protection system. Repairs undertaken included changeout of all air removal line flange bolts with a more resistant alloy, epoxy coating of the inlet side tubesheets and inlet waterboxes. Cathodic protection system setup, testing and operator training were performed to ensure optimum system performance.

While the measures were being taken to correct the cause and results of the corrosion, a full scale flushing program was performed on the condensate and feedwater system, up to feed stops, in order to remove the chloride contamination caused by the seawater intrusion. The chloride levels in the condensate and connected systems were brought to acceptable levels and with the mechanical repairs effected, testing was restarted on 08-13-85.

- <u>08-18-85 to 09-23-85</u>

  On 08-15-85 a crack was discovered in the upper crossover piping between the A and B condensers. Efforts to temporarily seal the crack using a mastic compound were unsuccessful and the secondary plant was shut down, vacuum broken and the hotwell pumped down to facilitate repairs. During the process of correcting the crack, additional internal condenser support damage was discovered. Engineering analysis indicated insufficient internal bracing had been installed, and supplemental supports were specified. After this additional material had been installed, a further delay was experienced while the ESF test with loss of normal power (3-INT-2004) was performed. During this latter delay, the outlet side of the waterboxes were epoxy coated as a preventative measure.

This was the last delay due to an equipment malfunction. TBHFT testing was recommenced on 09/23/85. By this point the Precore Hot Functional Test was underway, plant heatup was in progress, and the remaining TBHFT activities were performed in parallel with HFT.

In addition to the initial scoped testing for TBHFT on 10-17-85, the initial roll of the main turbine took place. On 10-19-85, the main turbine was synchronized to the grid for the first time and approximately 65 MW generated. The TBHFT was concluded at this point.

All objectives of the test were met with minor exceptions. Due to testing and system grooming which took place during the TBHFT, the secondary side was able to fully support PCHFT and the subsequent startup testing.

Pre-Core Hot Functional Test

The Pre-core Hot Functional Test started on 09-27-85 and was completed on 11-02-85. In general, all systems required for plant operation were tested under normal operating conditions. The major objectives of the test were to take the unfueled plant from a cold shutdown condition, through heatup, testing at normal operating temperature and pressure, and return to a cold condition. During this time the following design requirements and system functions were verified:

- Freedom of movement during thermal expansion for major components.
- The capacity of the Chemical and Volume Control System to maintain Reactor Coolant System (RCS) pressure during solid pressure control and to purify the letdown steam while the RCS was at operating pressure.
- The operation of the atmospheric steam dump valves and the condenser dump valves during cooldown and at normal operating system conditions.
- The RCS heat loss to ambient at operating temperature and pressure.
- The operability of both the primary and secondary sample systems and chemical addition systems.
- The operability of both the main and auxiliary feedwater pumps.
- The starting up and paralleling of the main turbine-generator to the grid.
- The RCS leakage calculation method.
- The capability for remote shutdown and cooldown of the reactor plant.
- The initial vibration testing and monitoring of components during normal operation.
- The operability with a heat load of the plant's ventilation systems.
- The initial check of the RCS thermocouple/RTD cross-calibration.
- The ability to isolate an RCS loop while maintaining primary pressure control within the isolated loop.
- The operation of the pressurizer pressure and level control systems.
- The functionality of the Voice Page and Evacuation Alarm systems with normal plant background noise.
- The ability of the plant to withstand a loss of instrument air.

This test was also used to passivate the RCS by operating at an elevated (>500°F) temperature for 28 days and to obtain a minimum of 10 days of RCP flow induced vibration cycles on the reactor internals.

All testing was covered in the base procedure (3-INT-3000) and 34 associated appendices. All planned testing was completed except for that on the boron thermal regeneration system which, due to equipment problems, was delayed until a later date. The deficiencies discovered during testing were addressed on a schedule consistent with plant and system operability requirements.

4.0 INITIAL FUEL LOAD
3-INT-4000

OBJECTIVE

The Initial Fuel Load procedure provides a safe, organized method for the initial core load.

DISCUSSION

Initial fuel load was conducted over the period of 11-26-85 to 12-04-85. The operation is summarized in Section 4.1, Initial Fuel Load Chronology.

Prior to fuel load, proper alignment and calibration of the two Source Range channels (SR 31, 32) and the three Temporary Detectors (TD A, B, C) were verified in accordance with 3-INT-4000, Appendix 4003, Core Load Instruments and Neutron Source Requirements. Baseline background count rates were taken. In addition, a neutron source was lowered near each detector to verify correct channel response. This latter check was required to be performed within 8 hours of beginning core load.

From dry storage in the Spent Fuel Pool (SFP), each fuel assembly was transferred by the Spent Fuel Pool Bridge and Hoist (SFPBH) to the Fuel Transfer System (FTS). After the FTS cart moved the fuel into containment, the Single Integrated Gripper Mast Assembly (SIGMA) refueling machine would engage the fuel assembly and load it in the proper core location. Fuel movement in containment was under the direction of a fuel handling Senior Reactor Operator. Overall fuel load operations were directed by Reactor Engineering Personnel. The actual loading sequence was controlled by 3-INT-4000, Appendix 4005, Initial Core Loading. In addition to delineating all movements for each fuel assembly, this appendix also governed TD movement and provided guidance for obtaining count rate data.

Neutron monitoring was provided by SR 31 and 32 and TD A, B and C. As each fuel assembly was lowered into the core, count rates were monitored. During the loading sequence, count rate data was collected and analyzed in accordance with 3-INT-4000, Appendix 4004, Inverse Count Rate Ratio Monitoring. After count rates had stabilized, two counting trials of 100 seconds each were taken on all detectors. The counts were used to calculate an Inverse Count Rate Ratio (ICRR), which was then plotted versus the number of fuel assemblies loaded. The ICRR is used as an indicator of the approach to criticality and this plot ensured there was no unanticipated approach to criticality. Appendix 4004 also provided for statistical verification of detector performance during extended fuel load operation suspensions.

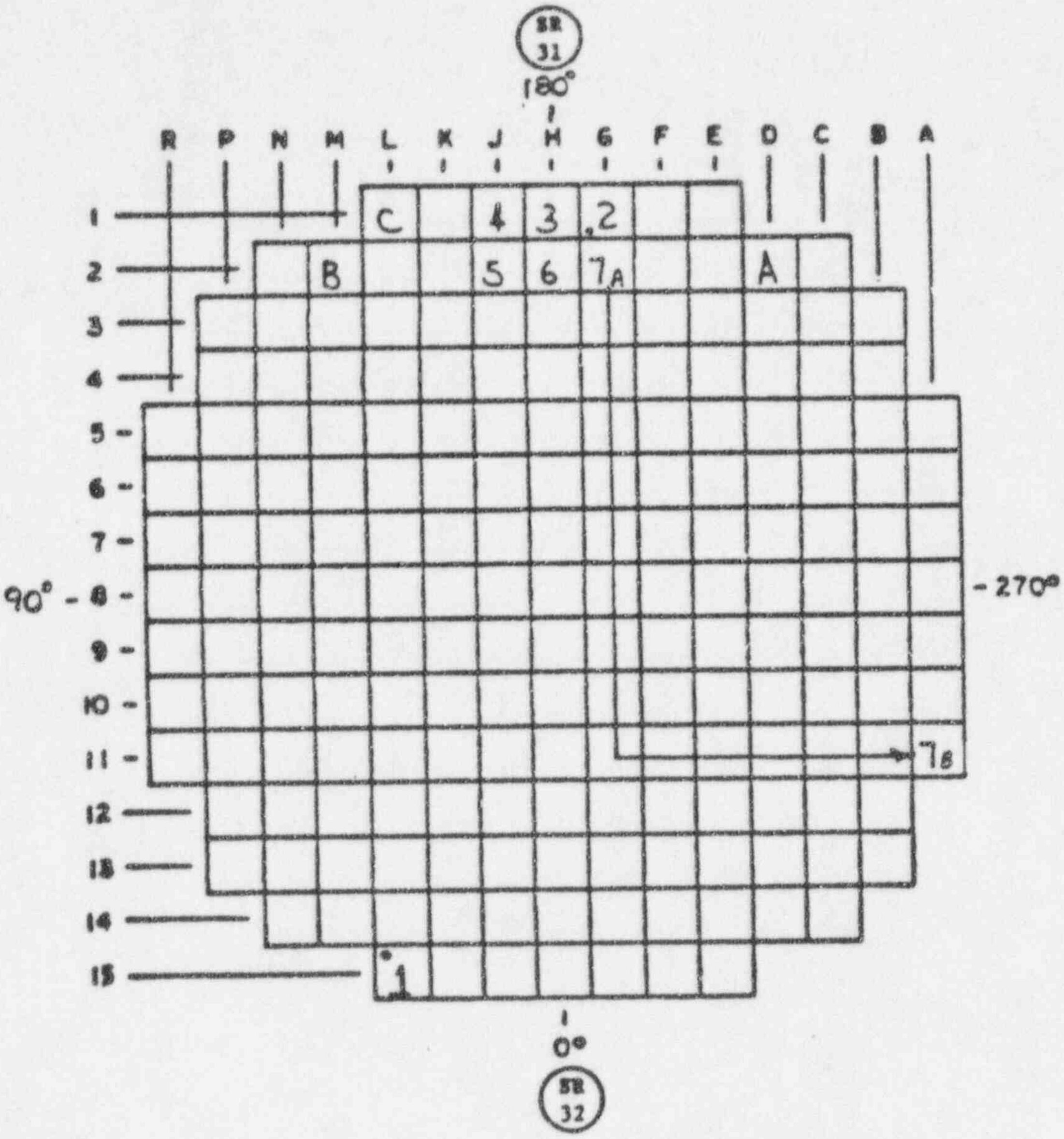
After the core was loaded, Appendix 4006, Core Map, was performed to verify correct core loading. Reactor Engineering and QA performed a visual scan of all fuel assemblies and inserts using an underwater camera. Correct fuel assembly, and fuel assembly insert locations were verified. The core was further verified to be free of debris. A permanent video record was also made.

RESULTS

As stated previously, the initial fuel load began on 11-26-85 at 1825 and was completed on 12-2-85 at 2310. The initial core loading sequence is shown in Figures 4.0-1 through 4.9-10. All five neutron monitoring channels responded as expected, and there were no unexpected increases in subcritical multiplication. Noise was intermittently observed on SR 31 and was determined to be from SIGMA machine movement and nearby welding activities. Inverse Count Ratio Response (ICRR) plots for SR31 and 32 and TD A, B and C are shown in Figures 4.0-11 and 4.0.12.

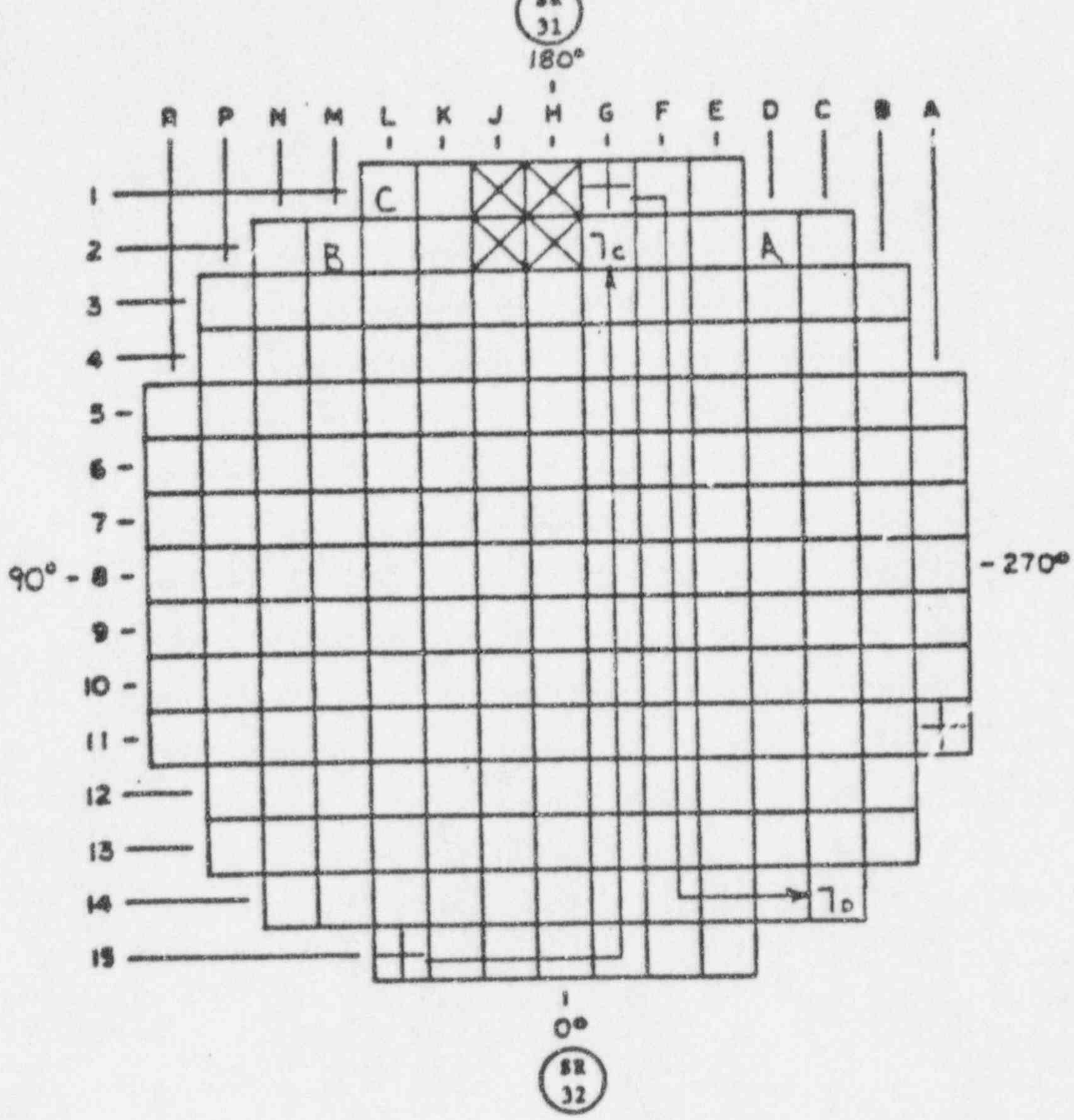
Due to a bow in an adjacent fuel assembly, assembly B49 could not be loaded into core location E04 per the loading sequence. The sequence was changed per the recommendation of Westinghouse Fueling Services personnel to leave E04 vacant and load around it. When E04 was "boxed in" by adjacent assemblies, B49 was successfully loaded into E04.

Throughout the entire loading operation, approximately 2½ days were lost due to various problems with the SFPBH and SIGMA machine. Problems with the SFPBH were mainly due to overload limit switches and spurious resetting of control setpoints. Problems with the SIGMA were mainly: 1) The SIGMA machine did not realize when it was fully down; 2) The overload/underload trips were set too low/high, respectively; and 3) The east side motor and associated drive system were not functioning properly. Corrective maintenance was performed in each case to allow fueling operations to continue. No problems were encountered with performing the Core Map.

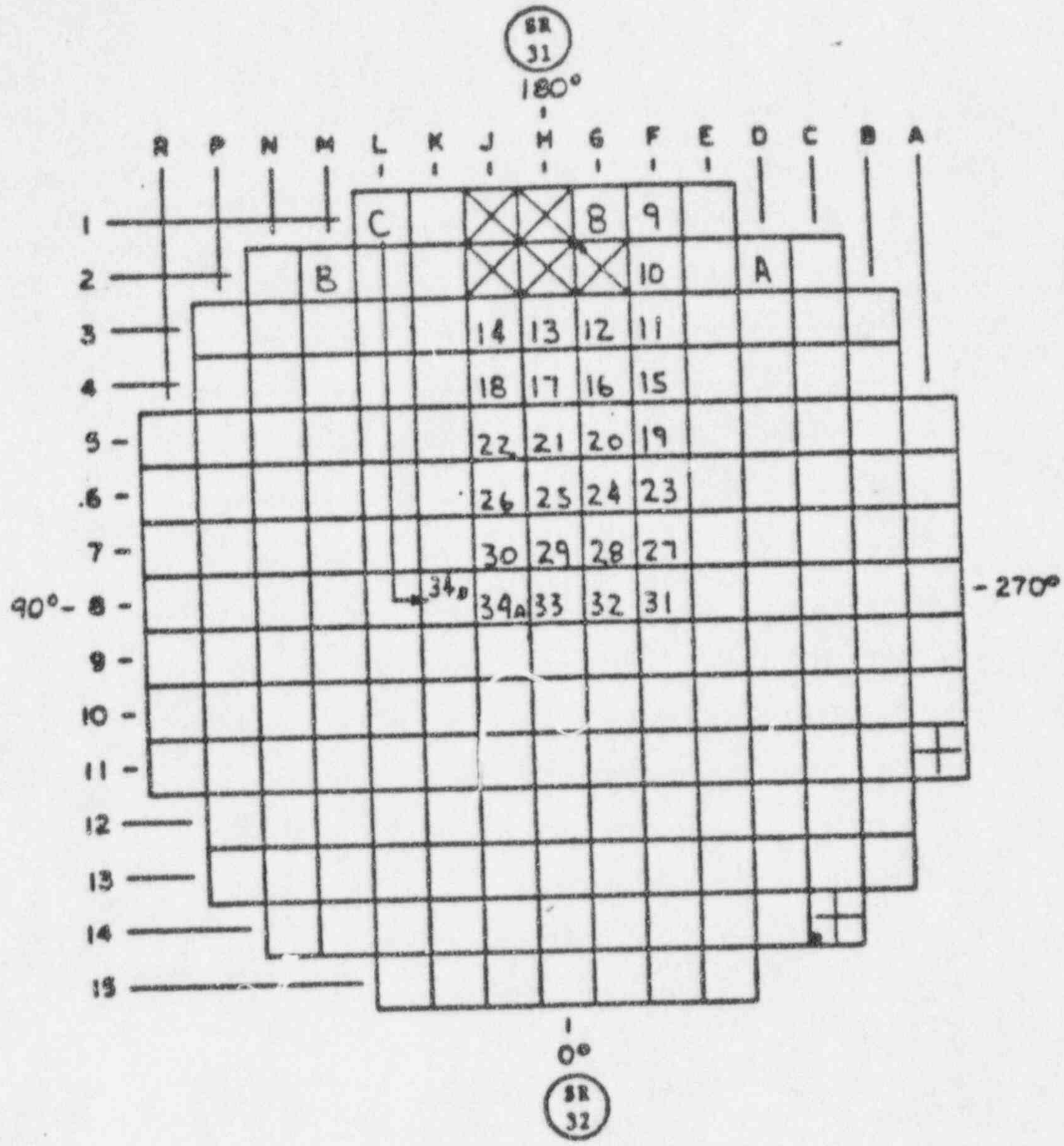


NOTE: See Figure 4.0-10 for the Figure Legend

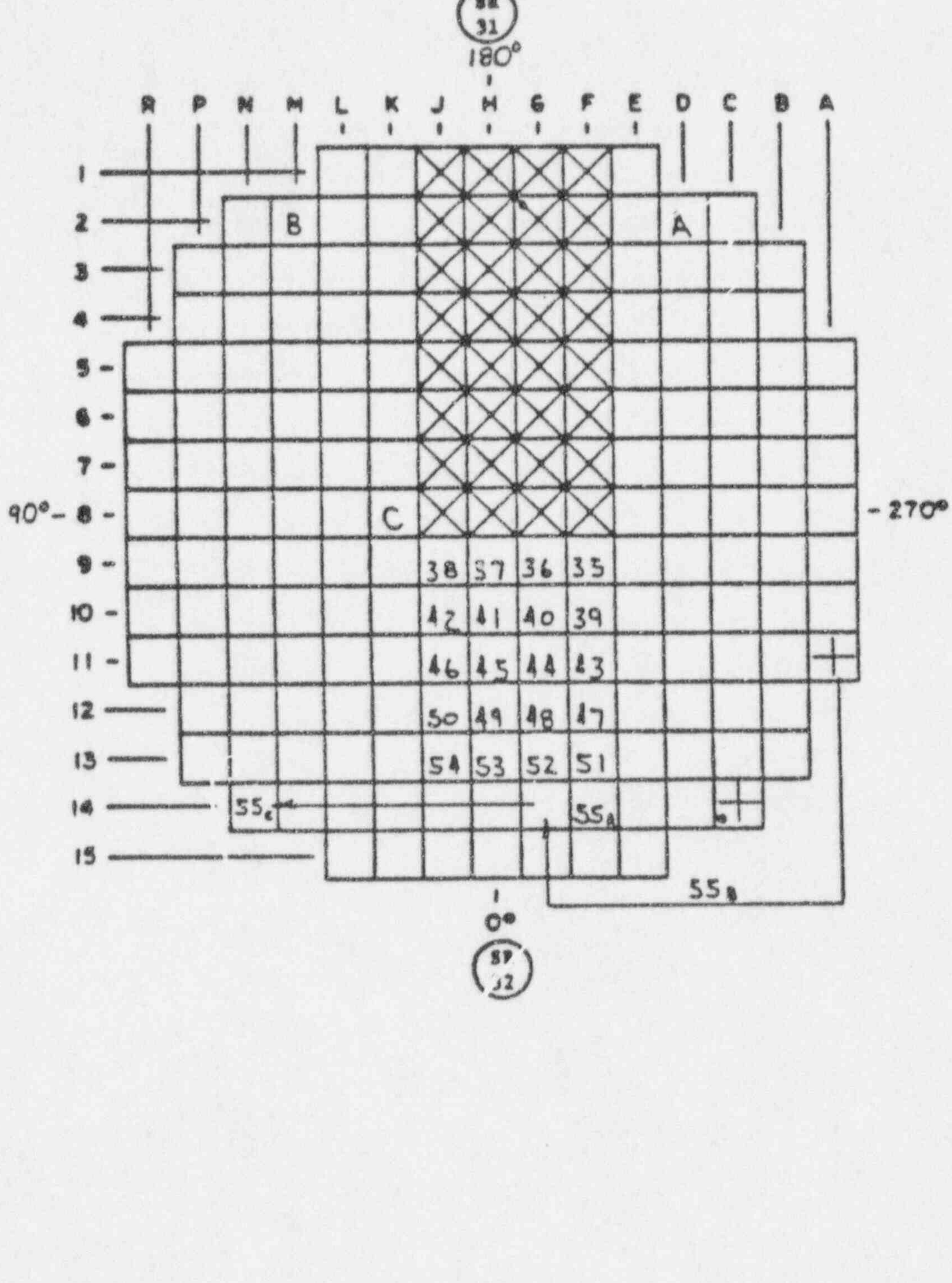
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 1 TO 7B	Figure 4.0-1



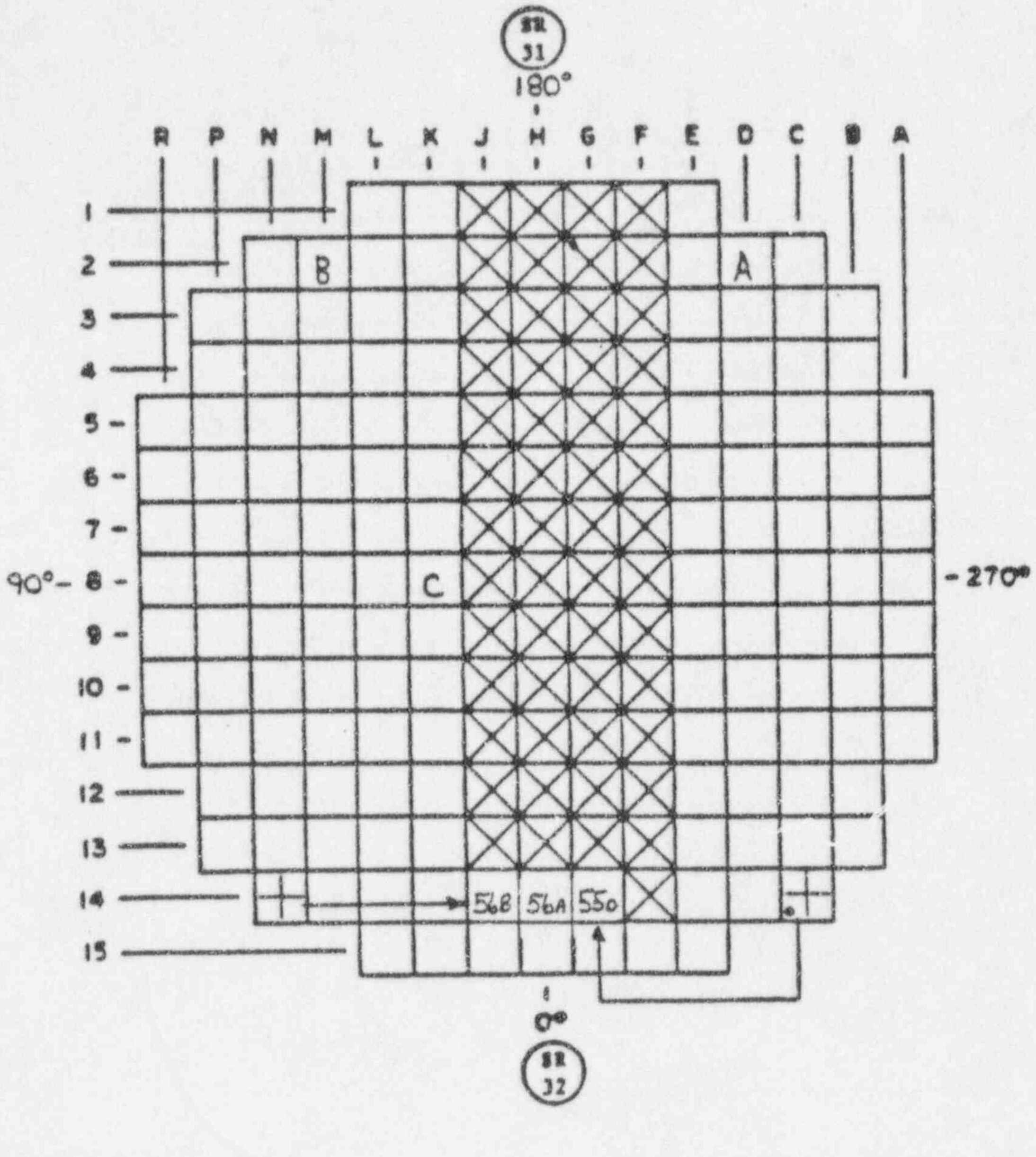
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 7C TO 7D	Figure 4.0-2



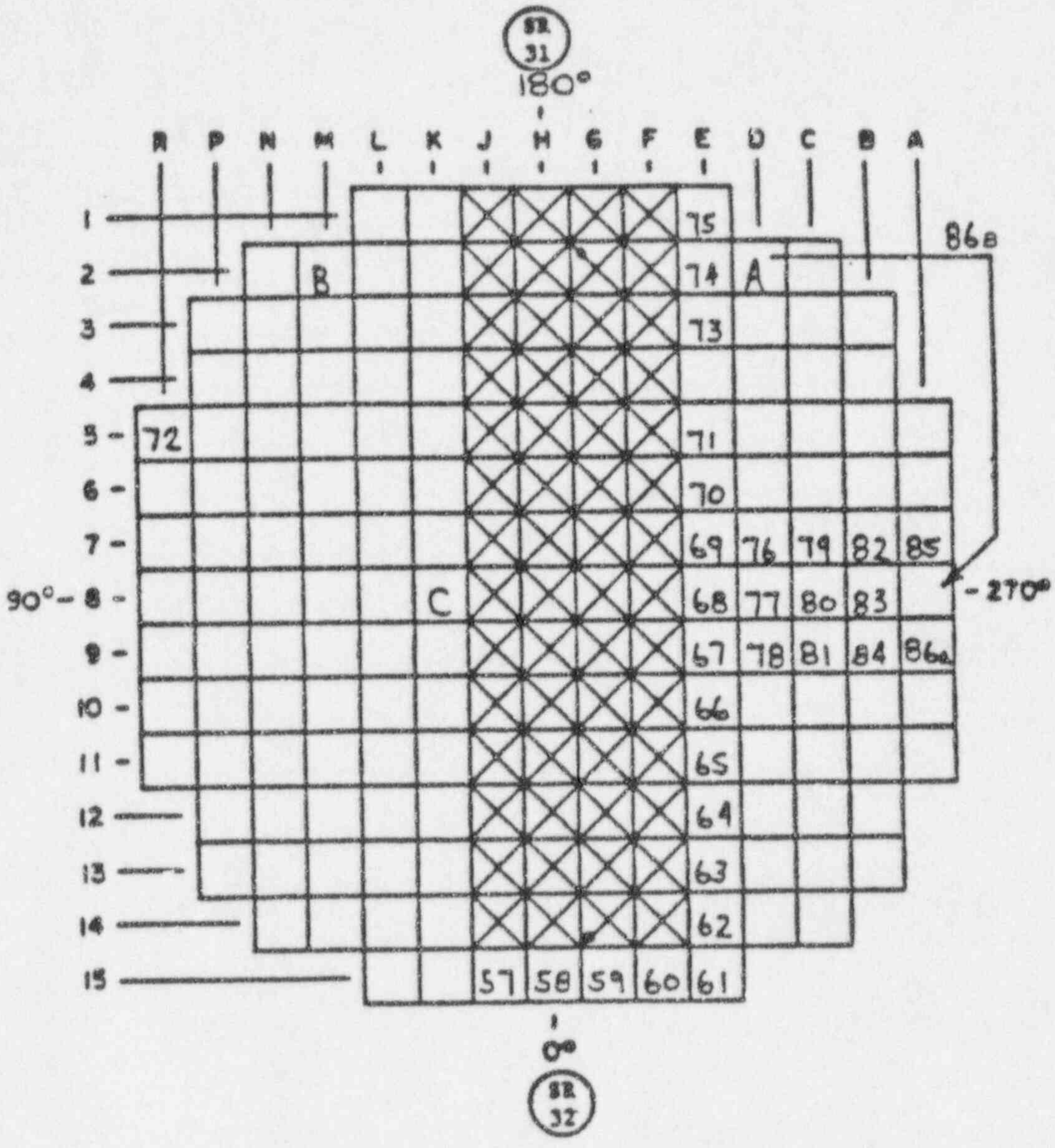
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 8 TO 34B	Figure 4.0-3



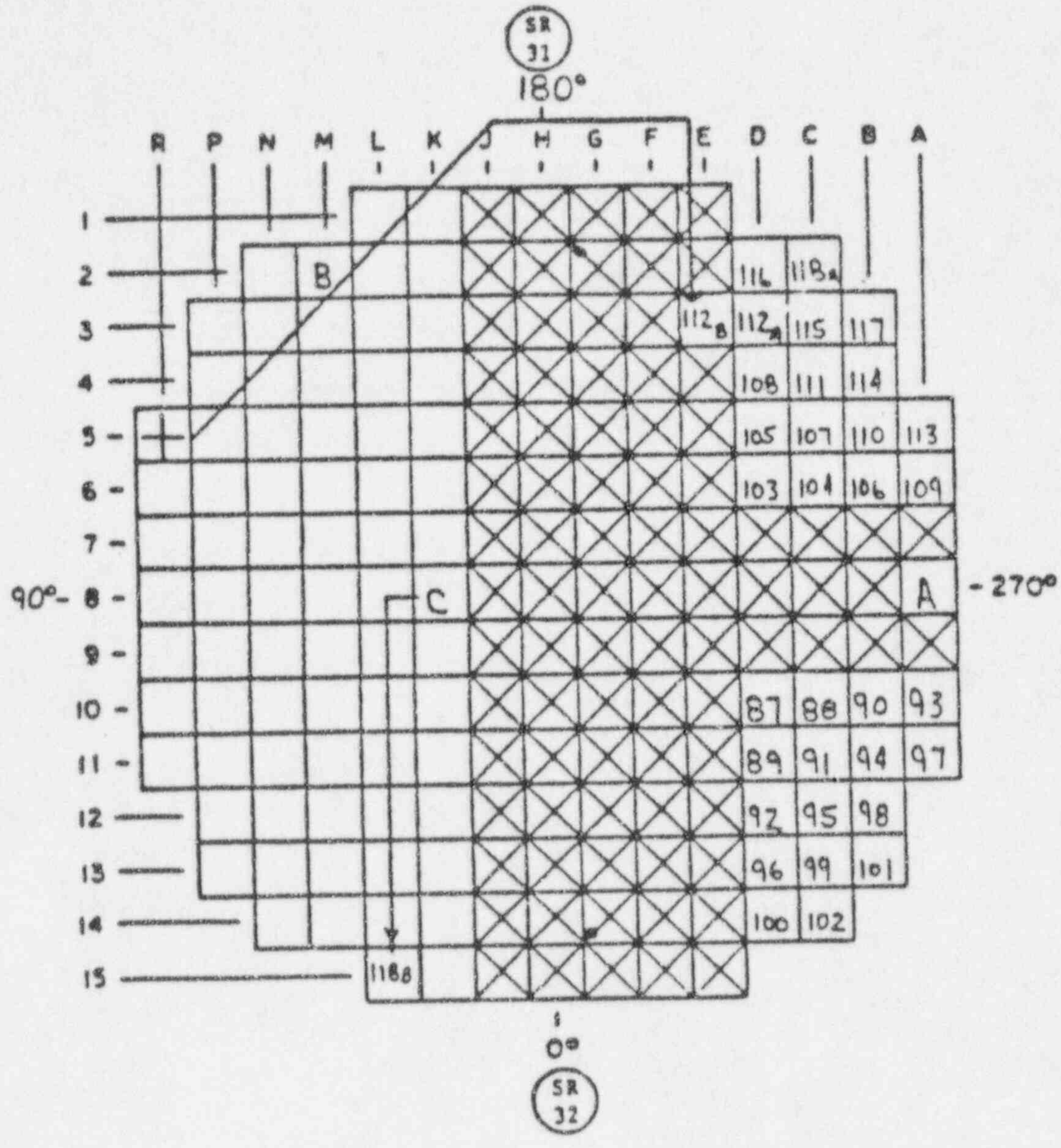
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 35 TO 55C	Figure 4.0-4



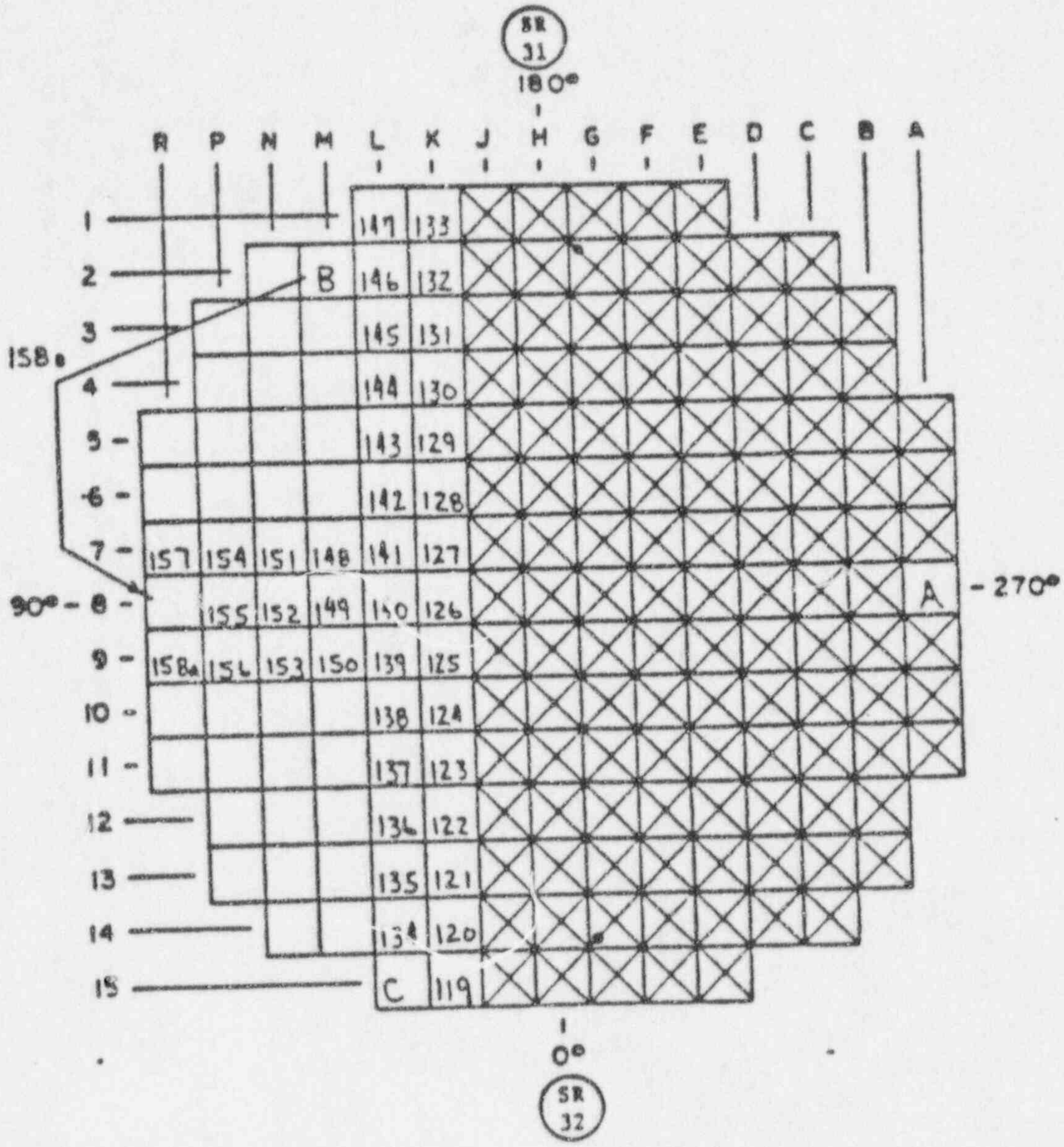
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 55D TO 56B	Figure 4.0-5



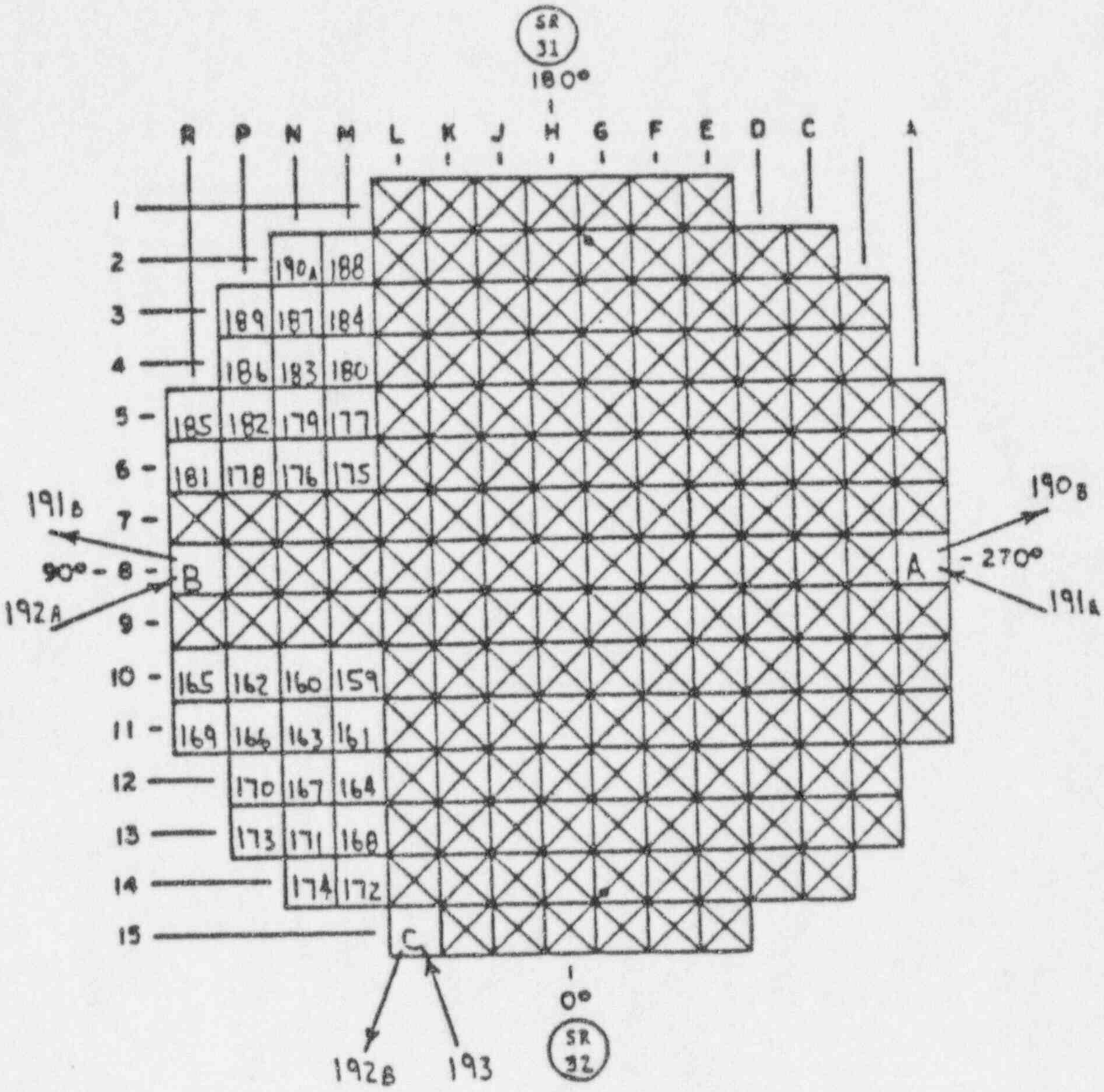
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 57 TO 86B	Figure 4.0-6



Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 87 TO 118B	Figure 4.0-7



Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 119 TO 158B	Figure 4.0-8



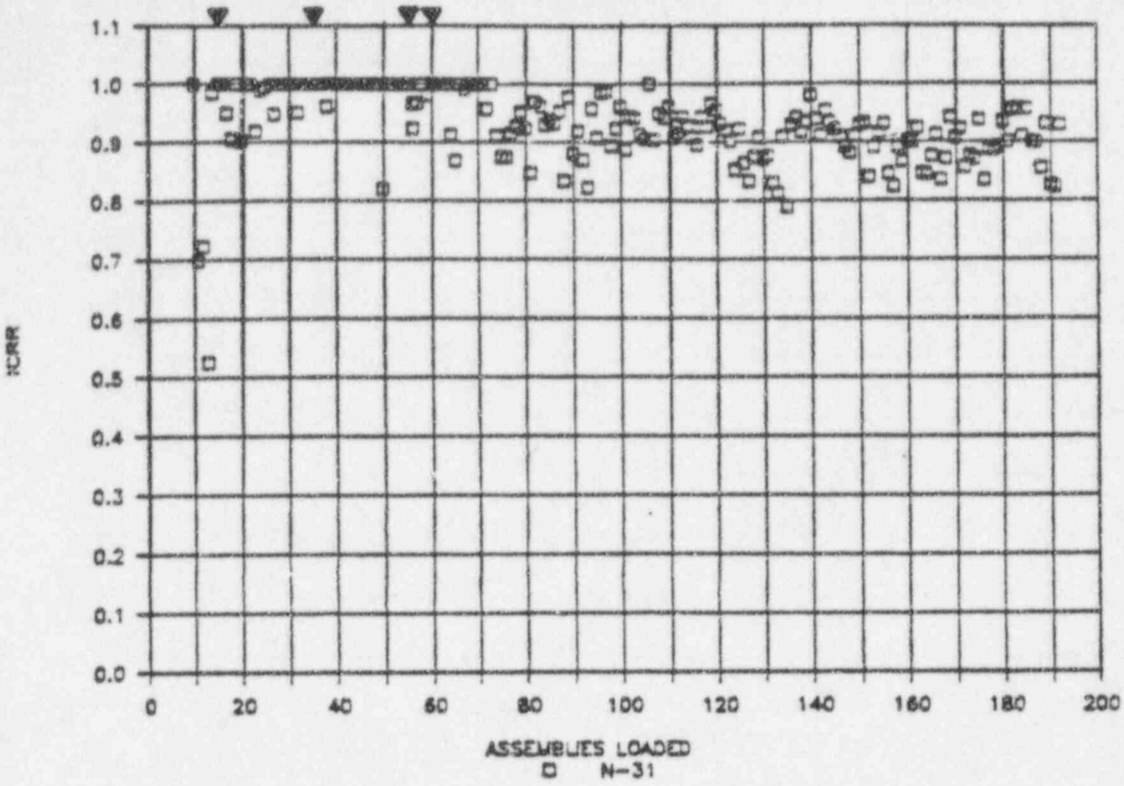
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE STEPS 159 TO 193	Figure 4.0-9

	Assembly loaded in permanent position in previous step.
	Assembly loaded in temporary position in previous step.
N	Assembly loaded into position during loading step number N.
A	Location of Temporary Detector A (B and C).
	Assembly with primary source insert.
	Not as yet loaded.

Note: Arrows indicate detector or fuel movement.

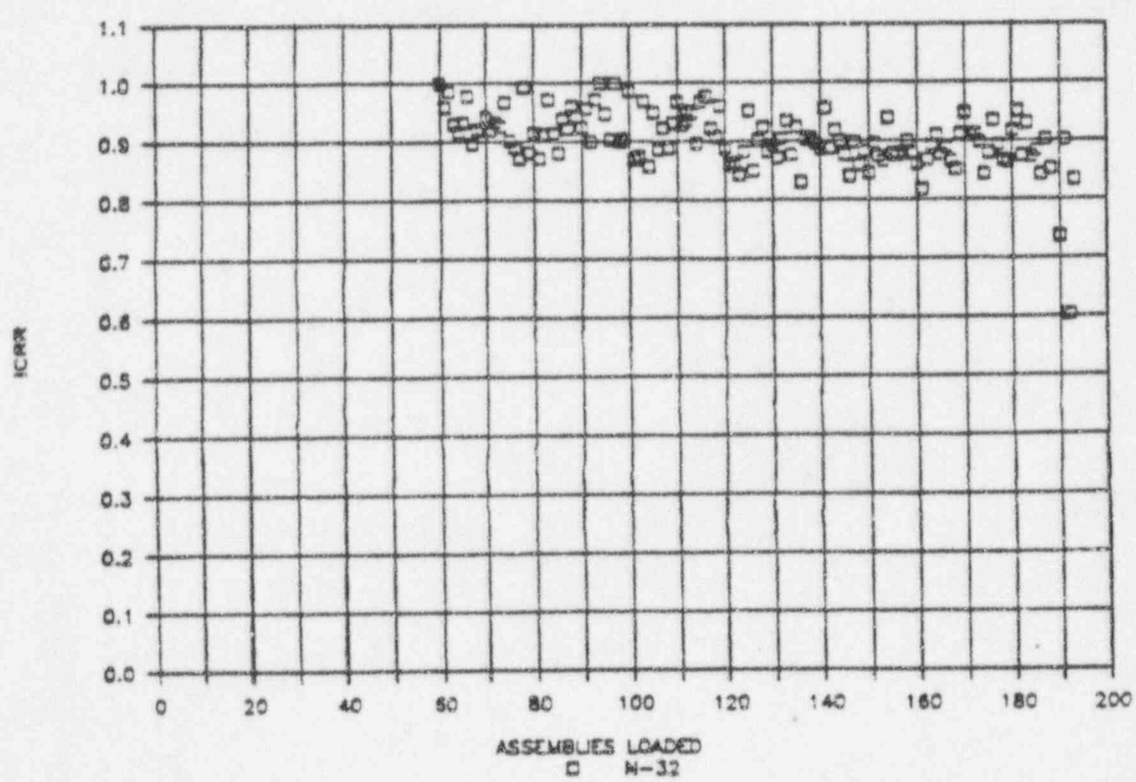
Millstone Nuclear Power Station Unit No. 3	INITIAL CORE LOADING SEQUENCE FIGURE LEGEND	Figure 4.0-10

Source Range N-31

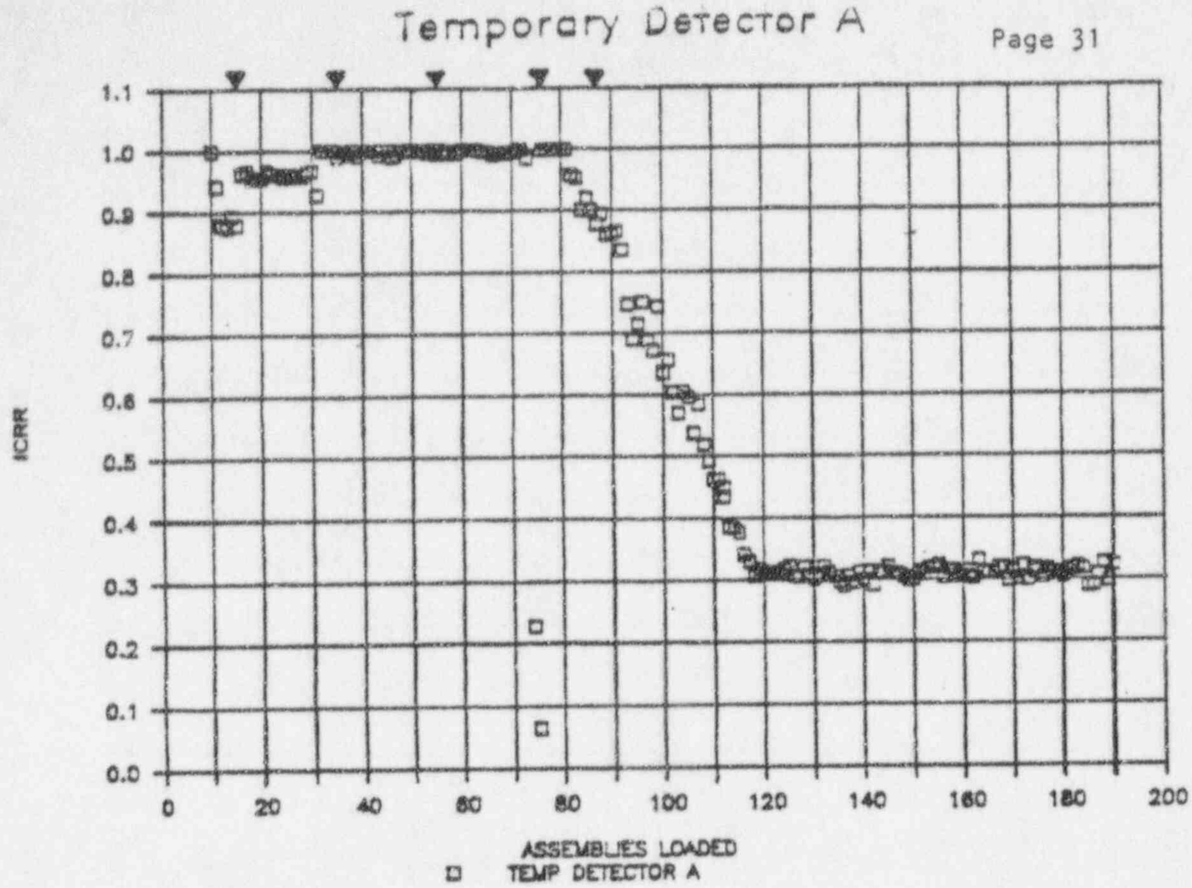


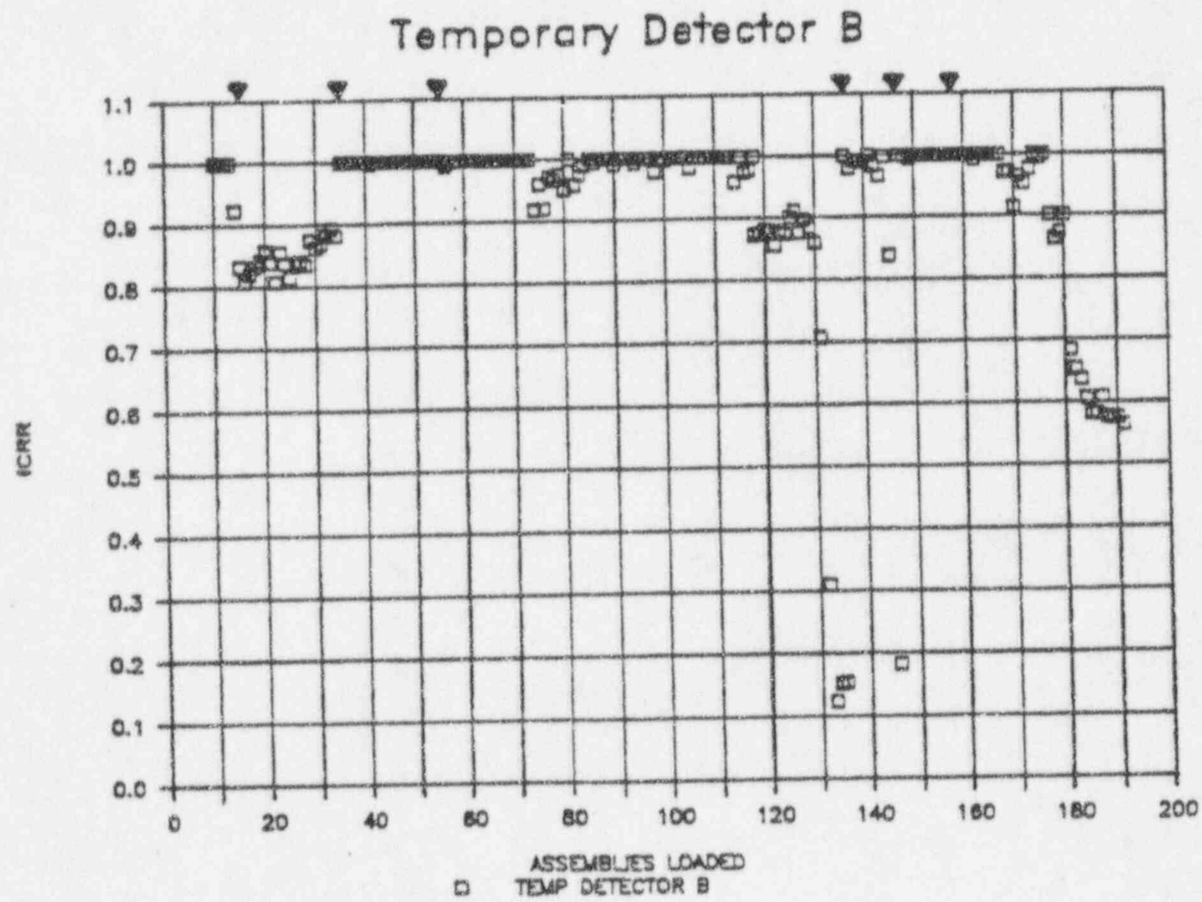
▼ New Reference Counts Taken

Source Range N-32



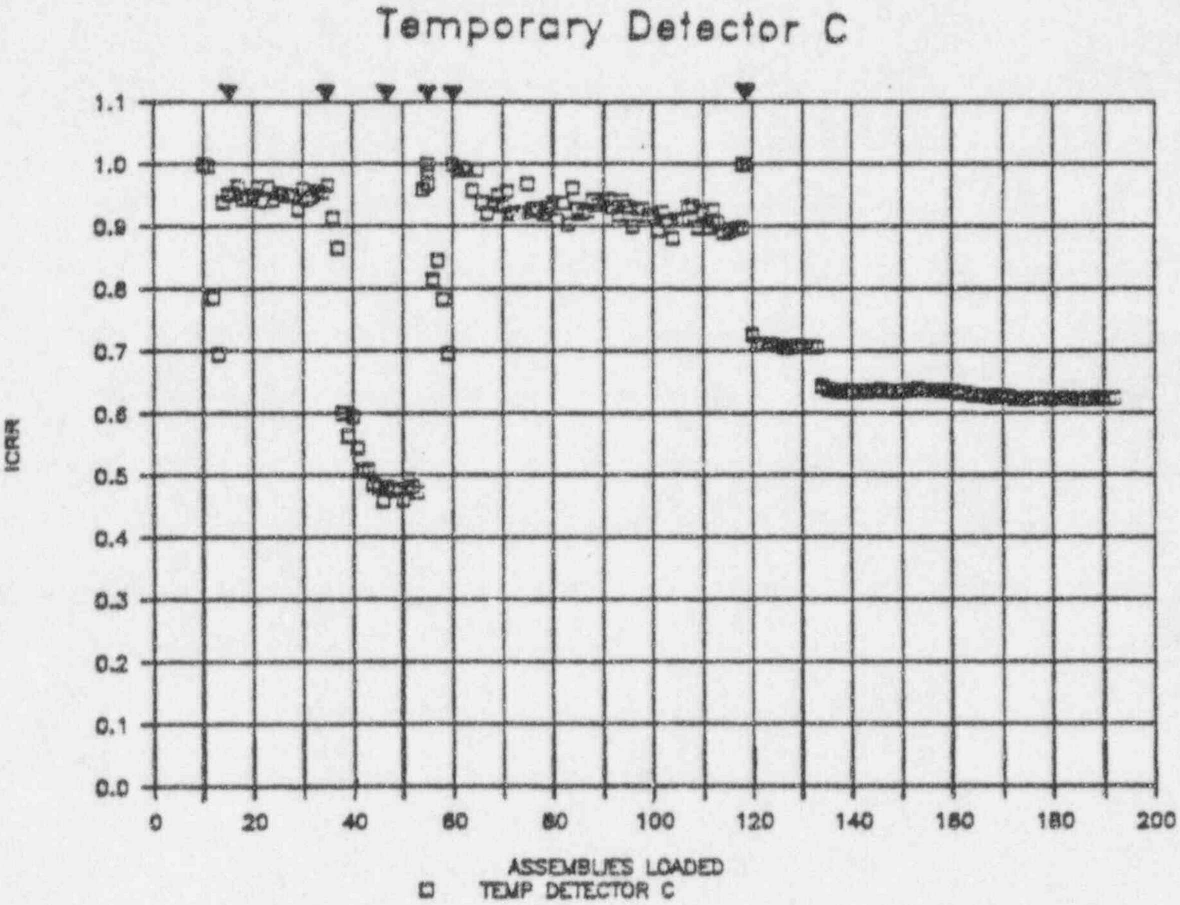
Millstone Nuclear Power Station Unit No. 3	SOURCE RANGE DETECTOR RESPONSE INITIAL CORE LOADING	Figure 4.0-11





▼ New Reference Counts Taken

Millstone Nuclear Power Station Unit No. 3	TEMPORARY DETECTOR RESPONSE INITIAL CORE LOADING	Figure 4.0-12 Page 1



▼ New Reference Counts Taken

Millstone Nuclear Power Station Unit No. 3	TEMPORARY DETECTOR RESPONSE INITIAL CORE LOADING	Figure 4.0-12 Page 2

4.1 INITIAL FUEL LOAD CHRONOLOGY

DATE	TIME	EVENT
11-26-85	1600	All Initial Conditions for fuel load met - core loading instrument alignment checks performed.
	1825	Primary source bearing assembly C04 loaded into core location L15.
	2200	Four fuel assemblies loaded.
11-27-85	0140	Operations personnel find bolt lying on control rod retainer plate in SIGMA mast. Fuel loading suspended.
	0245	Bolt removed by Operations personnel SIGMA machine inspected - two empty bolt holes found on mounting plate above SIGMA mast.
	0400	Visual scan of core and refueling cavity performed. No debris found.
	0730	Fuel load recommenced.
	1555	SIGMA machine inoperable. SFPBH inoperable.
	2300	Begin count rate data acquisition to verify detector performance (anticipating delay in fuel loading of greater than 8 hours).
11-28-85	0510	Recommenced fuel load.
	1525	I&C personnel working on SIGMA.
11-29-85	0605	I&C personnel working on SIGMA.
	1955	SIGMA now operable.
	2128	Seventy-one fuel assemblies loaded.
11-29-85	2200	Assembly B49 could not be lowered into core location E04 - adjacent assembly is bowed.
11-30-85	0240	Loading sequence modified to box in location E04 per Westinghouse recommendation.
	0300	Fuel load recommenced.
12-01-85	0155	113 assemblies loaded.
12-02-85	0100	145 assemblies loaded.
	0729	157 assemblies loaded.
	2310	193 assemblies loaded - fuel load complete.
12-04-85	2200	Core map complete.

5.0 POST CORE HOT FUNCTIONAL TEST

The major objectives of this test were to ensure all necessary plant systems were operable, Operations personnel were familiarized with the integrated operation of the plant, the RCS functioned properly with the core installed and that the initial conditions for initial criticality were met. The test procedure took the plant from a cold shutdown condition to a hot standby condition of 557°F and 2250 psia. Testing was conducted at various predetermined temperature plateaus.

Major testing conducted during this milestone involved:

- RCS loop RTD to incore thermocouple cross-calibration
- Functional verification of the RCS leak detection computer program and surveillance procedure
- Proper operation of the rod control slave cycler and CRDM operation with rods attached was verified
- Rod drop times were measured under cold no-flow, cold full-flow, and hot full-flow conditions
- Proper pressurizer spray and heater operation was verified
- Proper operation of the flux mapping and rod position indication systems was verified
- The RCS flow and RTD bypass flow were verified to be acceptable
- RCS flow coastdown timing following a trip of a single RCP and the simultaneous trip of all four RCPs was measured and compared to the FSAR assumed values
- Extensive operational testing of the CVCS system was conducted
- Proper operation of the RCS loop stop valve and RCP interlocks was verified

Testing was conducted over the period from 12-13-85 to 01-23-86.

5.1 SHUTDOWN MARGIN

3-INT-5000, Appendix 5001

OBJECTIVE

The objective of this test was to ensure that the core remains subcritical and that the Technical Specification Shutdown Margin (SDM) requirements are met throughout Post Core Hot Functional (PCHF) testing.

DISCUSSION

Based on information from the Westinghouse Nuclear Design Report, a RCS Boron concentration of $\geq$ 1850 ppm was determined to maintain adequate SDM in Modes 3, 4, 5 regardless of rod position and RCS Tavg. The following data was recorded at 24 hour intervals during PCHF testing: RCS boron concentration, pressurizer boron concentration, Tavg, reactor coolant pump status, residual heat removal system status and control rod position.

RESULTS

Adequate SDM was maintained throughout PCHF. RCS boron concentration was verified each day to be greater than 1850 ppm (average = 2054 ppm). Pressurizer boron concentration was verified to be within ± 50 ppm of the RCS while the RCS was in a cold condition. However, when the RCS heatup began, the pressurizer boron samples became unreliable. Investigation revealed that the loop seal drain line for the pressurizer safety valves was connected to the pressurizer liquid sample line. With the RCS heated, condensate from the pressurizer vapor space accumulated in the loop seals and diluted the pressurizer liquid samples. Plant deficiency DDR 996 covers this issue. While not affecting the ability to operate the plant safely, this situation represents an inconvenience. Engineering is investigating possible solutions to the problem.

5.2 INCORE THERMOCOUPLE/RTD TESTING
3-INT-5000, Appendix 5002

OBJECTIVE

The objectives of this test were to:

1. Perform a functional check and obtain cross-calibration data for core exit thermocouples and reactor coolant RTDs.
2. Verify expected resistance versus temperature characteristics of reactor coolant RTDs.
3. Verify expected millivolt versus temperature characteristics for core exit thermocouples.
4. Verify temperature and pressure of the Inadequate Core Cooling System (ICCS) at each temperature plateau.
5. Obtain data for preparation of the RTD calibration report.

DISCUSSION

The test was conducted on 01-15-86 and 01-16-86 during the heatup of the plant. Data was collected from the incore thermocouples and RCS RTDs during four periods of constant RCS heatup instead of the traditional method where data is collected during four periods of isothermal RCS conditions. The constant heatup rate method greatly increased testing flexibility and reduced the amount of time required for the test.

During each of the data collection periods, a constant rate of RCS heatup was achieved by first placing steam generator levels in the normal operating band with all generator levels approximately equal. Feedwater flow and blowdown were secured 30 minutes prior to collecting the data. Data collection began when a constant heatup rate was achieved. Data was collected in the RCS temperature bands of 355-365°F, 415-425°F, 480-490°F and 530-550°F.

Incore thermocouple temperature data was obtained by initiating a plant process computer printout at the beginning of the collection period. The incore temperature data was from the Inadequate Core Cooling System (ICCS). Data from the RCS RTDs was obtained from the RTD inputs to the Westinghouse 7300 process control system. Additional measurements of signal and compensating lead resistances were made for the three-wire RCS wide range hot leg RTDs so that the actual RTD resistance could be determined. After each RTD in the loop under test was measured, the procedure was repeated for the remaining loops. Four sets of data from each loop were collected during each temperature band.

RCS wide range pressure was obtained from the ICCS computer via the plant process computer, and appeared on the printout of incore thermocouple temperatures. RCS narrow range pressure was obtained from the control room main control board indicators.

RESULTS

The incore thermocouple to RTD cross-calibration acceptance criteria was achieved in that the incore thermocouple temperatures were within 2°F of each other, and within 2°F of the RTD cross-calibration results. The acceptance criteria for RCS and ICCS pressure indication was also satisfactorily met in that the RCS wide range and narrow range pressures were within 40 psia of each other.

The RTD data was supplied to Westinghouse for evaluation and preparation of the RTD calibration report.

5.3 ROD CONTROL SLAVE CYCLER AND CRDM TIMING TEST
3-INT-5000, Appendix 5004

OBJECTIVE

Under cold shutdown conditions, provide verification of proper slave cycler timing and Control Rod Drive Mechanism (CRDM) timing, and an operational check of each CRDM with a Rod Cluster Control Assembly (RCCA) attached.

DISCUSSION

The test was performed from 12-15-85 to 12-27-85 under a Cold Shutdown (Mode 5) condition.

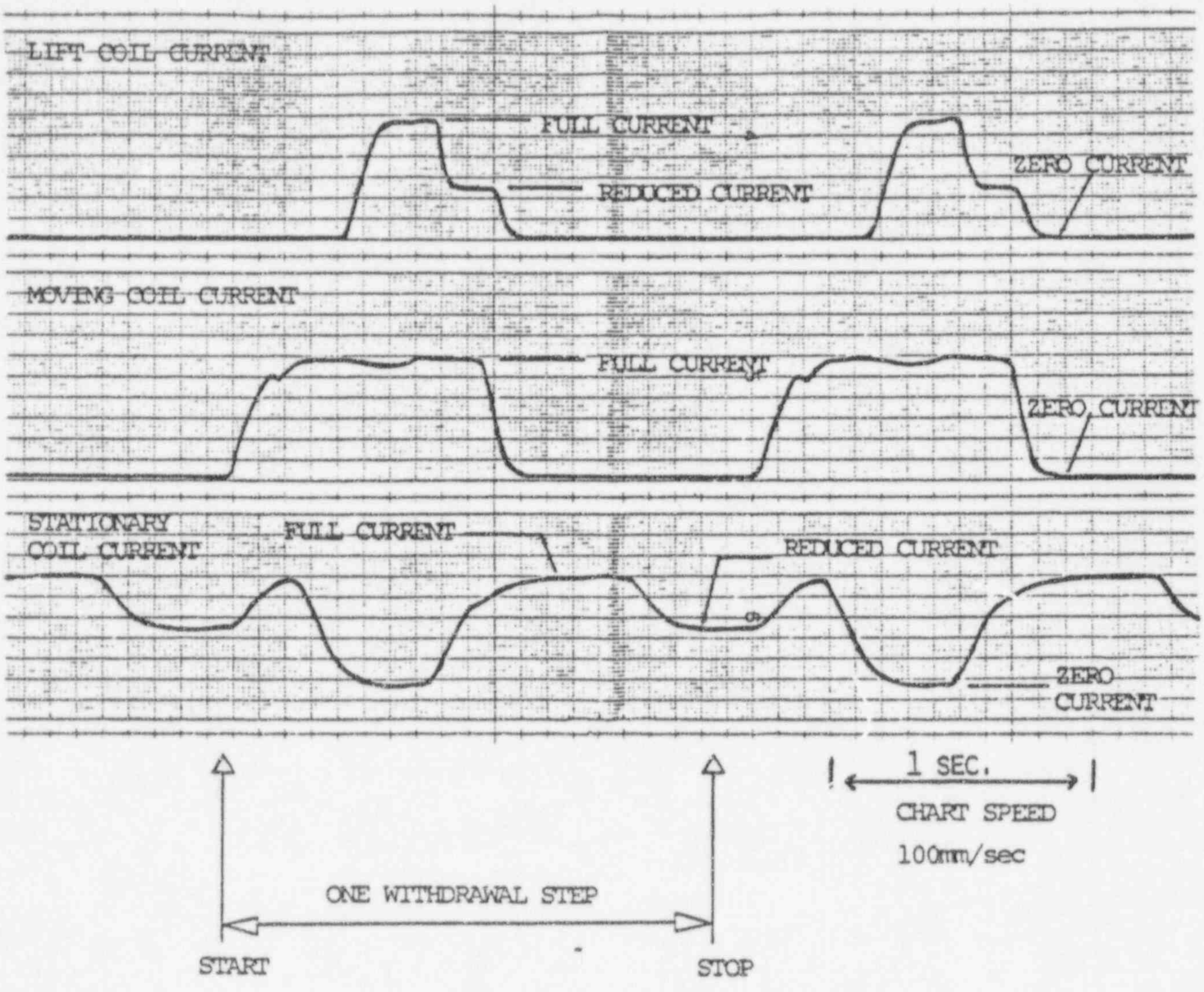
Proper slave cycler timing was verified by, in turn, selecting one rod from each rod control power cabinet and monitoring the CRDM lift coil, stationary coil, and moving coil currents, and the CRDM microphone output, while moving the rod from zero to 48 steps and then back to zero. All other rods in the group under test were prevented from moving by opening the appropriate lift coil disconnect switches. Proper slave cycler timing was verified by comparing the CRDM coil current oscillograph traces with examples provided in the Westinghouse CRDM technical manual.

The operational check of each CRDM was accomplished by, in turn, withdrawing each shutdown and control bank to 48 steps, disabling all rods in the group except the one under test, and then alternately withdrawing and inserting the rod under test 10 steps while obtaining oscillograph traces of the lift, stationary, and moving coil currents. This process was repeated twice for each rod, and the resulting oscillograph traces were compared for timing to each other and to examples provided in Westinghouse CRDM Technical Manual.

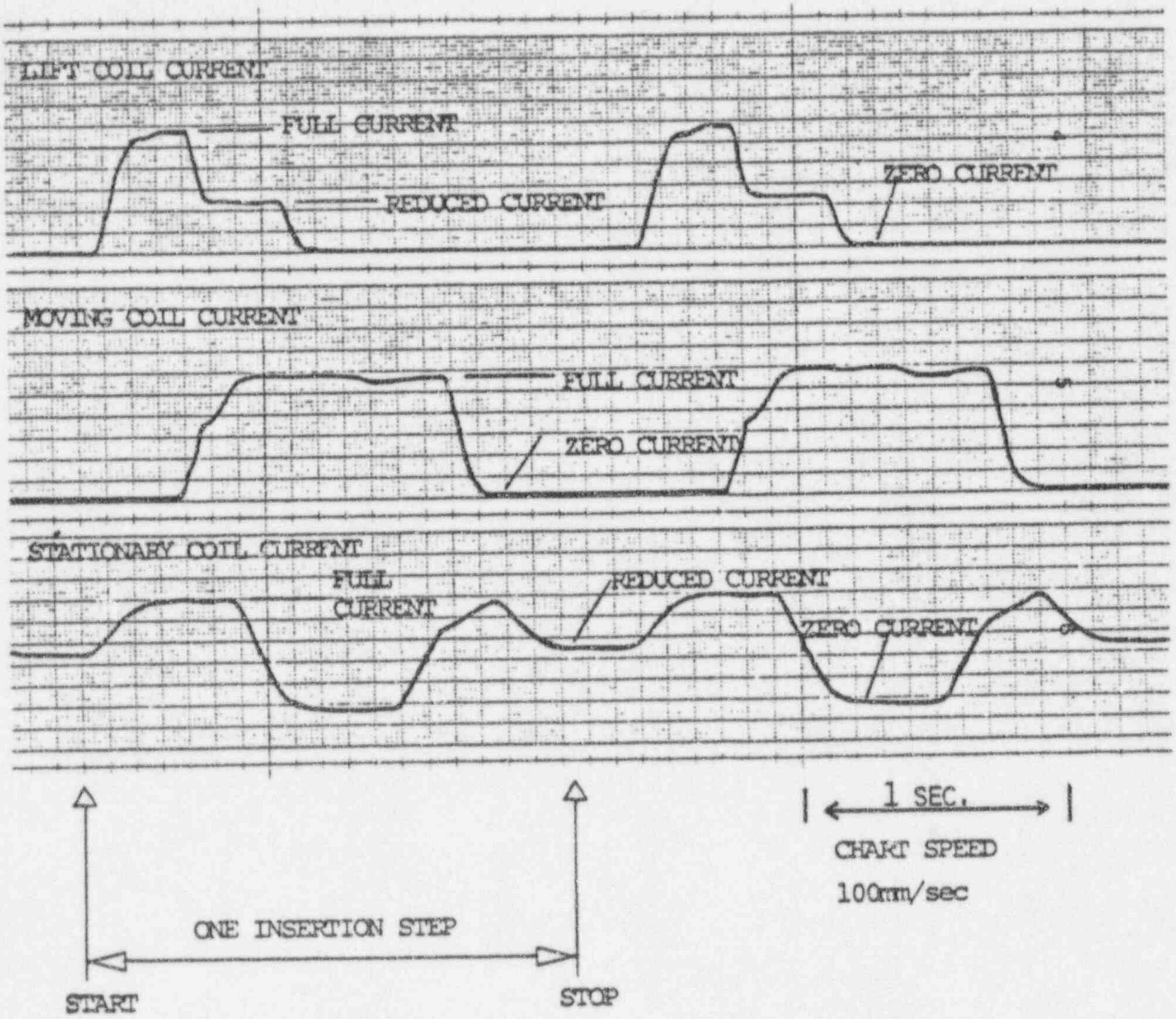
Figure 5.3-1 shows a typical oscillograph trace of lift, moving, and stationary coil currents during rod withdrawal operation. Figure 5.3-2 shows the same during an insertion operation.

RESULTS

Proper slave cycler timing and CRDM timing were verified by comparing lift, moving, and stationary coil current oscillograph traces to examples provided in the Westinghouse CRDM Technical Manual. All comparisons indicated satisfactory equipment performance.



Millstone Nuclear Power Station Unit No. 3	TYPICAL CRDM OSCILLOGRAPH TRACE ROD WITHDRAWAL	Figure 5.3-1



Millstone Nuclear Power Station Unit No. 3	TYPICAL CRDM OSCILLOGRAPH TRACE ROD INSERTION	Figure 5.3-2

5.4 RCS LEAK RATE

3-INT-5000, Appendix 5006

OBJECTIVE

This test performed two functions:

1) It reverified that the plant's computer Leakage Calculation Program, SP 3J3, could detect a 1 gallon per minute (GPM) UNIDENTIFIED LEAK from the Reactor Coolant System (RCS) and connected portions of the Chemical and Volume Control System (CVCS).
2) In parallel, it validated the manual RCS Leak Test Surveillance Procedure, SP 3601F.6.

DISCUSSION

This test was essentially a repeat of the Pre-Core Hot Functional Leak Rate Test, 3-INT-3000, Appendix 3030.

This test was performed on 01-22-86 with the plant stable at normal operating temperature and pressure (557°F and 2250 PSIA). The boundary of the test included the entire RCS, those portions of the CVCS that delivered letdown to the Volume Control Tank and returned it to the RCS, and to the first isolation valve of all systems connected to the CVCS and the RCS. No changes were made to any of the valve lineups associated with the RCS or CVCS during the leak rate test. All normal means of removing or adding water to the RCS and CVCS were secured and then a mass balance was performed using the change in pressurizer and primary tank levels. These volume changes were individually adjusted for any change in temperature over the test period.

The test's initial conditions required extensive system lineup verifications. Once these were complete and the plant was verified in a stable condition, a 4 hour manual mass balance calculation was performed concurrently with both the computer program, SP 3J3, and the surveillance procedure SP 3601F.6. This 4 hour test run was to obtain baseline information on the stable plant leak rate and to document in Appendix 5006 that the plant met the Technical Specifications of no greater than 1 GPM UNIDENTIFIED LEAKAGE from the RCS (TS 3.4.6.2.b) and no greater than 10 GPM IDENTIFIED LEAKAGE from the RCS (TS 3.4.6.2.d).

The 4 hour mass balance portion of the test was successfully completed with the following data being obtained:

1) IDENTIFIED LEAK RATE = 0.74 GPM
2) UNIDENTIFIED LEAK RATE = 0.73 GPM

Upon completion of the 4 hour test run, a 1 GPM known leak was induced off the low pressure section of the CVCS letdown line. The failed fuel radiation monitor drainline was chosen for the source of the leak so as to allow the use of permanently installed flow detector (3CHS-FI391) to monitor the induced leak.

After stabilizing the 1 GPM known leak (actual reading on 3CHS-FI391 varied between 0.98 GPM and 1.17 GPM), a 2 hour mass balance calculation was performed, again, concurrently with both the Computer Program SP 3J3, and the Surveillance Procedure, SP 3601F.6. The data from the 2 hour test run and the change in relation to the 4 hour test run was compared to the following acceptance criteria for both the computer program and the surveillance procedure.

1) No greater than 10 GPM IDENTIFIED LEAKAGE from the RCS (TS 3.4.6.2.d).

2) The change in the UNIDENTIFIED LEAKAGE shall be one 1 GPM ±9 percent (0.91 to 1.09 GPM).

The outputs of the leak rate tests were recorded as follows:

	SP 3J3	SP 3601F.6
IDENTIFIED LEAK RATE	0.61 GPM	0.654 GPM
Change in UNIDENTIFIED LEAK RATE	1.263 GPM	1.263 GPM

The leak rate change gave a conservative output since it actually indicated slightly more leakage than was present. However, the change in UNIDENTIFIED LEAK RATE did not meet the acceptance criteria of 0.91 to 1.09 GPM. To document this, plant deficiency UNS 7495 was submitted.

RESULTS

Performance and evaluation of test results for the RCS Leakage Program, SP 3J3, showed generally satisfactory performance. Although programmed-calculated leakage was higher than that for the hand-calculation, identified in plant deficiency UNS 7495, this anomaly is explainable by a varying induced leakage flow (0.98 GPM to 1.17 GPM). The deficiency recommended to accept-as-is, in part, due to the conservative results of the test, i.e., indicating more leakage flow (1.263 GPM) than was actually present (acceptance criteria 0.91 to 1.09 GPM). The proposed disposition of UNS 7495 was approved by the Joint Test Group with the added requirement that it be sent to the Unit 3 Reactor Engineer for review. The subsequent review by the Reactor Engineer determined the installed leak detection program to be satisfactory.

5.5 PRESSURIZER HEATERS AND SPRAY TESTING
3-INT-5000, Appendix 5007

OBJECTIVE

The objectives of this test were to:

1. Establish optimum pressurizer spray valve bypass valve position in order to maintain the spray lines in a warmed condition (to minimize thermal shock on the lines when pressurizer spray is initiated) and at the same time maintain bypass flow so that proportional heater output is kept at approximately 50 percent of rated capacity. Once the final position for the bypass valves have been set, the spray line low temperature alarms will be set. It should be noted that a preliminary setting of the bypass valves was completed during the precore hot functional test (3-INT-3000, Appendix 3011).
2. Verify pressurizer spray effectiveness is within design tolerances.
3. Verify pressurizer heater effectiveness is within design tolerances.
4. Verify pressurizer heater capacity is within design tolerances.

DISCUSSION

The test was conducted between 01-20-86 and 01-21-86 with the plant in a Hot Standby (Mode 3) condition.

The first objective was to be accomplished by recording pressurizer spray line temperatures while incrementally opening the spray valve bypass valve. This data would then be plotted and the optimum position of the bypass valves selected. The optimum positions correspond to the point on the curves where spray line temperature flattens out. The spray valve bypass valves would then be set to these optimum throttle positions and plant conditions maintained at steady state so that equilibrium data could be taken on the pressurizer spray lines.

The purpose of this data is to confirm that the spray line temperature is at ≥ 540°F and the proportional heaters are at approximately 50 percent of rated capacity. Adjustments to the valve position would be made as required to achieve these desired conditions. Once final bypass valve positions were established, spray line low temperature alarm setpoints would be established and reset if required. These setpoints were required to be ≥ 530°F so as to conform to the Westinghouse Precautions, Limitations and Setpoints (PLS) Document.

The second objective of the test was accomplished by establishing normal no-load operating temperature and pressure in the RCS with the charging system flow controller in manual and all pressurizer backup and control heaters off. Once these conditions were established, both pressurizer spray valves were fully opened and kept open until RCS pressure was reduced to approximately 2000 psia.

The third and fourth objectives were accomplished by reestablishing normal no-load RCS temperature, pressure and pressurizer level with both pressurizer Power Operated Relief Valves (PORVs) in the closed position, the charging system flow controller in manual and both pressurizer spray valve controllers in manual with the spray valves closed. At that point, all pressurizer backup and control heaters were energized manually to full output and RCS pressure monitored until it reached approximately 2300 psia. Once this pressure was reached, all pressurizer heaters were returned to automatic as well as the charging system and pressurizer spray valve controllers. Concurrent with this transient, 3-phase voltages and currents were taken on all pressurizer heater groups to verify that they were within design specification.

RESULTS

The setting of the pressurizer spray valve bypass valve positions could not be performed as initially proposed in the test procedure due to excessive pressurizer spray valve seat leakage. A test change was written to first monitor proportional heater output and pressurizer spray line temperature with the spray valves shut (as indicated on the main control board) and then secure instrument air to the valves (the valves are fail-closed in design) to determine if the valves were being maintained partially opened due to improper control signals. Results of this test change indicated the valves were in fact fully closed. However, the seat leakage past these valves with the bypass valves open 1/16 of a turn was such that the proportional heaters were operating at 100 percent of rated capacity. As a next, step the bypass valves were fully closed to determine if the leakage past the spray valves was sufficient to maintain the pressurizer spray line temperature above the low temperature alarm setpoint of 530°F. With the bypass valves shut, the flow was not sufficient and the low temperature alarm was received. The bypass valves were then opened approximately 1/16 of a turn. This resulted in spray line temperatures of 539°F for loop 1 and 543°F for loop 2 with the proportional heaters operating at approximately 80 percent capacity. A unit deficiency, UNS 7485, was written to document the inability to generate the required spray line temperature versus bypass valve position curves and the excessive proportional heater output. The deficiency was reviewed by Engineering and Westinghouse and dispositioned accept-as-is. The spray line temperature alarms setpoints were left at their initial settings of 530°F. This was due to spray line equilibrium temperatures being approximately 10°F higher than the setpoints and the requirement not to lower the setpoints below the 530°F Westinghouse Precautions, Limitations and Setpoints (PLS) Document design value.

The pressurizer spray effectiveness was successfully verified. The verification was done with the plant at a no-load temperature and pressure with the charging system flow controller in manual and all pressurizer heaters turned off. Initial pressurizer level was 26 percent. The pressurizer spray valves were then fully opened using the RCS master pressure controller. The RCS pressure was lowered from an initial value of 2240 psia to the desired endpoint of 2000 psia in 114 seconds. While this time was slightly slower than the nominal response, it was well within design tolerances and test acceptance criteria.

The pressurizer heater effectiveness was successfully verified. Two runs of the test were performed. These values were within the acceptance criteria. During both runs, the overall pressurizer heater capacity was below design specification, being 1703.7 KW versus the design range of 1710-1890 KW. In addition, the group C proportional heater capacity was 393.99 KW versus the design range of 394.25-435.75 KW; the group D heater capacity was 324.3 KW versus the design range of 328.7-363.3 KW; and the group E heater capacity was 325.5 KW versus the design range of 328.7-363.3 KW. Pressurizer heaters groups A and B (which are powered off vital buses) had capacities of 329.9 KW and 330.0 KW, respectively. These values were within the 328.7-363.3 KW acceptance criteria. Therefore, all Technical Specification requirements were met.

Plant deficiencies UNS 7489 and UNS 7496 were initiated to document the discrepancies in heater capacities. Both deficiencies were reviewed by Engineering and Westinghouse and dispositioned accept-as-is.

5.6 ROD DROP TESTING
3-INT-5000, Appendix 5008

OBJECTIVE

The objectives of the test were to:

1. Determine the drop time of each control rod with the Reactor Coolant System in a cold condition. The drop times were measured at no-flow and again at full-flow.
2. Determine the drop time of each control rod with the Reactor Coolant System at normal operating temperature. The drop times were measured at full-flow conditions.

Any rods having a drop time exceeding the acceptance criteria were required to be dropped 10 additional times. In addition, any rods having a drop time exceeding the average drop time for all rods by more than the two (two standard deviations) sigma limit were dropped three additional times.

DISCUSSION

The test was performed between 12-19-85 and 1-20-86 during Cold Shutdown (Mode 5) and Hot Standby (Mode 3) conditions. During the test, the drop time of each control rod was measured under cold no-flow, cold full-flow, and hot full-flow conditions. The acceptable rod drop time in each case was less than 2.2 seconds from the beginning of the decay of the stationary gripper coil to dashpot entry. Any rods which failed the 2.2 second acceptance criteria were required to be dropped ten additional times and any rods with drop times outside the two sigma limits were dropped three additional times.

Rod drop times were determined by simultaneously dropping all rods in a group from a fully withdrawn position (228 steps). Data from the group under test was collected using a computer based data acquisition and analysis system developed by Westinghouse exclusively for rod drop testing. Drop data for the group under test was collected from the Digital Rod Position Indication (DRPI) system. Testing progressed through each group in sequence until all rods had been dropped.

Once all data had been collected, it was analyzed to determine the drop and turnaround time of each rod, and the mean and two sigma limits. Hardcopy drop traces for each rod were provided as well as summary tables listing individual rod drop times and indicating those rods falling outside the two sigma limits. Figure 5.6-1 provides a typical rod drop trace. Table 5.6-1 summarizes the rod drop times for cold no-flow, cold full-flow, and hot full-flow conditions.

During the cold full-flow portion of the test, rods K14, J03, H06, and H10 exceeded the two sigma limits and were each dropped three additional times. As a result of these additional drops, K14 remained outside the two sigmas limit but varied only 10 msec from the initial drop. J03 was within the two sigma limit on two of the three additional drops; H06 remained outside the two sigma limits, but within 2 msec of the initial drop time; and H10 was within the two sigma limits on two of the three additional drops. The additional drop data was reviewed and determined acceptable.

During the cold, full-flow portion of the test, rods H06 and F08 were determined to be outside the two sigma limits and were each dropped three additional times. The supplemental drop times were within the two sigma limits.

During the hot, full-flow rod drop data, rods B04, M02, and L05 were determined to be outside the two sigma limits and were each dropped three additional times. The additional drops of M02 and L05 were within the two sigma limits so that only B04 remained outside the limit. This was reviewed and determined acceptable.

RESULTS

All rod drop times under cold no-flow, cold full-flow, and hot full-flow conditions were less than the 2.2 second acceptance criteria. The performance of the rods was demonstrated to be acceptable.

ROD DROP TIME TO DASHPOT ENTRY (msec)

ROD BANK	CORE POSITION	COLD NO FLOW	COLD FULL FLOW	HOT FULL FLOW
SBA	D02	1302	1500	1412
	B12	1294	1492	1402
	M14	1296	1506	1416
	P04	1294	1508	1422
	H04	1288	1514	1404
	B04	1298	1492	1274
	D14	1308	1488	1398
	P12	1298	1496	1394
	M02	1298	1494	1418
	H12	1304	1508	1408
SBB	G03	1300	1486	1396
	C09	1312	1498	1400
	J13	1290	1480	1376
	N07	1308	1500	1416
	D08	1290	1496	1410
	C07	1310	1492	1400
	G13	1290	1504	1398
	N09	1300	1498	1402
	J03	1320	1494	1398
	M08	1308	1494	1406
SBC	E03	1298	1492	1398
	C11	1294	1512	1396
	L13	1294	1502	1388
	N05	1298	1512	1398

ROD DROP TIME TO DASHPOT ENTRY (msec)
(Continued)

ROD BANK	CORE POSITION	COLD NO FLOW	COLD FULL FLOW	HOT FULL FLOW
SBD	C05	1300	1476	1394
	E13	1302	1496	1402
	N11	1302	1498	1402
	L03	1288	1494	1392
SBE	A07	1296	1486	1398
	G15	1294	1494	1402
	R09	1294	1498	1396
	J01	1298	1480	1406
CBA	H06	1320	1558	1406
	F08	1304	1542	1386
	H10	1284	1488	1364
	K08	1300	1494	1408
	E05	1296	1510	1392
	E11	1306	1508	1410
	L11	1298	1498	1400
	L05	1300	1498	1366
CBB	F02	1304	1504	1420
	B10	1302	1490	1418
	K14	1326	1506	1422
	P06	1308	1500	1410
	B06	1294	1480	1400
	F14	1290	1482	1398
	P10	1296	1496	1408
	K02	1288	1482	1386

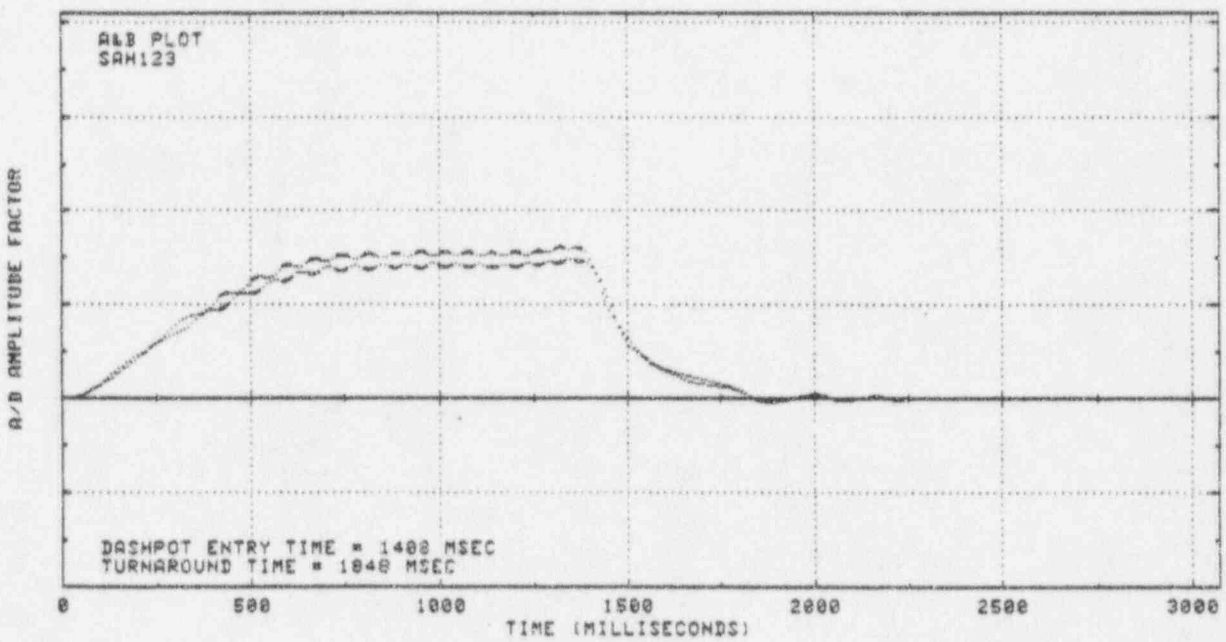
ROD DROP TIME TO DASHPOT ENTRY (msec)
(Continued)

ROD BANK	CORE POSITION	COLD NO FLOW	COLD FULL FLOW	HOT FULL FLOW
CBC	H02	1286	1488	1420
	B08	1296	1498	1400
	H14	1302	1498	1398
	P08	1304	1519	1396
	F06	1310	1490	1402
	F10	1302	1496	1404
	K10	1294	1492	1392
	K06	1298	1492	1402
CBD	D04	1296	1480	1392
	M12	1292	1482	1406
	D12	1294	1488	1382
	M04	1290	1498	1402
	H08	1296	1508	1398
MEAN DROP TIME		1299	1497	1399
MEAN MINUS 2 SIGMA		1283	1471	1361
MEAN PLUS 2 SIGMA		1315	1523	1437
RODS OUTSIDE LIMITS		K14	H06	B04
		J03	F08	
		H06		
		H10		

Acceptance Criteria: Rod Drop Time < 2200 msec

PLANT NAME: MILLSTONE III
TEST OPERATOR: D SIPPLE
REACTOR OPERATOR: P LANG
TEMPERATURE: 553 DEG F
PRESSURE: 2250 PSIG
FLOW RATE: 100 % FLOW
DATE: 01/'.7/86
TIME: 01:32

ROD #: H12 DROP #: 3



Millstone Nuclear Power Station Unit No. 3	TYPICAL ROD DROP TESTING TRACE ROD DROP TESTING	Figure 5.6-1

5.7 PRECRITICAL REACTOR COOLANT SYSTEM FLOW MEASUREMENT
3-INT-5000, Appendix 5009

OBJECTIVE

The objective of this test was to obtain the data necessary to relate reactor coolant system (RCS) installed elbow tap differential pressure (D/P) to RCS flow and to determine RCS flow.

DISCUSSION

The test was conducted on 01-18-86 with the reactor plant at steady-state conditions, temperature at approximately 557°F, pressure at approximately 2250 psia and four reactor coolant pumps running. The test consisted of collecting voltage data from the RCS flow elbow tap transmitters. From this data, the RCS flow was numerically determined. Acceptance criteria required that each loop flow be at least 90 percent of the FSAR design value of 94,600 gpm (85,140 gpm) and the total RCS flow to be at least 90 percent of the FSAR design value of 378,400 gpm (340,560 gpm).

RESULTS

All data was successfully obtained with the exception of RCS-F436 and RCS-F446 on RCS loops 3 and 4, respectively. These transmitters read abnormally low. Plant deficiency UNS 7466 was issued to document this problem. Upon evaluation, it was decided the data on the two good transmitters on each of loops 3 and 4 along with the data from loops 1 and 2 would be adequate for RCS flow determination. The RCS flow which was calculated met all acceptance criteria and is summarized on Table 5.7-1.

Subsequent to the test, corrective maintenance was performed on the two transmitters which were the subject of UNS 7466. Subsequent performance of the units has been satisfactory.

Loop 1

RCS-F414	102,087 GPM
RCS-F415	103,679 GPM
RCS-F416	102,359 GPM
Loop Average	102,708 GPM

Loop 2

RCS-F424	102,220 GPM
RCS-F425	103,520 GPM
RCS-F426	101,560 GPM
Loop Average	102,433 GPM

Loop 3

RCS-F434	102,806 GPM
RCS-F435	104,918 GPM
RCS-F436	see text
Loop Average	103,862 GPM

Loop 4

RCS-F444	101,462 GPM
RCS-F445	104,681 GPM
RCS-F446	see text
Loop Average	103,072 GPM

Total Calculated Core Flow: 412,075 GPM

Acceptance Criteria:

Calculated Loop Flow ≥ 85,140 GPM

Calculated Core Flow ≥ 340,560 GPM

Millstone Nuclear Power Station Unit No. 3	RCS FLOW DATA PRECRITICAL RCS FLOW MEASUREMENT TEST	Table 5.7-1

5.8 RTD BYPASS FLOW VERIFICATION

3-INT-5000, Appendix 5010

OBJECTIVE

The objectives of this test were to:

1. Measure the flow rate in each RTD bypass loop to verify acceptable bypass loop coolant transport time.
2. Establish the alarm setpoints for the RTD bypass flow alarm in the control room.

DISCUSSION

Prior to performing the test, the RTD bypass line as-built measurements were obtained. Based on these measurements, the minimum flow rates to obtain a 1-second bypass loop transport time were calculated.

The test was performed over the period 01-21-86 to 01-23-86. With all four reactor coolant pumps in operation and the RCS at hot zero power, no-load condition, the RTD bypass loop flow measurements were taken. The measurements were obtained by recording total RTD bypass flow in each loop with the manifold isolation valves open. After the total flows were obtained, the hot leg RTD bypass manifold isolation valves were closed and the cold leg flow was recorded. The process was then reversed in order to record hot leg flow. The individual hot leg and cold leg bypass loop flows were then compared to the minimum acceptable flow established based on bypass loop configuration. Then, using the total measured flow values for each loop, the RTD bypass loop lo flow alarm setpoints were established at 90 percent of the total flow in each RTD bypass flow manifold.

RESULTS

All acceptance criteria were met. The results of the flow measurements are presented in Table 5.8-1.

LOOP	TOTAL VOLUME (FT^3)	MINIMUM FLOWRATE (GPM)	MEASURED FLOWRATE (GPM)	CALC ALARM SETPOINT (GPM)
Hot Leg 1	0.216	105.9	117	N/A
Cold Leg 1	0.115	51.6	155	N/A
Total Loop 1	N/A	N/A	266	239.4
Hot Leg 2	0.242	108.6	118	N/A
Cold Leg 2	0.111	49.8	160	N/A
Total Loop 2	N/A	N/A	265	238.5
Hot Leg 3	0.230	103.2	115	N/A
Cold Leg 3	0.117	52.5	150	N/A
Total Loop 3	N/A	N/A	263	236.7
Hot Leg 4	0.235	105.5	118	N/A
Cold Leg 4	0.097	43.5	158	N/A
Total Loop 4	N/A	N/A	269	242.1

5.9 MOVABLE INCORE DETECTORS

3-INT-5000, Appendix 5011

OBJECTIVE

The objective of this test was to demonstrate the operability of the movable incore detector system (flux mapping) by:

1. Demonstrating proper system performance in manual and automatic modes of operation.
2. Determining actual detector path lengths.
3. Verifying all detector thimbles free of obstructions.
4. Installation of permanent system detectors.

DISCUSSION

The test was performed on an intermittent basis over the period of 12-12-85 through 01-02-86. Proper system operation was verified with dummy incore detectors installed. All operations were performed from the flux mapping console located in the control room. In addition, detector path lengths were measured in order to provide alignment data for the automatic flux mapping control system. Once these steps were performed, the actual detectors were installed and proper system operation, including performance of an automatic full core flux map, was verified.

Although the majority of the test was performed with the plant in a cold condition, a full core map was taken under hot standby conditions to ensure the detector paths were free of obstructions and binding would not occur. During this operation, the data link between the flux mapping system and the plant process computer was verified.

RESULTS

The test was performed satisfactorily with no deviation from test acceptance criteria. All thimbles were satisfactorily accessed with both the dummy and permanent detectors. No evidence of binding was experienced. Some minor equipment problems were encountered, but these were readily resolved and operation of control circuitry and indicators was satisfactory.

A problem was encountered when the path lengths determined using the dummy detectors were utilized with the permanent core assemblies. Normal manufacturing tolerances associated with the drive cables results in each cable being inserted a slightly different length for each revolution of the drive wheel. By performing a path length measurement for several paths using the permanent detectors, a correction factor was derived to allow using the original path length data without repeating every path length measurement following the installation or replacement of detector core assemblies.

5.10 DIGITAL ROD POSITION INDICATION

3-INT-5000, Appendix 5015

OBJECTIVE

To verify that the Digital Rod Position Indication (DRPI) satisfactorily provides the required indication for each individual rod, under Hot Shutdown conditions (Mode 3).

DISCUSSION

The test was performed over the period from 01-17-86 to 01-21-86. Each bank of shutdown and control rods was individually withdrawn in 24 step increments to 228 steps. At each 24 step increment, the DRPI on the main control board was compared to the group step counter and plant computer. The DRPI display was required to be within 12 steps of the group step counter and the plant computer. In addition, the control group step counters were required to be within one step of the rod control pulse-to-analog converter output at every 24 step increment.

Each bank was then inserted to within 6 steps of the bottom and jogged to the zero position. The rod bottom indicators were required to actuate at zero steps on the group step counters.

The DRPI main control board display and group step counters were continuously monitored during rod withdrawal and insertion for any indications of improper rod motion.

Initially the plant computer was not providing rod positions due to a software problem in the program that processed the data from DRPI. This was corrected and the test was completed satisfactorily.

RESULTS

The DRPI system provided indications of rod position that agreed with the group step counters and plant computer. No indications of improper rod motion were observed. Rod bottom indication occurred at zero steps. Control bank group step counters agreed within one step with the rod control pulse-to-analog converter.

5.11 LOOSE PARTS MONITORING

3-INT-5000, Appendix 5016

OBJECTIVE

The objectives of this test were to:

1. Obtain baseline system signal data during the reactor plant heatup.
2. Obtain baseline system signal data with the plant at normal operating temperature and pressure.
3. Determine the approximate frequency of spurious alarms.

DISCUSSION

The majority of testing was performed from 01-13-86 to 01-18-86 during the plant heatup at RCS temperatures of 250°F, 350°F, 420°F and 557°F. Testing was completed on 01-20-86.

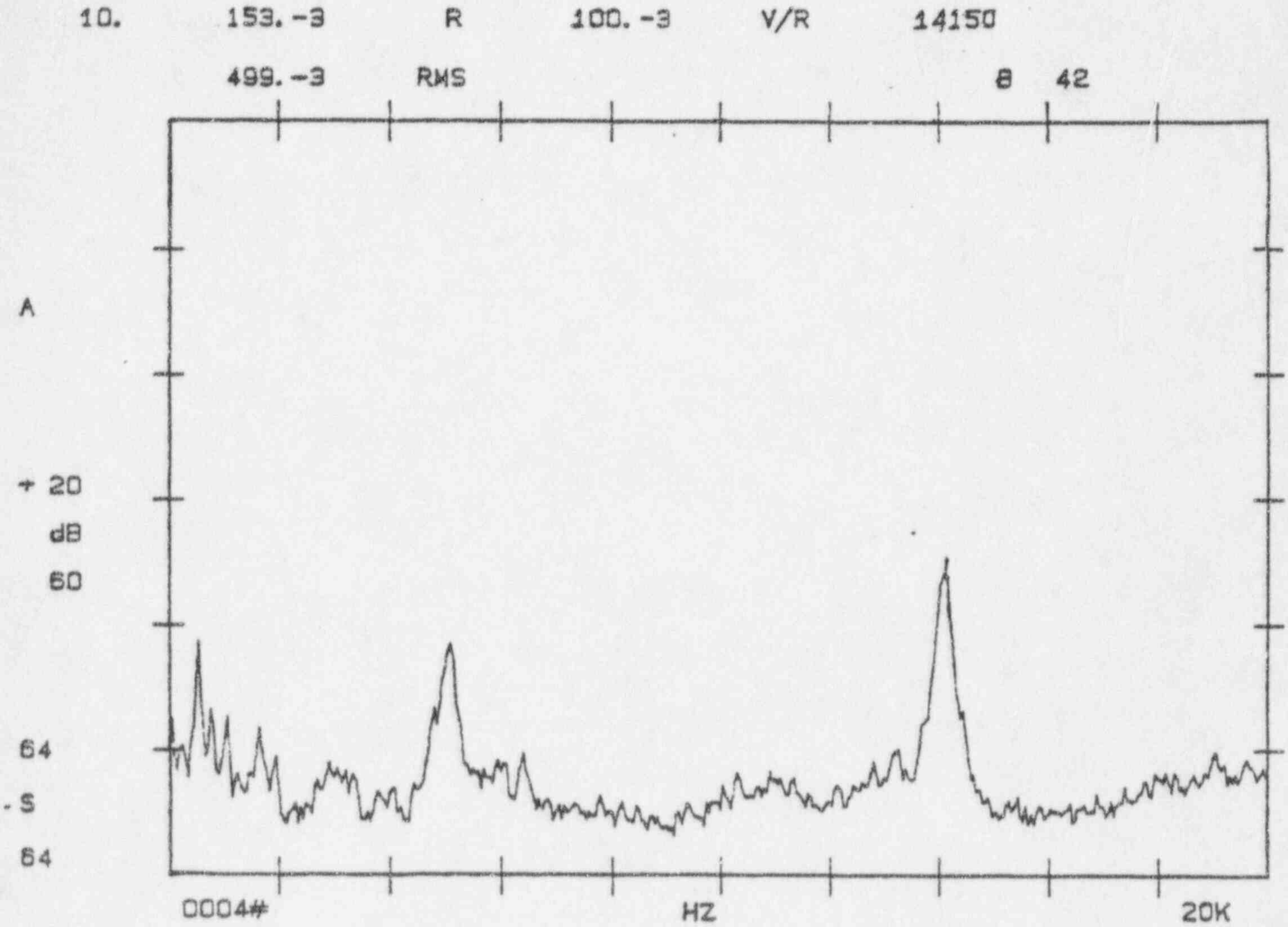
Baseline signal data was obtained by using a spectrum analyzer which was connected to the auxiliary output jack on the Loose Parts Monitoring system (LPM) cabinet. Hardcopy spectrum analysis data was obtained for all 8 monitoring channels during the various heatup temperature plateaus. Additional data was taken at normal operating temperature and pressure by, in sequence, stopping a single reactor coolant pump and monitoring LPM response. The frequency of spurious alarms caused by noise of normal plant operation was also monitored.

The LPM was supplied by Rockwell and consists of a monitoring cabinet with audio output system and integral cassette recorder. There are eight accelerometers located on the primary system: 2 located on the reactor vessel head, 2 located on the lower reactor vessel and one on each steam generator in the channel head area. Prior to test performance, the system was modified by the addition of a 1500 hertz bandpass filter to enhance the capabilities to detect loose parts of a large mass (30 pounds).

RESULTS

All baseline LPM signal data was obtained with no problems encountered. Refer to Figure 5.11-1 for an example of a typical spectrum analyzer output. However, during the test, an excessive number of alarms were received from the lower reactor vessel channels. The accelerometers for these channels are mounted on the incore detector guide tubes just below the bottom of the reactor vessel. Further investigations indicated the alarms were being caused by the noise generated by the incore detector thimbles which were rattling in the guide tubes due to RCS flow. Based on engineering analysis, gain adjustments on the system's 1500 hertz filter were recommended on the affected channels.

Refer to Section 8.5.11 for a discussion of LPM testing conducted during the power ascension program.



Millstone Nuclear Power Station Unit No. 3	TYPICAL SPECTRUM ANALYZER PLOT LOOSE PARTS MONITORING SYSTEM	Figure 5.11-1

5.12 REACTOR COOLANT SYSTEM FLOW COASTDOWN

3-INT-5000, Appendix 5017

OBJECTIVE

The objectives of this test were to:

1. Verify for a trip of one Reactor Coolant Pump (RCP) with the other three pumps in operation that the low flow time delay is less than 2.5 seconds.
2. Verify for a trip of one RCP that all points on the faulted loop flow coastdown curve are above the corresponding points on the predicted curve assumed in the FSAR.
3. Verify for a trip of one RCP that all points on the total core flow coastdown curve are above the corresponding points on the predicted curve as assumed in the FSAR.
4. Verify the Reactor Coolant System (RCS) low flow reactor trip response time is less than the value assumed in the FSAR for the case of four RCPs coasting down.
5. Verify that all points on the total core flow coastdown curve are above the corresponding points on the predicted curve in the FSAR.

DISCUSSION

1. One Loop Coasting Down

   Strip chart recorders were connected to the process rack cards containing the elbow tap d/p transmitter output for all four RCS loops, RCP breaker position, and reactor trip breaker position. A data logger was connected to the process rack cards containing the signals for all three low flow bistables on the RCS loop (loop 1) to be tripped. Once the recorders were connected, the P-8 permissive was simulated (indicating reactor power above 37.5 percent) by jumpering a relay in the SSPS cabinets. With the P-8 permissive present, a reactor trip occurs by tripping one RCP.

The RCP in loop 1 was manually tripped from the control room to initiate the test. The traces, data logger output and plant process computer sequence of events output were then analyzed to measure the trip delay time and to create the RCS flow coastdown curves for comparison to the FSAR curves.

2. Four Loops Coasting Down
During this portion of the test, the strip chart recorders were again connected to all four RCS loop elbow tap d/p transmitter outputs and the contacts to monitor reactor trip breaker position and RCP breaker position. The data logger was connected to all twelve RCS low flow bistables. As before, the P-8 permissive was simulated.

The test was initiated by simultaneously tripping all four RCPs via a common RCP trip switch installed for the test. The traces, data logger output and plant process computer sequence of events data were again analyzed to determine the RCS loop low flow reactor trip response time and the total RCS flow coastdown rate for comparison to the FSAR curves.

RESULTS

1. One Loop Coasting Down
The low flow response time for the one loop coating down case was 0.88 seconds which was less than the acceptance criteria of 1.00 second. A break down of the results is as follows:

Time from when the measured loop flow had decreased to the low flow trip setpoint until the last reactor trip breaker had changed position:

0.43 seconds (from sequence of events data)

Sensor delay time:

0.40 seconds

Gripper delay time:

0.05 seconds

Total:

0.88 seconds

Acceptance Criteria: $\leq$ 1 second

A secondary acceptance was that the time from the reactor coolant pump breaker opening to the time that the rods were free to fall be less than 2.5 seconds. Actual test results are:

Time from the Reactor Coolant Pump Breaker opening to the Reactor Trip breaker opening:

1.8 seconds

Gripper Response time:

.05 seconds

Total:

1.85 seconds

Acceptance Criteria: $\leq$ 2.5 seconds

In addition to the response time, the total core flow was compared to the flow assumed in the FSAR following a pump trip. As shown in Figure 5.12-1, the total core flow remained above the FSAR assumed value.

2. Four Loops Coasting Down

The acceptance criteria for the four loops coasting down test was that the time from when the loop flow had decreased to the low flow trip setpoint until the control rods were free to fall shall be < 1.00 second when considering the worst possible case. The results were:

Time from when the measured loop flow has decreased to the low flow trip setpoint until when the last Reactor Trip Breaker has changed state:

Loop 1	Loop 2	Loop 3	Loop 4
0.327	0.317	0.287	0.232 seconds
0.367	0.327	0.252	0.332 seconds
0.397	0.327	0.252	0.262 seconds

Maximum T_1 = 0.397 seconds

Sensor delay times:

Loop 1	Loop 2	Loop 3	Loop 4
0.271	0.593	0.435	0.450 seconds
0.346	0.515	0.495	0.354 seconds
0.321	0.609	0.454	0.373 seconds

Maximum T_d = 0.609 seconds

Gripper Release Time T_g = 0.05 seconds

Low Flow Trip Time Delay ($T_1 + T_d + T_g$):

T_{LF} = 1.056 seconds

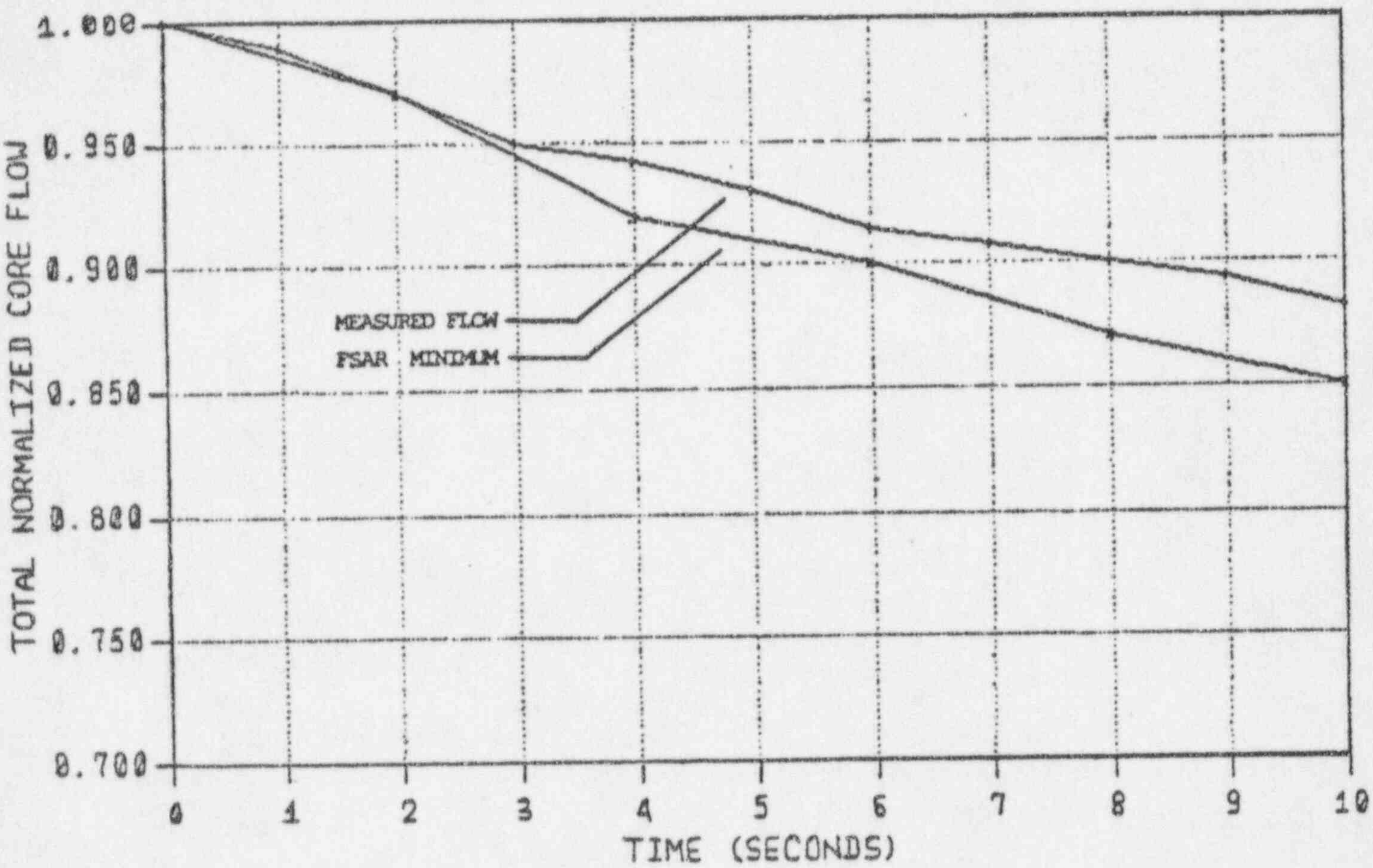
Acceptance Criteria T_{AC} = 1.00 seconds

As the test was originally written, the worst case value from each of the measurements was added to the worst case sensor time delay, and then to the gripper coil release time to determine the overall response time. This process yielded a result of 1.056 seconds which exceeded the test acceptance criteria of 1 second. After discussions with Westinghouse, a different analysis technique was used in determining the response times. This method involved calculating the response times on a loop by loop/sensor by sensor method rather than on a worst case basis. The new results are as follows:

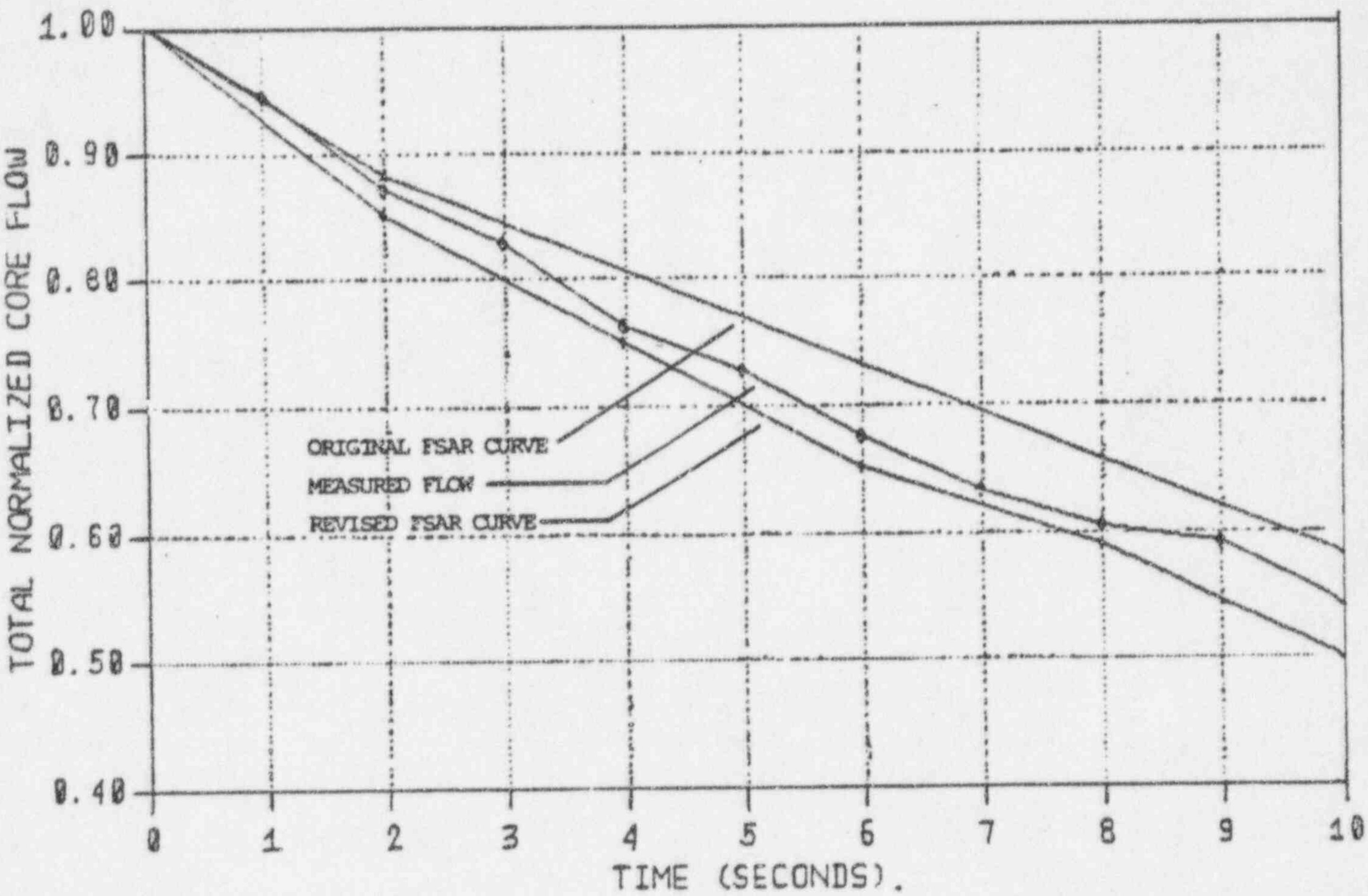
Loop 1	Loop 2	Loop 3	Loop 4
0.648	0.960	0.772	0.732 seconds
0.772	0.892	0.797	0.736 seconds
0.759	0.986	0.756	0.685 seconds

All values were below the acceptance criteria of 1.00 seconds.

The second acceptance criteria for the four loops coasting down test was that the total normalized core flow for the ten seconds of the test was to be greater than the value assumed in the FSAR. Initial review of the test results indicated that the acceptance criteria was not met. However, prior to performance of this test, Westinghouse had performed a reanalysis of the RCS loss of flow accidents. Based upon the new FSAR curves which had been generated by Westinghouse, the acceptance had been met. These test results can be seen in Figure 5.12-2.



Millstone Nuclear Power Station Unit No. 3	TOTAL NORMALIZED CORE FLOW ONE LOOP COASTING DOWN	Figure 5.12-1



Millstone Nuclear Power Station Unit No. 3	TOTAL NORMALIZED CORE FLOW FOUR LOOPS COASTING DOWN	Figure 5.12-2

5.13 ROD CONTROL OPERATIONAL TESTING
3-INT-5000, Appendix 5018

OBJECTIVE

To demonstrate and document, prior to initial criticality, that the rod control system satisfactorily performs the required control and indication functions.

DISCUSSION

The test was performed on 01-20-86 with the plant in Hot Shutdown (Mode 3). Prior to the start of the test, the rod speed control was adjusted to permit maximum rod speed, and the bank overlap setpoints were adjusted to permit the verification of proper operation with minimum rod motion.

The test began by withdrawing each shutdown and control bank, in turn, to 48 steps while comparing Digital Rod Position Indication (DRPI), group step counters and rod motion lights to verify that all rods in the bank under test were being withdrawn. Each bank was then inserted, again verifying proper rod motion on the DRPI, group step counters, and rod motion lights.

After verifying the rod control system could reliably control bank positions, the control bank overlap feature, control bank D full rod withdrawal limit (C-11 interlock) and rod bottom alarms and annunciators were verified. As a prerequisite, all shutdown banks were withdrawn to 30 steps to provide a large source of negative reactivity that could rapidly be inserted, if required. Then control banks A, B, C and D were withdrawn in manual control, while verifying that each bank began motion and ceased motion in accordance with the bank overlap settings in the rod control logic cabinet. During this process, all control banks were stopped at 30 steps. Banks A, B, and C were stopped automatically by bank overlap settings, and D by manual operator control. With all control banks now at 30 steps, the rod control pulse-to-analog converter was advanced to 220 steps using the test pushbutton in the logic cabinet. Manual control bank withdrawal of the D bank was then resumed, and proper operation of the control bank D full rod withdrawal limit (C-11 interlock) was verified by observing that bank D withdrawal halted at 223 steps on the pulse-to-analog converter and that this action was properly annunciated on the main control board.

At that point Bank D was then returned to 30 steps and the pulse-to-analog converter was decremented using the test pushbutton in the logic cabinet, while verifying that the C-11 interlock annunciator cleared.

Next, the "one rod bottom" and "two rod bottom" annunciators were tested by opening the control rod drive mechanism lift coil disconnect switches for all but one rod in shutdown bank E, and inserting the bank E rod in manual. When the single operable rod in shutdown bank E reached zero steps, the "one rod bottom" annunciator was observed to energize. A second rod in shutdown bank E was then enabled by shutting its lift coil disconnect switch and manually inserting this rod. When the second rod reached zero steps, the "two rods bottom" annunciator was observed to energize. At this point, the two shutdown bank E rods were returned to 30 steps and lift coil disconnect switches for all shutdown bank E rods were shut, restoring the rods to service.

With all shutdown and control rods at 30 steps, manual control was again selected and control banks A, B, C and D were inserted while verifying proper bank overlap. The shutdown banks were then restored to zero steps.

Restoration included returning the rod control logic cabinet bank overlap settings, shutdown banks C, D an E rod speeds, and process control system shutdown and control bank speeds to their normal settings.

RESULTS

Proper operation of control and shutdown banks, and proper control bank overlap was demonstrated. Operation of the control bank D full rod withdrawal limit, and rod bottom alarms and annunciators were verified.

5.14 CHEMICAL AND VOLUME CONTROL SYSTEM
3-INT-5000, Appendix 5031

OBJECTIVE

The objectives of this test were to:

1. Verify the ability of the chemical and volume control system to perform boration and dilution of the reactor coolant system.
2. Verify the hot functional degasification capability of the letdown system using the degasification portion of the radioactive gaseous waste (GWS) system.

DISCUSSION

The test was performed over the period of 01-18-86 to 01-22-86. Testing consisted of a series of operational verifications of the Chemical Volume and Control System (CVCS) to operate as intended and meet the limits of the acceptance criteria listed below. All system operations were controlled from the control room. Test data was obtained from permanent plant instrumentation, augmented as required with local test instrumentation.

The acceptance criteria for the test can be summarized as follows:

1. The GWS degasifier operates within design limits for feed pressure inlet temperature, operating pressure, level and return flow temperature.
2. The Charging System (CHS) is capable of increasing or decreasing RCS boron concentration by 100 ± 10 ppm within one hour.
3. The letdown system operates within design limits for flow rates, temperature and filter differential pressure across various system filters. This also served to verify proper sizing of letdown system flow restricting orifice.

4. The hydrogen regulator is capable of maintaining pressure on the CVCS Volume Control Tank (VCT) within design limits.
5. The boric acid and makeup flow controllers are capable of maintaining flow within design limits.

RESULTS

Test acceptance criteria were met with the following exceptions:

1. GWS degasifier feed pressure controller did not operate within design limits. Plant deficiencies, UNS 7477 and UNS 7478, document this problem. Corrective maintenance was performed on the controllers with satisfactory retests.
2. Once testing began, the VCT high temperature alarm setpoint was determined to be too low. Plant deficiency DDR 815 documents this problem. The setpoint was revised and the alarm recalibrated satisfactorily.
3. Differential pressure across various letdown filters exceeded acceptance criteria. Plant deficiencies, UNS 7472 and UNS 7473, document this problem. Based on review of each specific situation, the filter(s) were either replaced or determined to be acceptable as installed.
4. The degasifier outlet conductivity cell provided readings which exceeded the actual conductivity of the outlet flow. Plant deficiency UNS 7476 documents this problem. The conductivity cell was determined to be defective. A replacement unit was installed and satisfactorily retested.
5. The manual makeup to the VCT could not be controlled in accordance with system design. Plant deficiency UNS 7484 documents this problem. Corrective maintenance and recalibration of the controllers was performed. The system was satisfactorily retested.

6. During letdown flow orifice verification, the letdown flowrate through 3CHS*FCV121 exceeded the nominal design limit by approximately 20 percent. Plant deficiency UNS 7488 documents this problem. The actual flowrate was reviewed by Engineering and determined acceptable.
7. During testing, the design VCT hydrogen concentration could not be obtained. Plant deficiency UNS 7491 documents this problem. Further purging of the VCT with hydrogen achieved an acceptable hydrogen concentration. The deficiency was closed based on this action.
8. The desired RCS dilution rate of 100 ppm/hr was not achieved during the test. Plant deficiency UNS 7490 documents this problem. Further investigation revealed a system lineup problem. This was corrected and a satisfactory dilution rate verified by retest.

In addition, as noted under Section 5.1, Shutdown Margin, it was not possible to obtain accurate pressurizer boron samples once the plant was hot. This was because the loop seal drain line for the pressurizer safety valves is connected to the pressurizer sample line. With the RCS heated, condensate from the pressurizer vapor space that had accumulated in the loop seals diluted the pressurizer liquid samples. Plant deficiency DDR 996 covers this issue, and is currently under evaluation.

5.15 REACTOR COOLANT SYSTEM LOOP STOP VALVE AND PUMP INTERLOCKS.
3-INT-5000, Appendix 5033

OBJECTIVE

The objectives of this test were to verify:

1. RCS loop stop valves and bypass valves are capable of being operated only when the appropriate RCS temperature and valve position criteria are satisfied.
2. Remote valve position in the control room corresponds satisfactorily to actual valve position.
3. Opening and closing stroke times for the RCS loop stop valves are $\leq$ 210 seconds.
4. Opening and closing stroke times for the RCS loop bypass valves are $\leq$ 40 seconds.
5. RCPs can be operated when the oil lift pump pressure criteria (< 600 psig) and loop stop valve position criteria (stop valves open) are met.
6. RCP breaker will trip if locked rotor signal is present or if the associated loop stop/bypass valves are in an unacceptable position.

DISCUSSION

The test was performed over the period of 12-28-85 through 01-03-86 with the reactor in a Cold Shutdown (Mode 5) condition. All system manipulations were performed from the control room. Where possible, personnel were positioned to observe equipment operation.

RESULTS

The acceptance criteria were met with the following exceptions:

1. A pressure switch on the D RCP oil lift pump did not function properly. Plant deficiency UNS 7420 was written to document the problem. Corrective maintenance was performed and the component was satisfactorily retested.

2. The closed loop stop valve annunciators on the B and D loop did not function properly. Plant deficiency UNS 7381 was written to document the problem. Corrective maintenance was performed and the components retested satisfactorily.
3. Several loop stop and bypass valves exceed the stated stroke times. No valve exceeded the acceptance criteria by greater than 5 percent. Plant deficiency UNS 7417 was written to document the problem. The stroke times were evaluated by engineering and determined to be accept-as-is.

6.0 INITIAL CRITICALITY

OBJECTIVE

The objective of this testing was to ensure that criticality was achieved in a safe and controlled manner and to verify that the critical boron concentration was within 1 percent ΔK/K of the Westinghouse Nuclear Design Report predicted value.

DISCUSSION

Testing was conducted on 01-23-86. Two procedures were used; the 3-INT-6000 base procedure covered the majority of testing and Appendix 6001 to the base procedure controlled the collection and analysis of Inverse Count Rate Ratio (ICRR) data. A summary chronology is provided in Section 6.1.

Prior to starting the approach to initial criticality, a verification of all Mode 2 Technical Specification requirements was performed. In addition, the startup related surveillances were performed on the Source Range (SR) and Intermediate Range (IR) nuclear instrumentation. Baseline count rates were determined and RCS samples were taken for determination of boron concentration. Initial RCS boron concentration was measured at 1870 ppm. The approach to criticality was begun at 1410 on 01-23-86. The shutdown and control banks were withdrawn, observing proper sequence and overlap in 114 step increments, until control bank D was at 160 steps. ICRR data was taken after each rod pull and plotted. When control bank D was at 160 steps, rod bank withdrawal was stopped and a new set of baseline data was taken. The reactor coolant system dilution was then begun at a rate of approximately 80 gpm. During this procedure, boron samples were taken at 30 minute intervals and ICRR data was taken every 15 minutes. One hour and forty-five minutes after the dilution was started, the dilution rate was reduced to 30 gpm. Ten minutes later the ICRR indicated .2 and the dilution was stopped. The RCS and

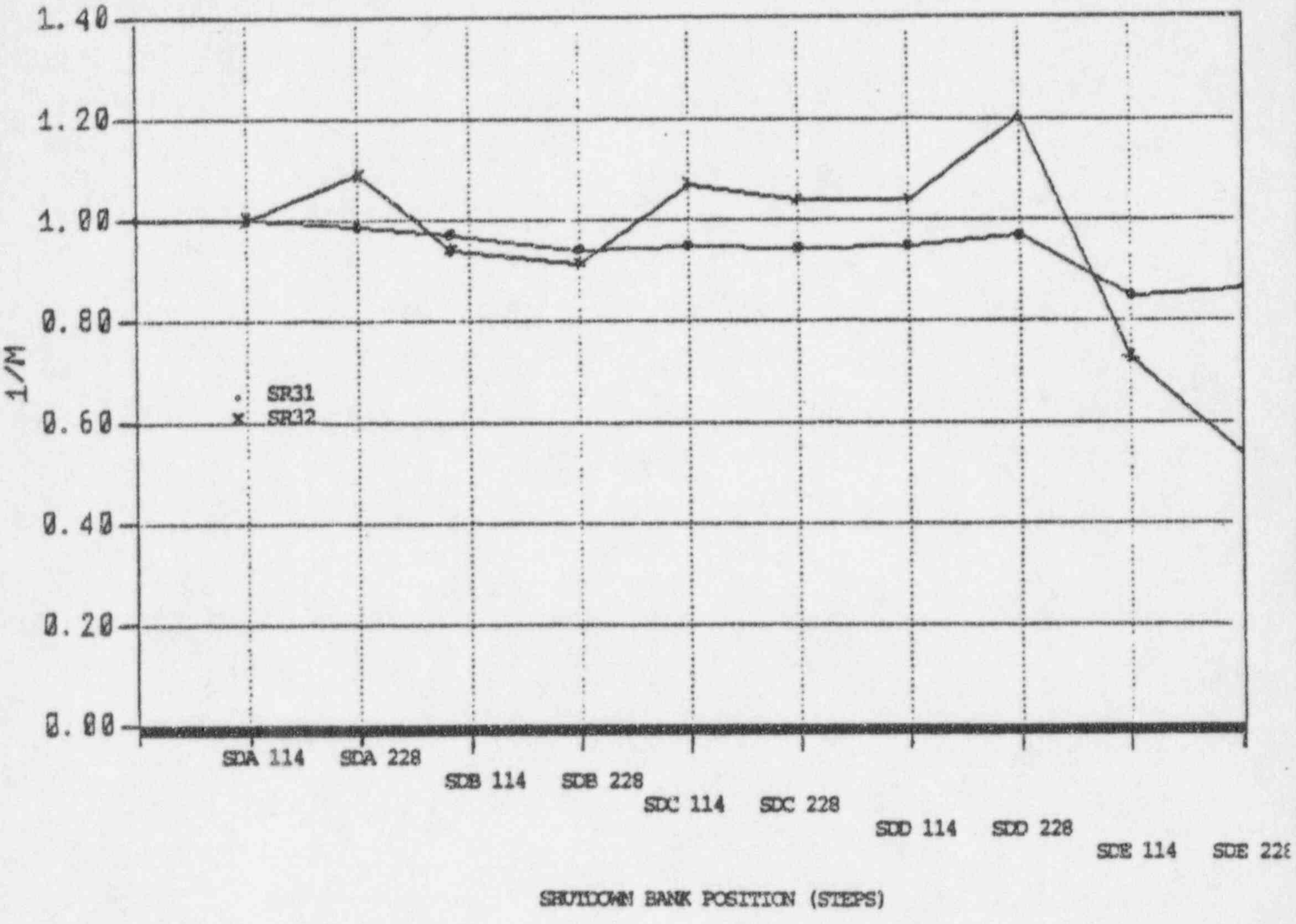
CVCS were allowed to mix until criticality was achieved. The reactor was declared critical 20 minutes after the dilution was stopped at 2200 on 01-23-86. ICRR data for rod withdrawal and dilution to criticality is shown on Figures 6.0-1 through 6.0-4.

RESULTS

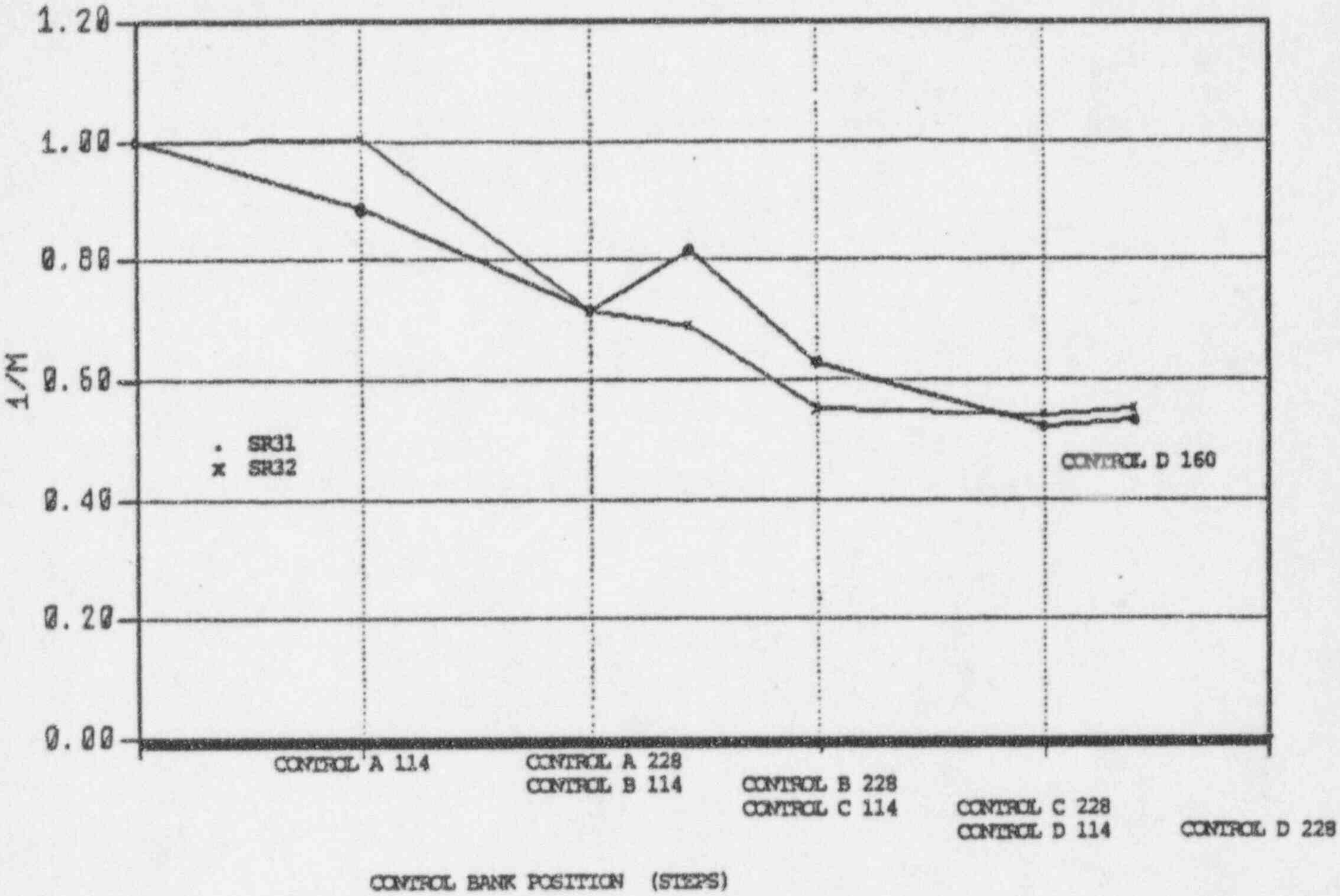
The initial criticality test results are as follows:

	Measured	Predicted
Control Bank D Position	160 steps	160 steps
RCS Boron Concentration	1591 ppm	1559 ppm
T_{avg}	557°F	557°F

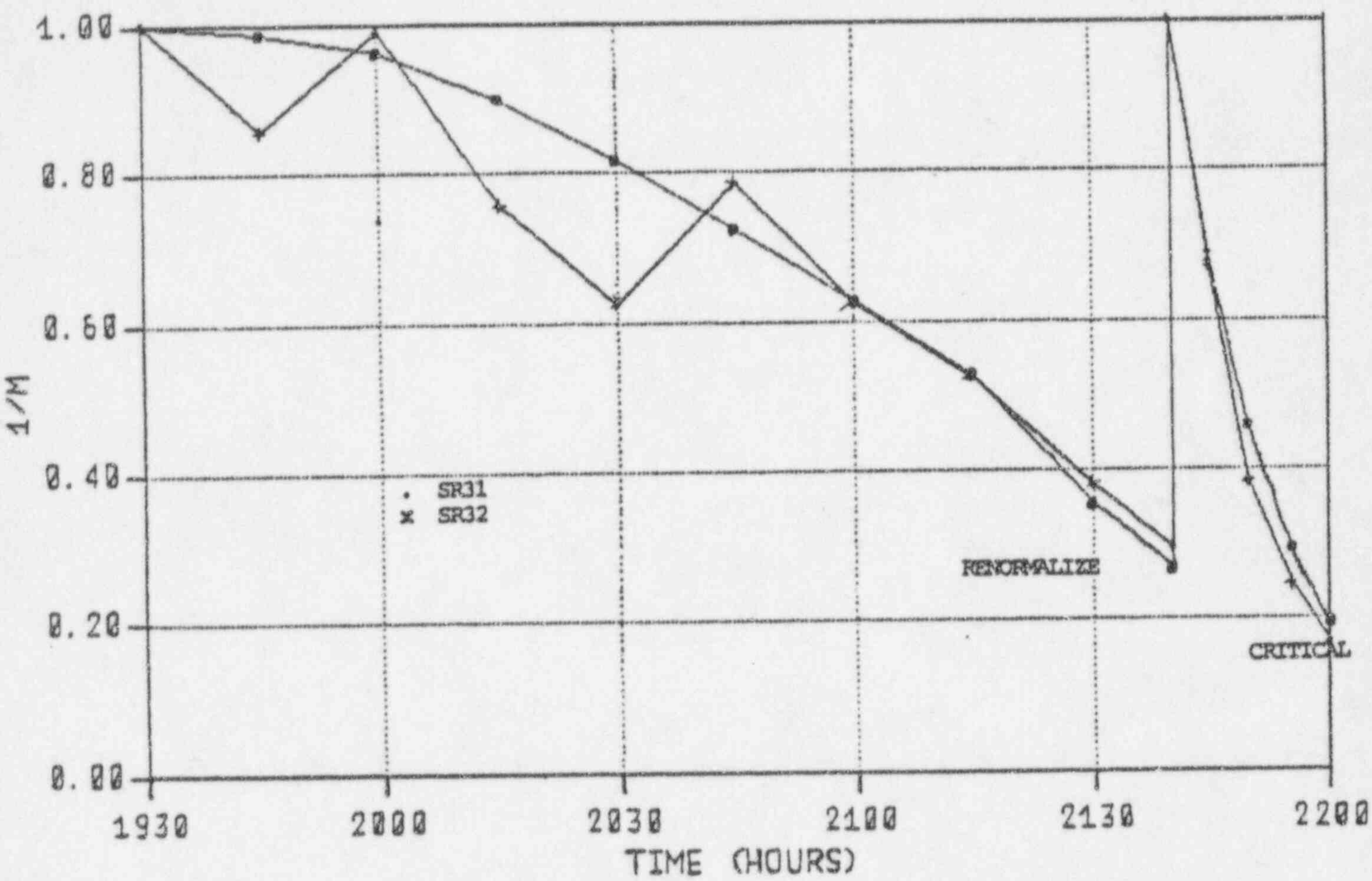
The acceptance criteria of 1%ΔK/K was met although the RCS boron concentration was slightly above the predicted value. This was due to boron mixing that was still occurring in the RCS and CVCS and due to increased Volume Control Tank (VCT) makeup. A more accurate measurement of the All Rods Out (ARO) critical boron concentration was made during low power physics testing.



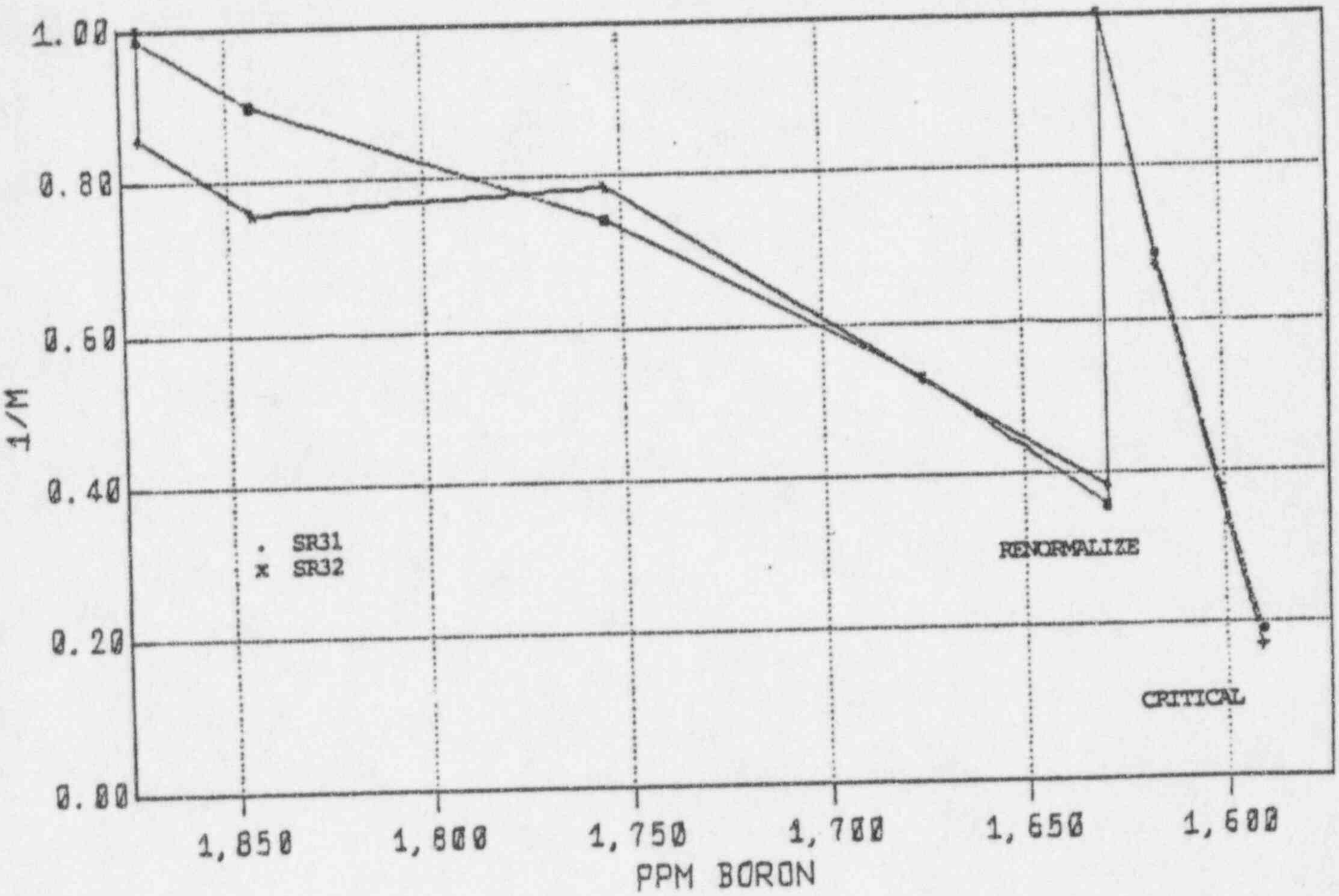
Millstone Nuclear Power Station Unit No. 3	ICRR VERSUS SHUTDOWN BANK POSITION PRECRITICAL ROD WITHDRAWAL	Figure 6.0-1



Millstone Nuclear Power Station Unit No. 3	ICRR VERSUS CONTROL BANK POSITION PRECRITICAL ROD WITHDRAWAL	Figure 6.0-2



Millstone Nuclear Power Station Unit No. 3	ICRR VERSUS TIME DILUTION TO CRITICALITY	Figure 6.0-3



Millstone Nuclear Power Station Unit No. 3	ICRR VERSUS RCS BORON CONCENTRATION DILUTION TO CRITICALITY	Figure 6.0-4

6.1 INITIAL CRITICALITY SUMMARY CHRONOLOGY

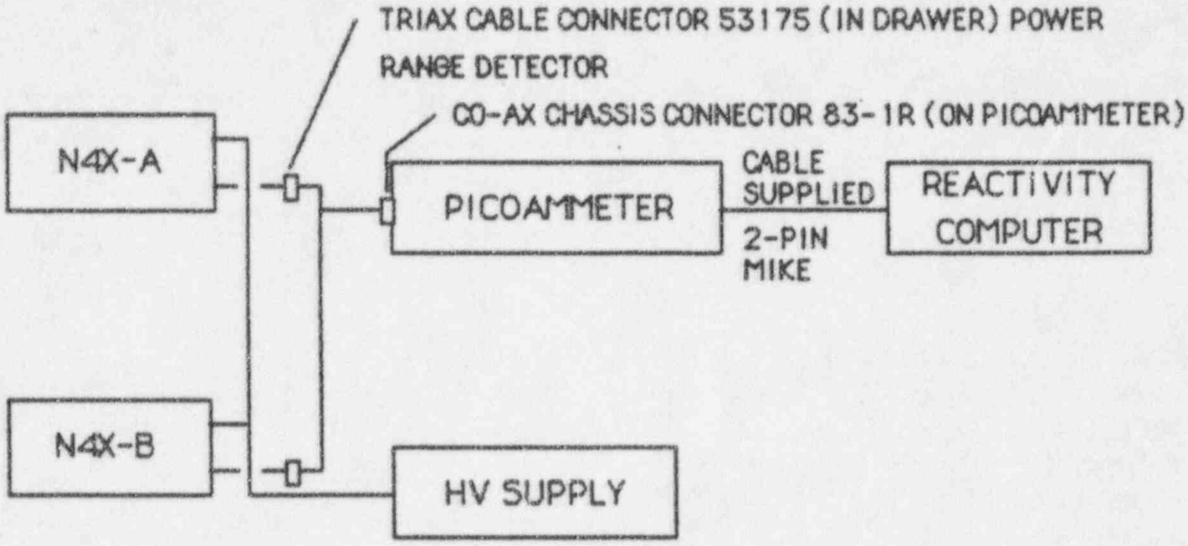
This section describes the major key events during the approach to initial criticality. All listed activities were performed on 01-23-86.

Time	Event
1400	All prerequisites and Initial Conditions are met.
1410	RCS boron concentration is measured as 1870 ppm. Started taking baseline counts for $^1/m$ plots during rod withdrawal.
1449	Started pulling shutdown bank A.
1602	All shutdown banks at 228 steps. RCS boron concentration measured as 1868 ppm.
1649	Started withdrawing control bank A.
1751	Control bank D is at 160 steps.
1800	RCS boron concentration measured as 1872 ppm. Started taking baseline counts for $^1/m$ plots during dilution.
1937	Started diluting the RCS at a rate of 80 gpm.
2130	Reduced dilution rate to 30 gpm.
2140	Dilution stopped.
2145	RCS boron concentration is 1616 ppm.
2200	Reactor critical. RCS boron concentration is 1591.
2215	P-6 interlock is met. The source range trip is blocked.
2318	Reactor power is in the zero power testing range and low power physics tests are started.

7.0 LOW POWER PHYSICS TESTING

The objectives of the low power physics testing (LPPT) program were to obtain the physics characteristics of the as-installed reactor core and to use this information to verify core design calculations. Demonstration of conformance with applicable Technical Specifications was also an objective. The LPPT was conducted with the RCS at normal operating temperature and pressure, 557°F and 2250 psia, respectively. Reactor power was maintained below 1 percent of full power. This power level ensured a good signal-to-noise ratio but was low enough to avoid nuclear heat effects. A reactivity computer system, diagrammed in Figure 7.0-1, was used for reactivity measurements.

The LPPT is summarized in the following sections. In addition to the core physics related testing, a low power natural circulation test was conducted under Appendix 7006 and is described in Section 7.8.



NOTES: The Nuclear Instrumentation Detector cables are Triax cables terminated with Amphenol 43175 connectors. The Keithley picoammeter and power supply inputs are 83-1R Co-ax connectors. Triax connector 53175 mates with Amphenol 52975 for cable-to-cable connection or Amphenol 34475 for cable-to-chassis termination. Chassis Co-ax connector 83-1R mates with cable connector 83-1SP.

Millstone Nuclear Power Station Unit No. 3	ZERO-POWER TESTING CONNECTIONS	Figure 7.0-1

7.1 DETERMINATION OF THE HOT ZERO POWER TESTING RANGE 3-INT-7000, Appendix 7001

OBJECTIVE

The objective of this test was to establish the hot zero power testing range to be used for Low Power Physics Testing (LPPT).

DISCUSSION

The test was performed on 01-24-86. In order to determine the point of adding heat, the core flux level was increased, at a rate of approximately 0.25 dpm, by manual withdrawal of control bank D. During the withdrawal, RCS temperature, intermediate range (IR) and power range (PR) nuclear instrumentation, and reactivity computer output were monitored. The core flux was increased until evidence of nuclear heat addition was detected by an increase in average RCS temperature and a decrease in reactivity. The point of adding heat is the upper limit of the testing range. The lower limit of the testing range was established 2 decades below the upper limit.

RESULTS

The addition of nuclear heat was observed at approximately 3×10^{-7} amps on both IR channels (N35 and N36) and at approximately 1.6×10^{-6} amps on PR channel N44. Channel N44 was used to provide the power input signal to the reactivity computer.

The range of 1.6×10^{-8} to 1.6×10^{-7} amps on PR channel N44 was used as the hot zero power testing range for LPPT.

7.2 REACTIVITY COMPUTER CHECKOUT

3-INT-7000, Appendix 7002

OBJECTIVE

The objective of this test was to verify proper operation of the analog reactivity computer as a prerequisite to performing LPPT.

DISCUSSION

This test was performed on 01-23-86 and 01-24-86. As a prerequisite to performing this test, the Beginning Of Life (BOL) delayed neutron parameters from the Westinghouse Nuclear Design Report were entered into the the reactivity computer. These BOL delayed neutron parameters are listed in Table 7.2-1. A dynamic check of the reactivity computer was then performed using the computer's internal exponential test circuit.

Following criticality, another dynamic check of the computer was performed by comparing the reactivity value calculated by the computer to an inferred value based on stable reactor period. Results of this dynamic test are listed in Table 7.2-2. During LPPT, daily response checks of the computer were performed using the internal exponential test circuit.

RESULTS

An internal exponential response check conducted on 01-24-86 indicated a malfunction with the reactivity computer. The unit was immediately replaced with a second unit. After satisfactorily checking out the replacement unit, LPPT proceeded. Results of the checkout of the replacement computer are listed in Table 7.2-2. In order to validate the test data from the original reactivity computer, the problem with the unit was investigated. This indicated a problem with the exponential test circuit of the computer. The malfunction only affected the output of the computer while in the exponential test mode. Based on this, the data collected during previous testing was determined to be valid. The replacement unit was used during the remainder of LPPT.

Group i	$\bar{\beta}_i$	$\lambda_i (sec)^{-1}$
1	0.000217	0.0125
2	0.001460	0.0308
3	0.001348	0.1153
4	0.002814	0.3113
5	0.000955	1.2466
6	0.000319	3.3466

Where $\ell^* = 18.92$ usec
$\bar{I} = 0.970$

Original Reactivity Computer

Indicated Reactivity ρ_{comp} (pcm)	Stable Reactor Period (pcm)	Inferred Reactivity ρ_{period} (pcm)	Percent * Difference $\frac{(\rho_{comp} - \rho_{period})(100)}{\rho_{period}}$
106.2	50.51	105.46	0.70
63.88	100.27	63.71	0.27
36.0	200.89	35.8	0.56
19.4	400.76	19.3	0.52

Replacement Reactivity Computer

105.35	50.59	105.36	-0.01
63.3	100.76	63.4	-0.16
35.7	200.94	35.8	-0.28
19.20	401.98	19.21	-0.05

*Checkout Acceptance Criteria: Percent Difference $\leq \pm$ 4.0%

Millstone Nuclear Power Station Unit No. 3	REACTIVITY COMPUTER CHECKOUT DATA	Table 7.2-2

7.3 BORON ENDPOINT

3-INT-7000, Appendix 7003

OBJECTIVE

The objective of this test was to determine the just-critical RCS boron concentration for the following control rod configurations:

1. All Rods Out (ARO)
2. Control Bank D in
3. Control Banks C and D in
4. Control Banks A, B, C and D in
5. All Rods In (ARI) except rod F-02[1]

DISCUSSION

For each of the desired control rod configurations, critical conditions were established in the reactor (through borations or dilutions) with the rods as close as possible to the desired configuration. The RCS boron concentration was allowed to stabilize and samples were taken, then the appropriate rods were withdrawn or inserted to achieve the desired configuration. During this final adjustment, the reactivity worth of the rods being moved was measured. The measured reactivity was then converted to an equivalent boron concentration. The RCS boron concentration was then adjusted using the equivalent value. The final adjusted number was the boron endpoint for the applicable control rod configuration.

RESULTS

The boron endpoints determined by this test are given in Table 7.3-1. Also given are the predicted endpoints from the Westinghouse Nuclear Design Report. All test-determined endpoints compared favorably with the design report values.

[1]Rod F-02 is the Most Reactive Rod Stuck Out

Bank Configuration	Measured (ppm)	Predicted (ppm)	M-P * (ppm)
ARO	1571	1566	+5
D In	1517	1499	+18
D+C In	1384	1357	+27
D+C+B+A In	1116	1086	+30
ARI Less RCCA F-02	767	725	+42

*Acceptance Criteria: Difference ≤ ± 100 ppm

7.4 ISOTHERMAL TEMPERATURE COEFFICIENT

3-INT-7000, Appendix 7004

OBJECTIVE

The objective of this test was to determine the Isothermal Temperature Coefficient (ITC). Using the measured ITC and the fuel vendor supplied design fuel temperature coefficient data, the Moderator Temperature Coefficient (MTC) was determined.

DISCUSSION

The test was performed from 01-23-86 to 1-25-86.

A heatup and cooldown of the RCS at a rate of between 10° and 20°F per hour was initiated. During this operation, the change in reactivity versus the change in temperature was recorded on an X-Y plotter. The ITC was determined by measuring the slope of the X-Y plot. The value of the MTC was determined by subtracting out the effect of the fuel temperature coefficient, supplied in the Nuclear Design Report from the ITC.

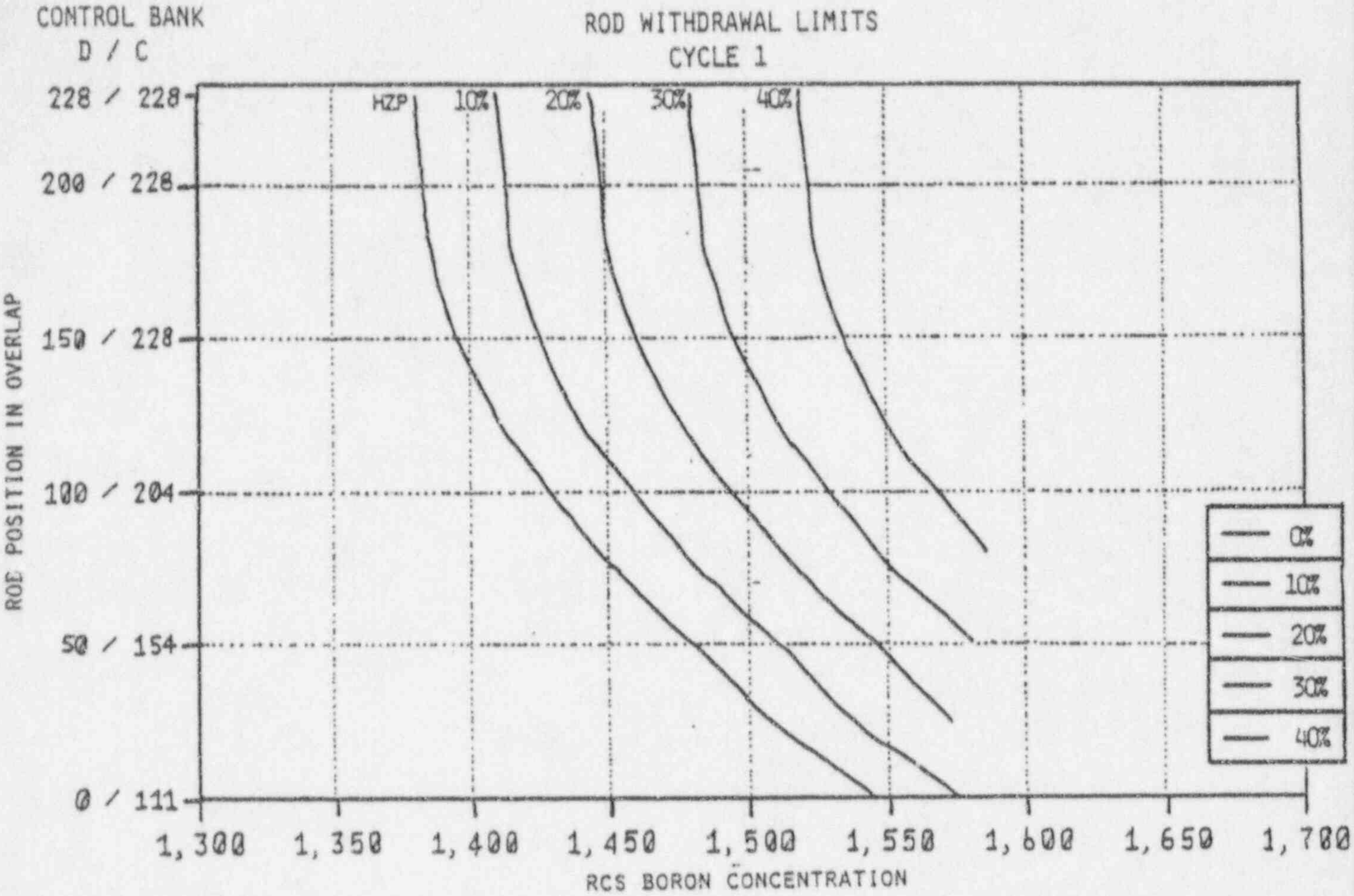
RESULTS

The test results of the ITC measurements are shown on Table 7.4-1. All results are all within the design acceptance criteria as supplied by the fuel vendor. The all rods out value of the MTC was found to be positive. Rod withdrawal limits, as required by Technical Specification 3.1.1.3, were established to maintain the MTC negative at all times during operation. The rod withdrawal limits are shown on Figure 7.4-1.

Bank Configuration	Measured (pcm/ °F)	Predicted (pcm/ °F)	M-P * (pcm/ °F)
ARO	-1.03	-1.69	+0.66
D In	-2.50	-3.24	+0.74
D+C In	-6.07	-6.52	+0.45

Acceptance Criteria: Difference ≤ ± 3 pcm/ °F

Millstone Nuclear Power Station Unit No. 3	SUMMARY OF ITC TEST RESULTS	Table 7.4-1



Millstone Nuclear Power Station Unit No. 3	ROD WITHDRAWAL LIMITS	Figure 7.4-1

7.5 CONTROL ROD WORTH MEASUREMENTS

3-INT-7000, Appendix 7005

OBJECTIVE

The objective of this test was to determine the differential and integral worths of the control and shutdown rod banks, both individually and in overlap.

DISCUSSION

The test was conducted from 01-24-86 to 01-28-86.

Starting from as close to the all rods out (ARO) critical condition as possible, control banks D, C, B, and A and shutdown banks E, D, and C were inserted individually. In each case, a dilution was started using primary grade water. As reactivity was added to the core from the dilution, control rods were inserted in increments to compensate for the reactivity addition. The reactivity inserted by each incremental rod insertion was measured using the reactivity computer. A typical rod worth trace during dilution is shown in Figure 7.5-1. At various points, the dilution was stopped to perform boron endpoint (Appendix 7003) and isothermal temperature coefficient (Appendix 7004) measurements. Prior to the insertion of shutdown bank E, a reactor trip was performed to meet the surveillance requirements of Technical Specification 3.10.1. When shutdown bank C was fully inserted, the dilution was stopped and the F-02 control rod[1] was borated out of the core. The remaining two shutdown banks, A and B, were then diluted into the core to measure the N-1 boron endpoint[2]. At the completion of the N-1 boron endpoint measurement, the reactor was tripped and then borated to the "shutdown banks out/control banks in" critical boron concentration. The reactor was then brought to a critical condition with all shutdown banks out and all control banks in.

[1]Rod F-02 is the Most Reactive Rod Stuck Out
[2]This is a condition with all rods inserted except the Most Reactive Rod Stuck Out

Following criticality, flux was increased to the zero power testing range, and the control banks were borated out in sequence and overlap. As boron was added to the RCS, the control rods were withdrawn in incremental steps, and the reactivity added by each increment was measured on the reactivity computer in order to measure control rod worth in overlap.

RESULTS

All acceptance criteria for the rod worths were met. Table 7.5-1 summarizes the measured rod worths. Figures 7.5-2 through 7.5-19 show the measured integral and differential rod worth curves.

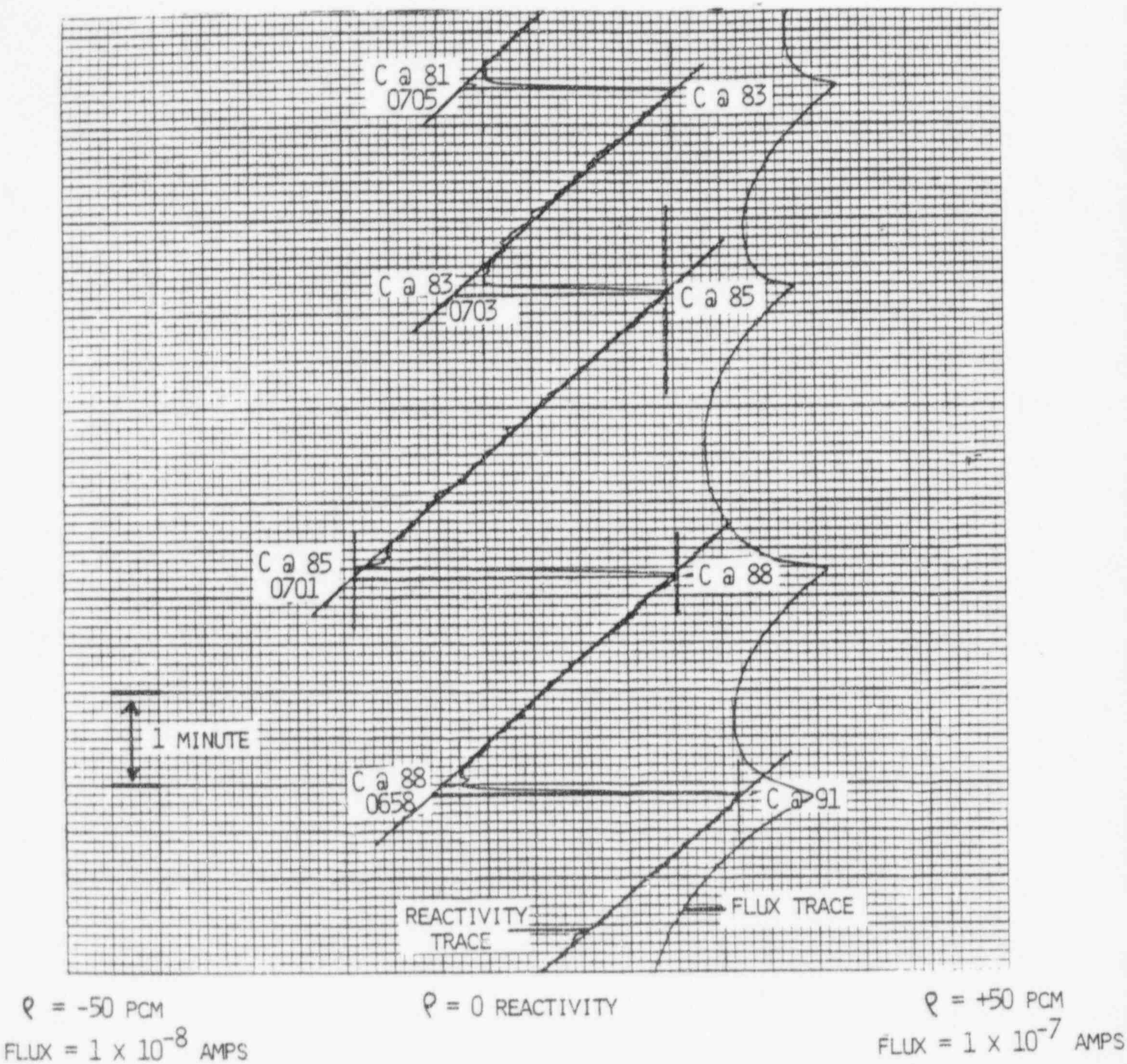
Bank	Measured (pcm)	Predicted (pcm)	(M-P)/P ** (%)
D	619.5	593	+4.46
C (D In)	1223.0	1254	-2.47
B (D+C In)	1239.5	1208	+2.61
A (D+C+B In)	1216.3	1239	-1.83
SDE (D+C+B+A In)	185.7	188	-1.22
SDD (D+C+B+A+SDE In)	547.8	526	+4.14
SDC (D+C+B+A+SDE+5 DD In)	679.6	655	+3.74
ARI Less RCCA F-02	7925.7	7571	+5.58
D-12 (HZP Ins Limit)	386.9	491 ***	N/A
Control Banks In Overlap	4365.6	4298.3*	+1.56

* Sum of individual predicted control bank values.

**Acceptance Criteria: Percent Difference ≤ ± 10%

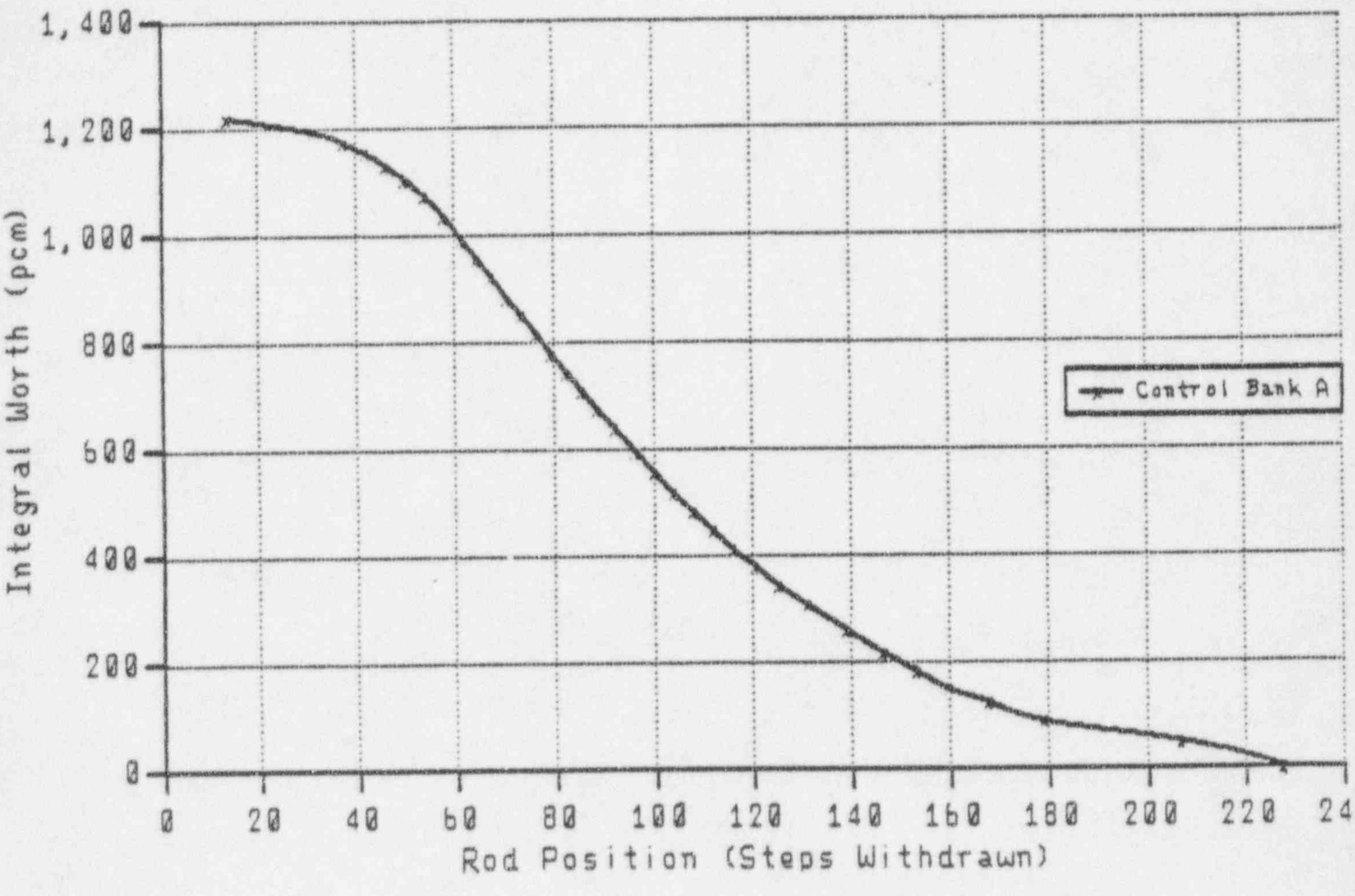
***Acceptance Criteria: Measured < 491 pcm

Millstone Nuclear Power Station Unit No. 3	SUMMARY OF ROD WORTH TEST RESULTS	Table 7.5-1

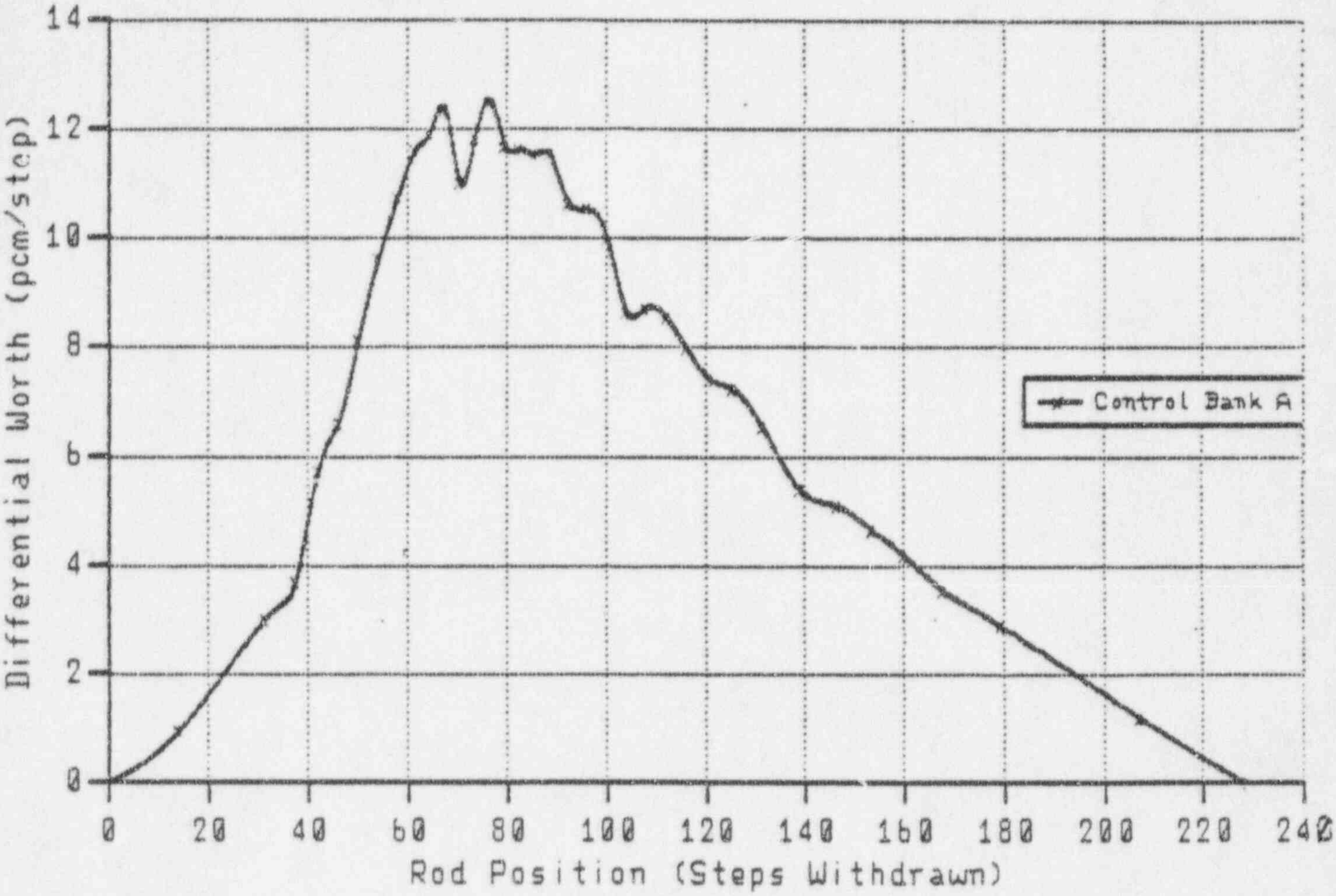


Note: Dilution of Control Bank C shown

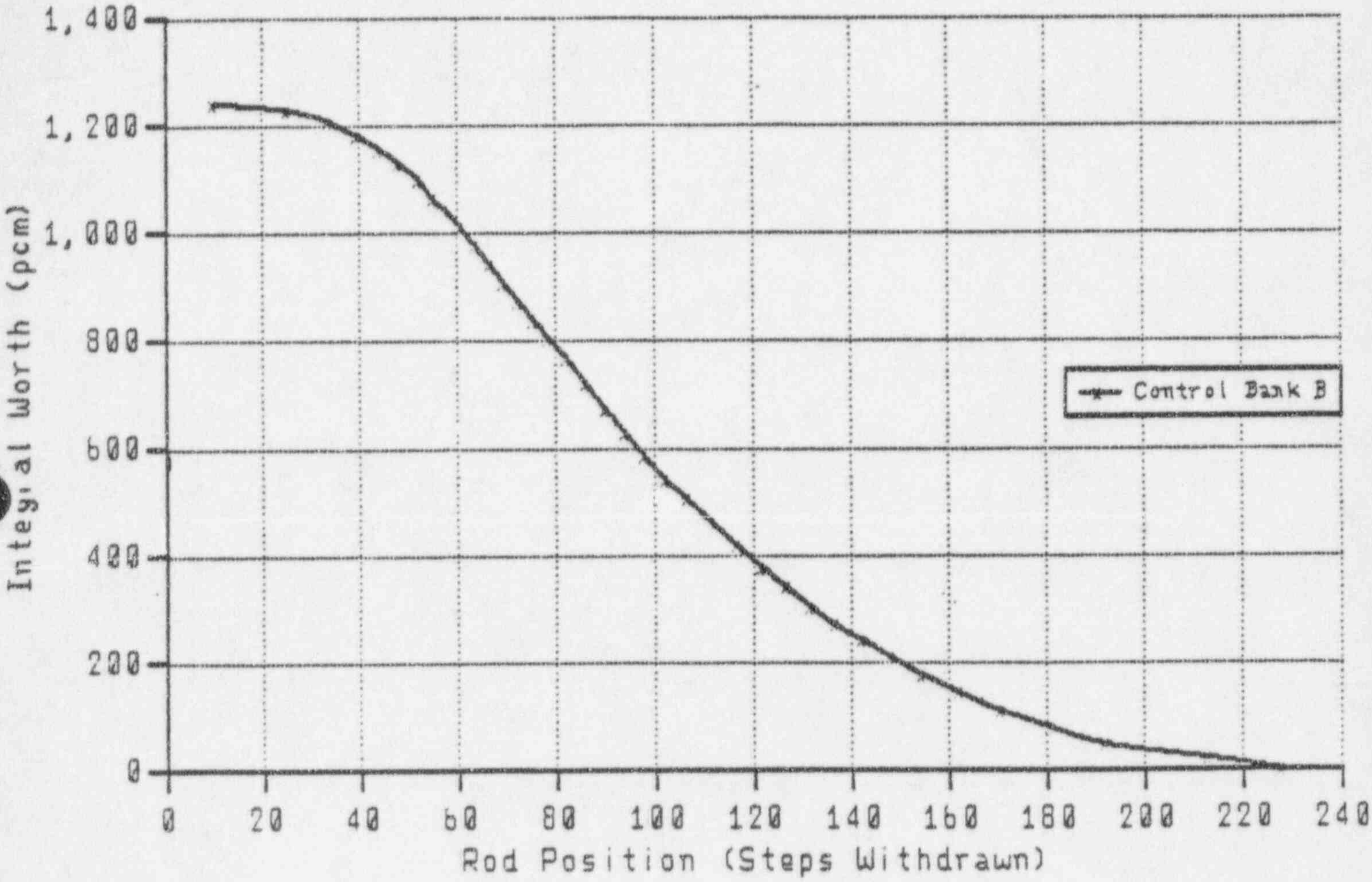
Millstone Nuclear Power Station Unit No. 3	CONTROL ROD WORTH MEASUREMENTS TYPICAL REACTIVITY TRACE	Figure 7.5-1



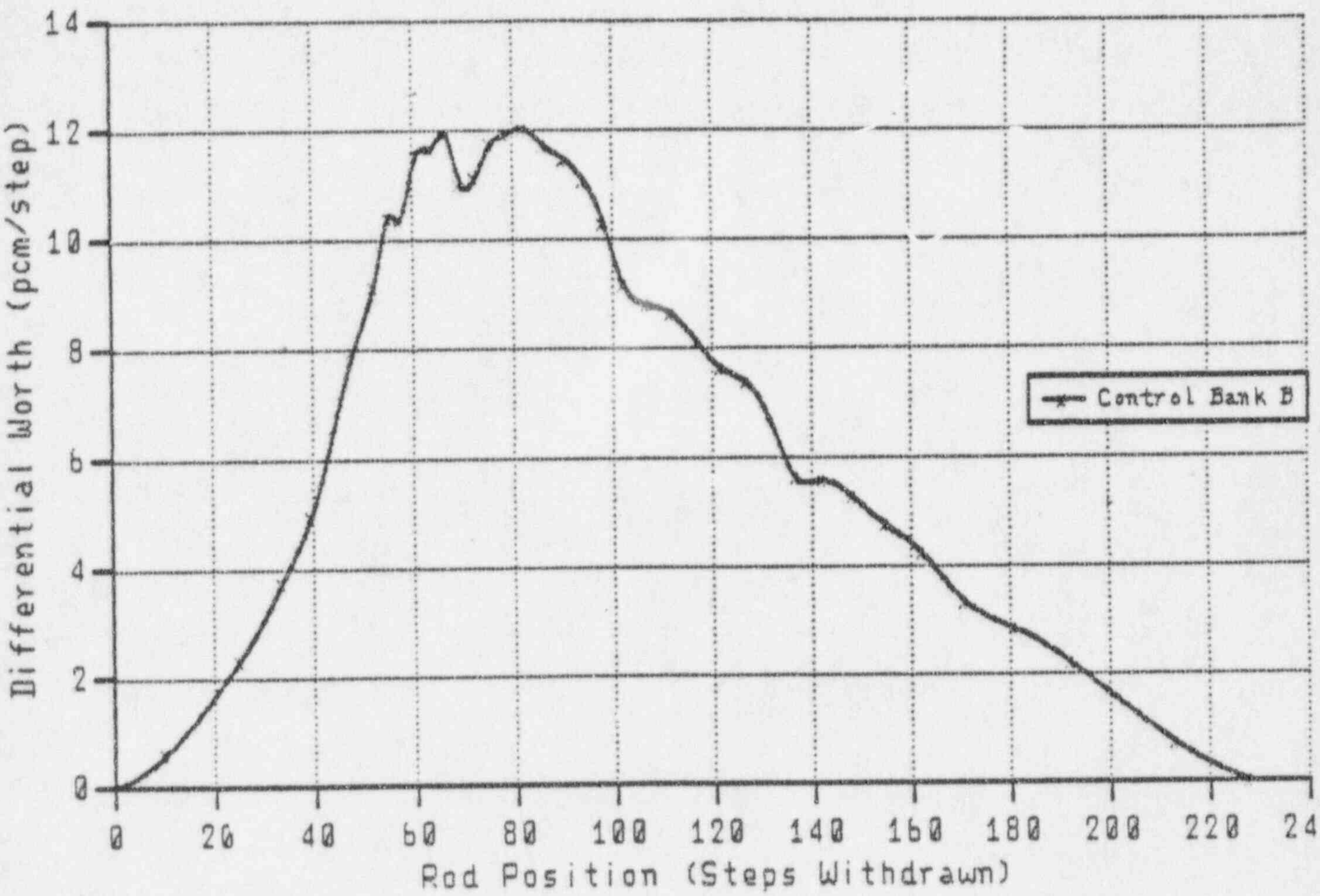
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH CONTROL BANK A	Figure 7.5-2



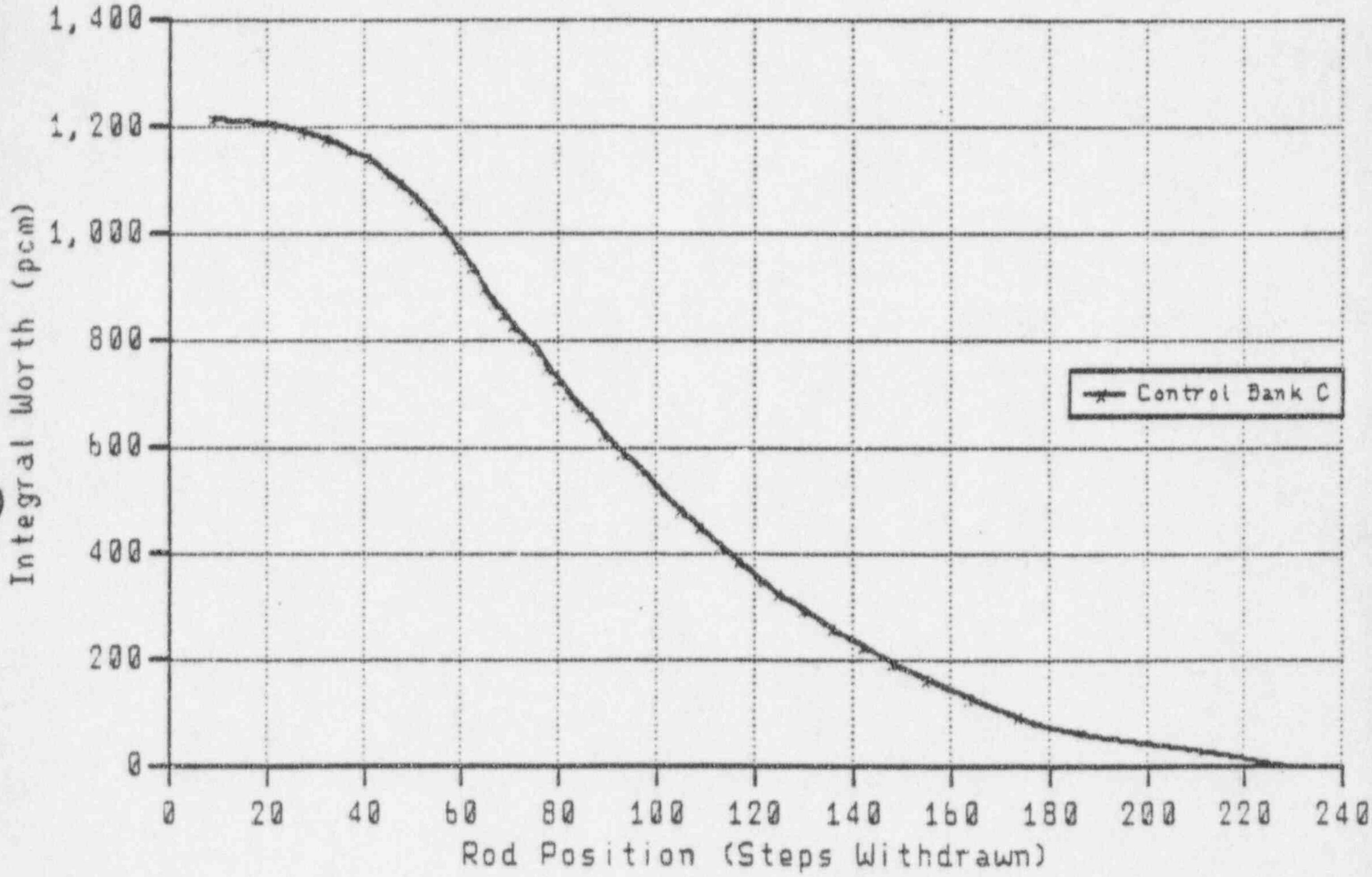
Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH CONTROL BANK A	Figure 7.5-3



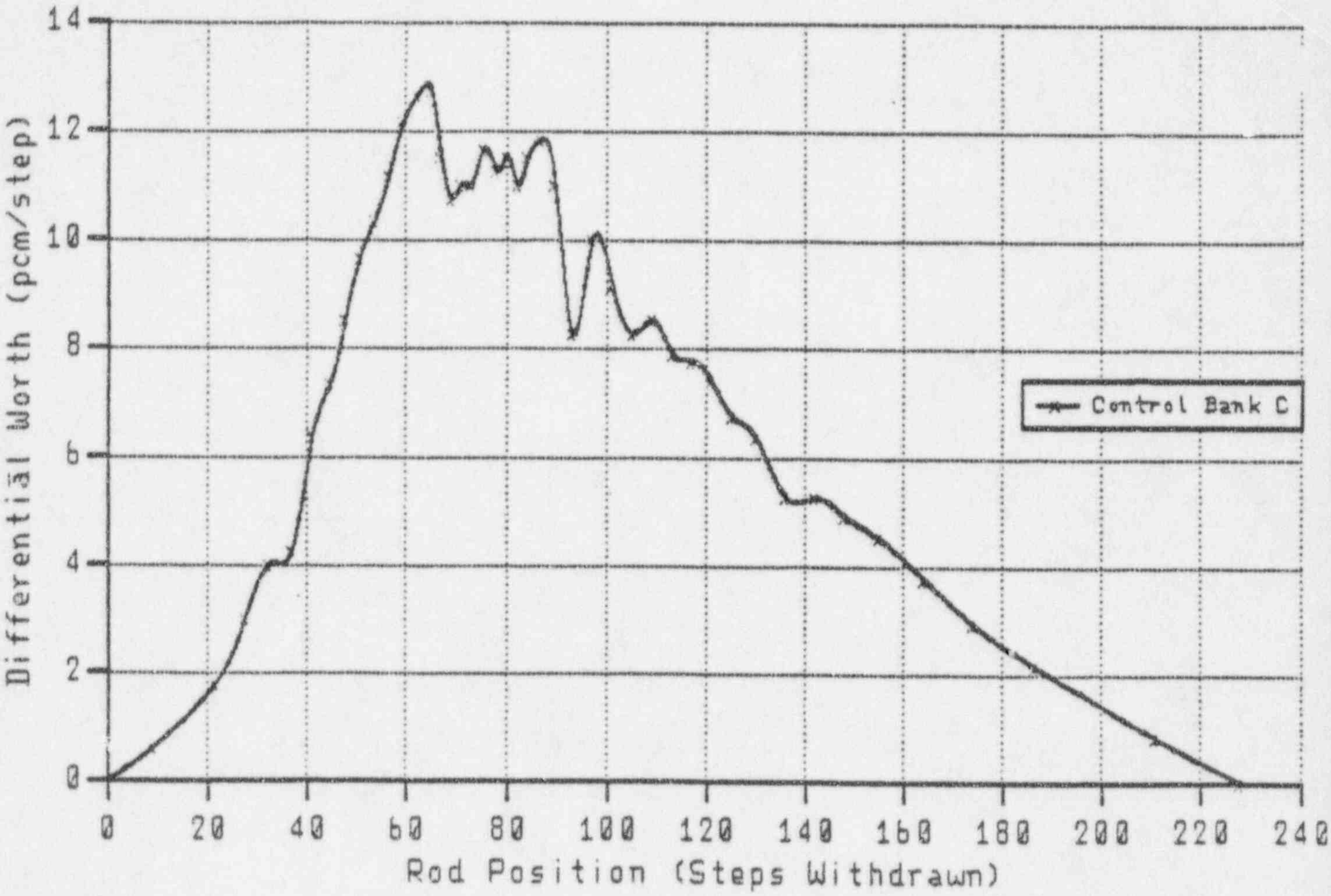
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH CONTROL BANK B	Figure 7.5-4



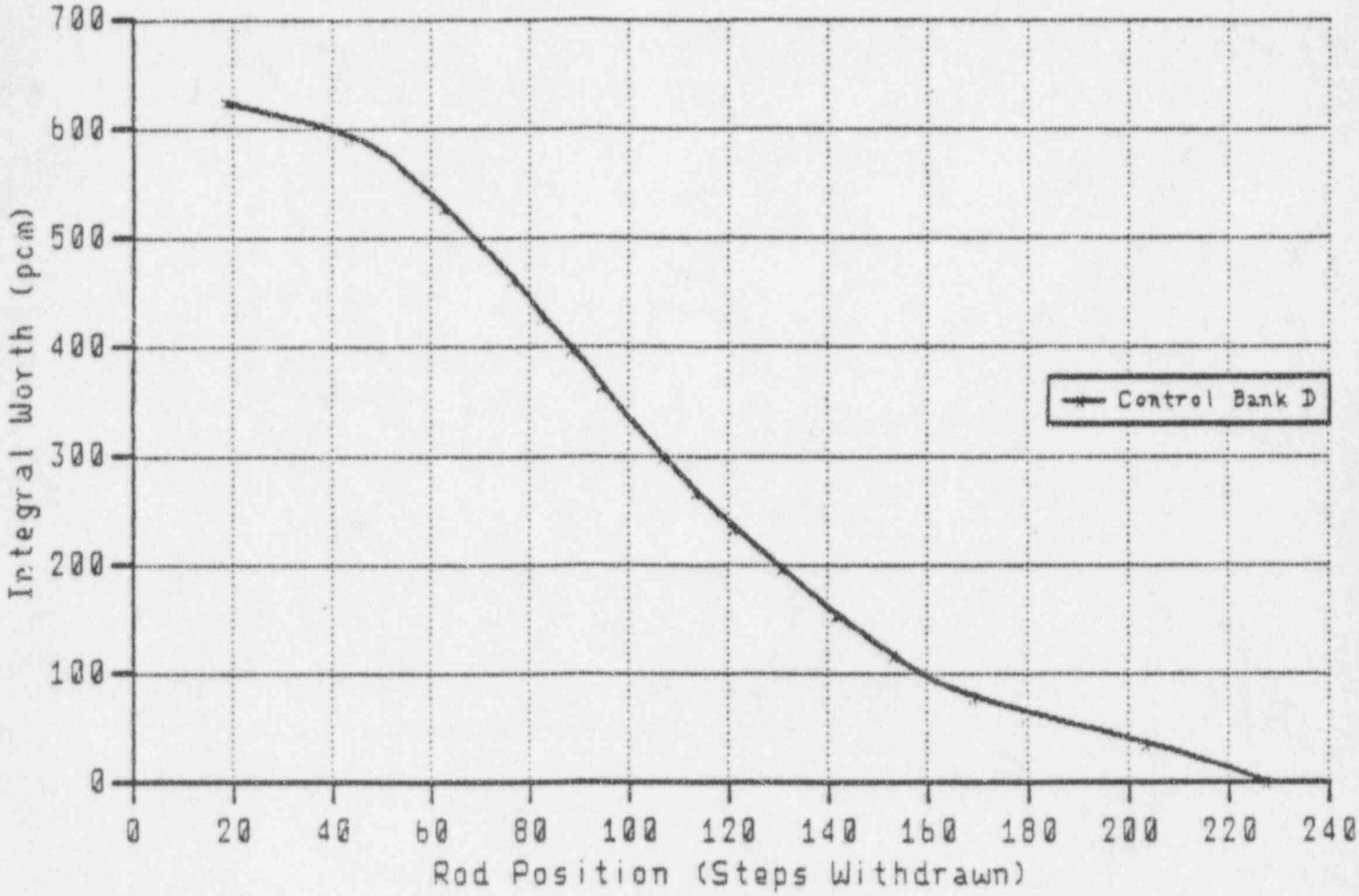
Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH CONTROL BANK B	Figure 7.5-5



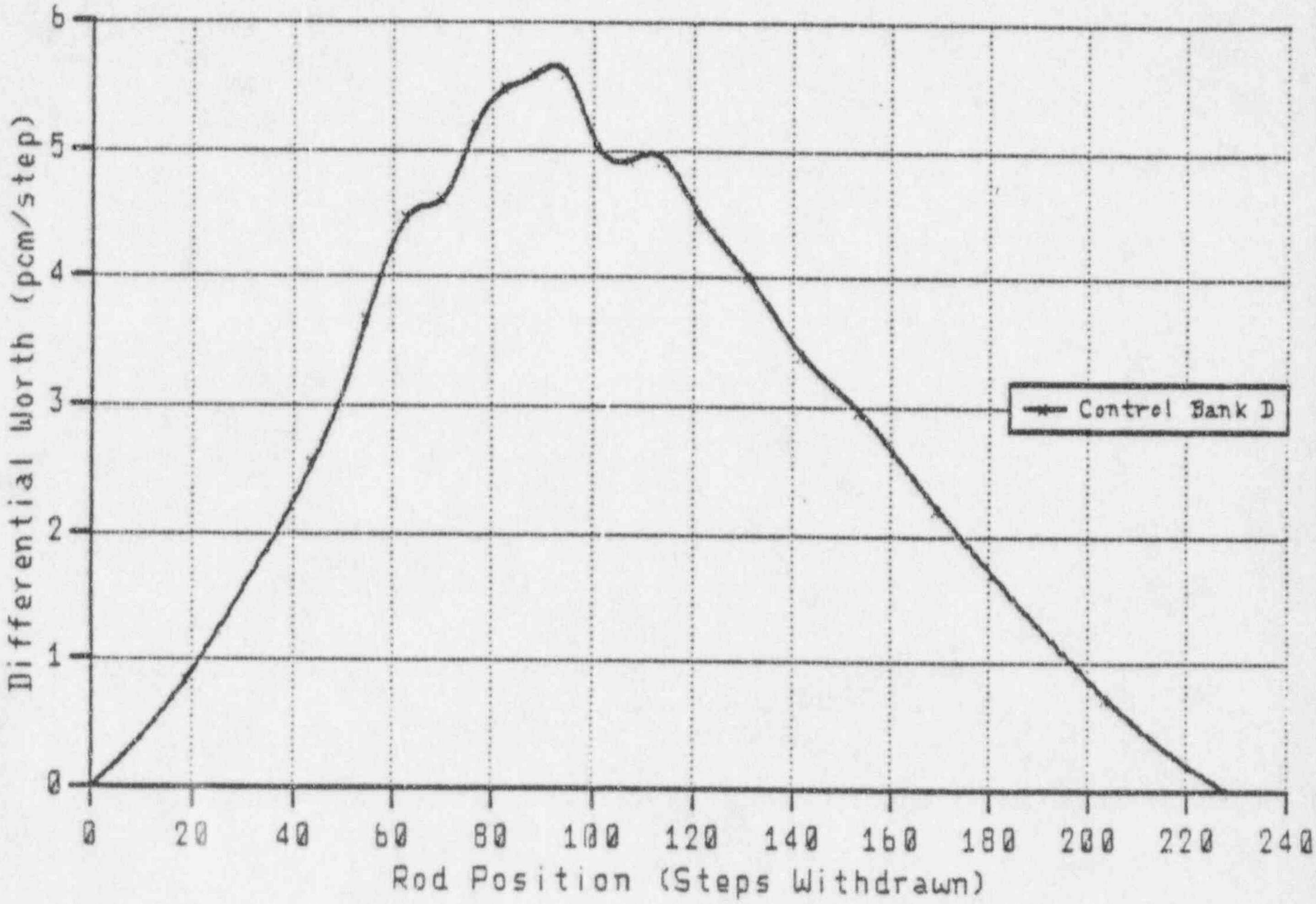
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH CONTROL BANK C	Figure 7.5-6



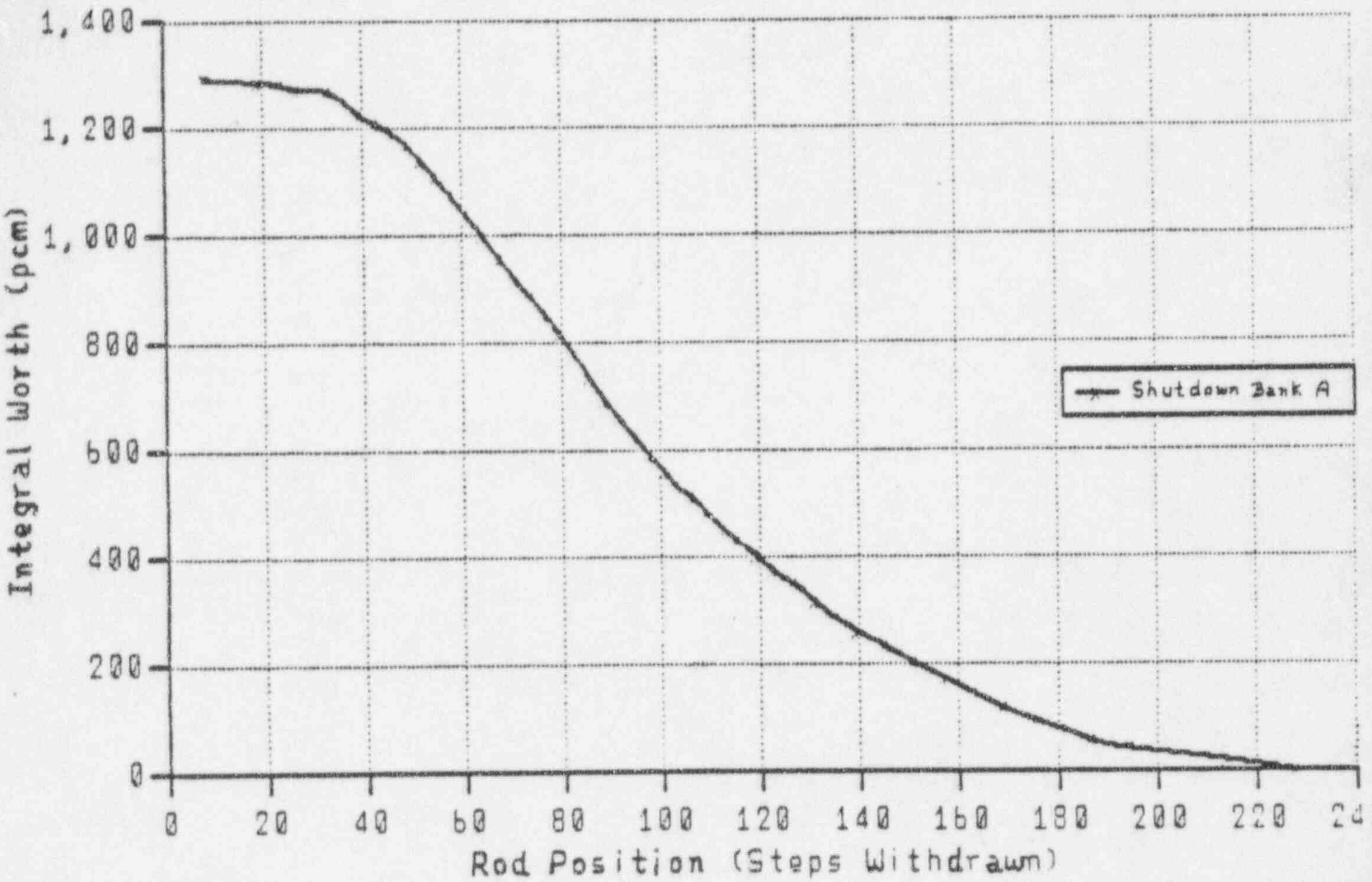
Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH CONTROL BANK C	Figure 7.5-7



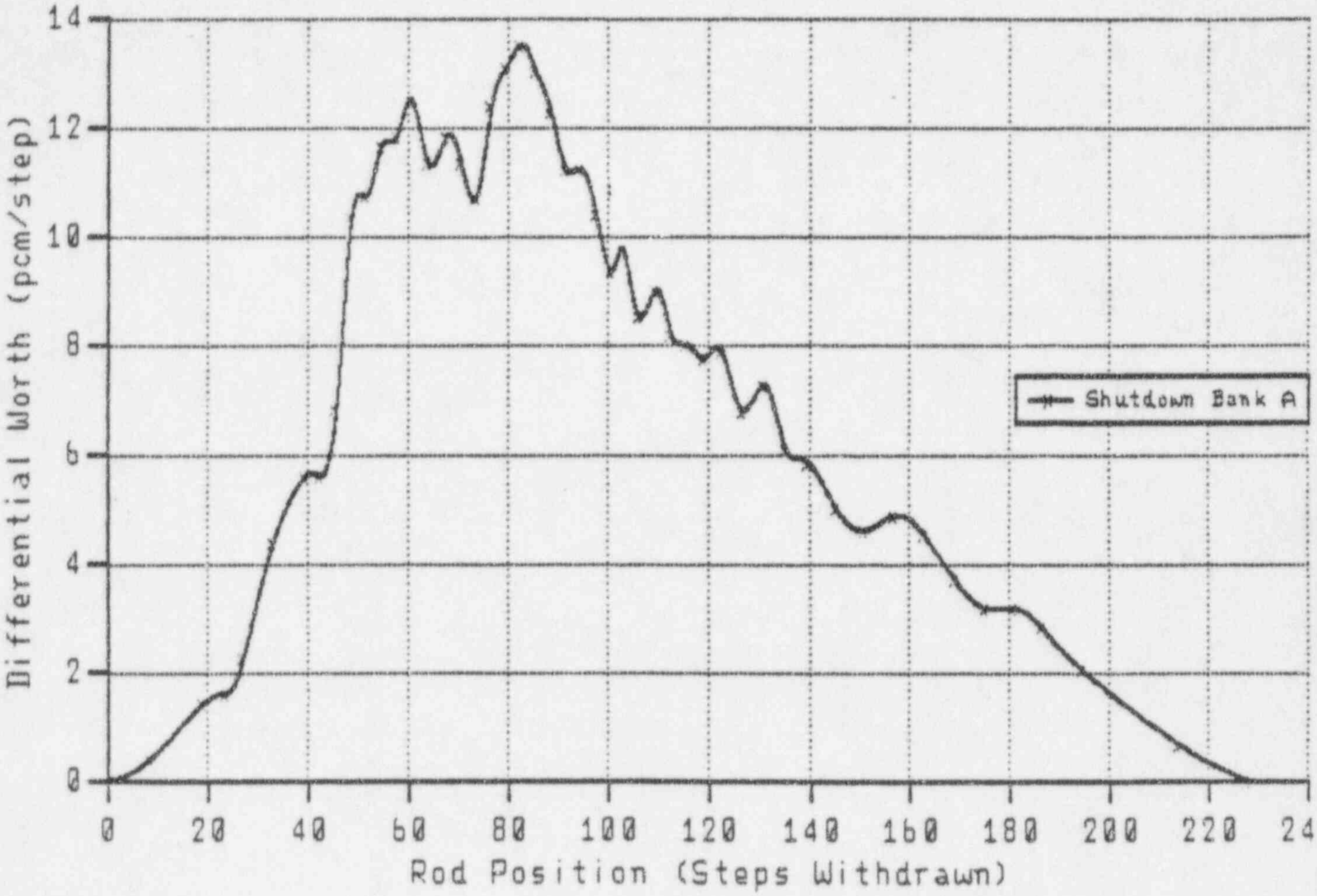
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH CONTROL BANK D	Figure 7.5-8



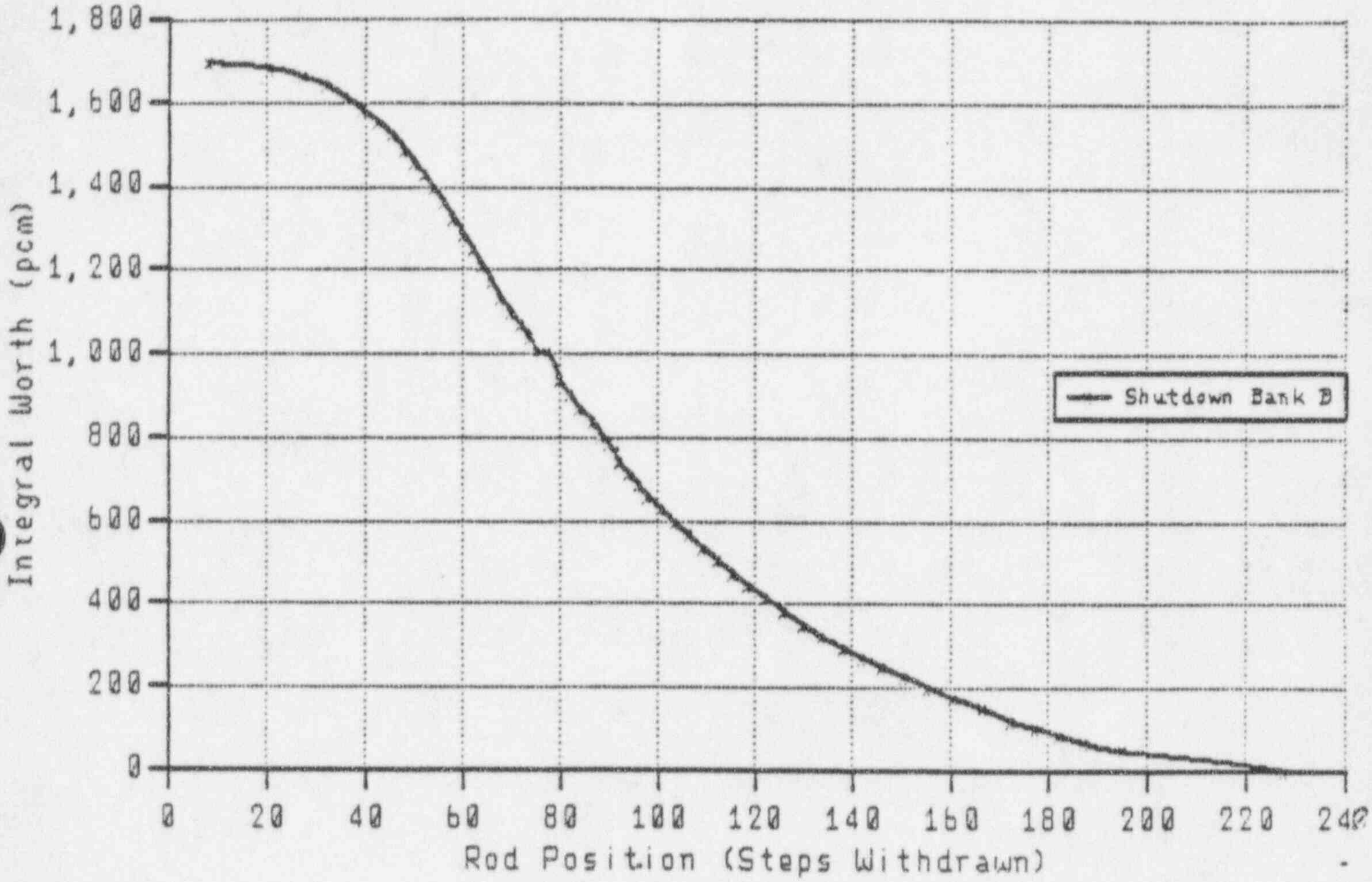
Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH CONTROL BANK D	Figure 7.5-9



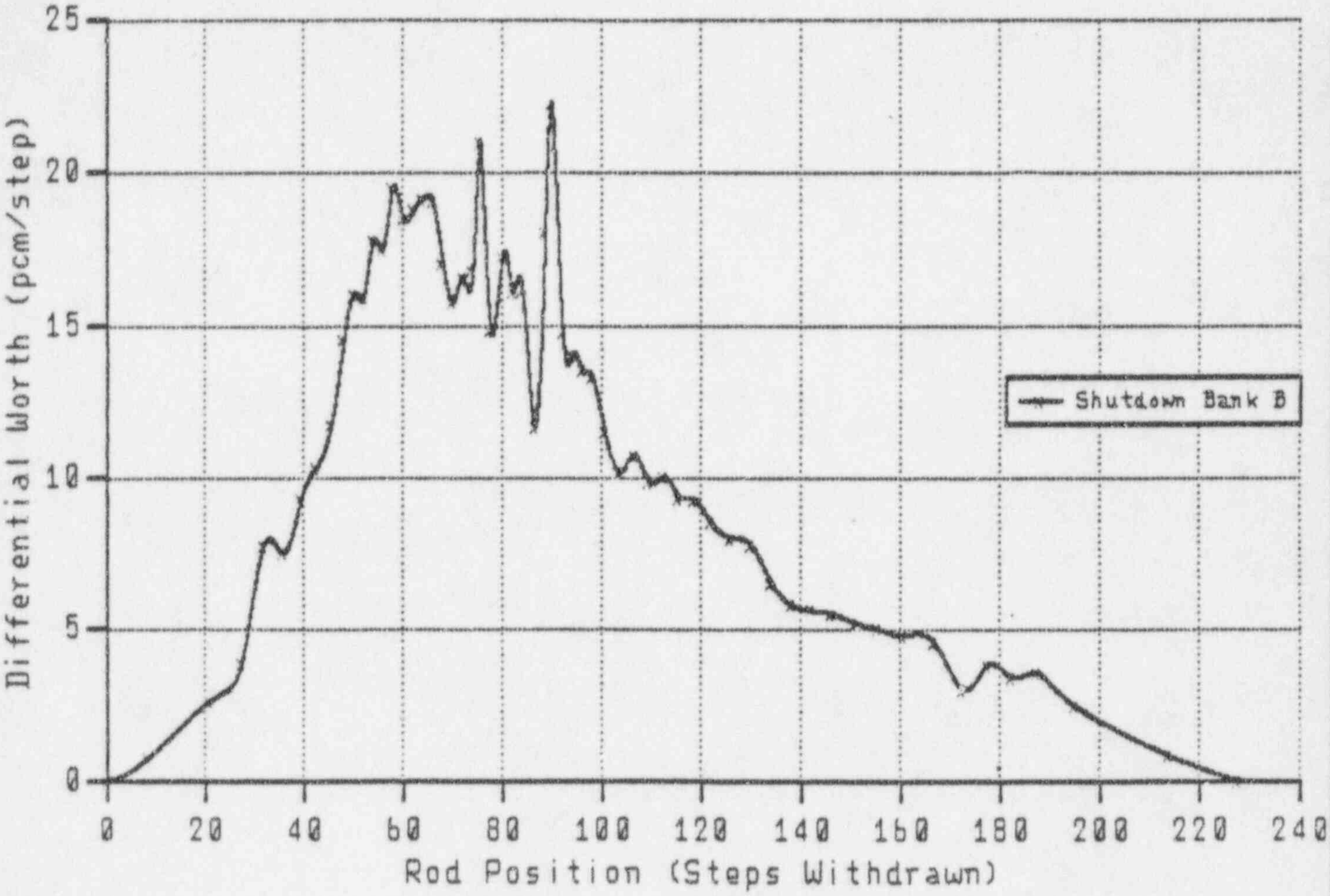
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH SHUTDOWN BANK A	Figure 7.5-10



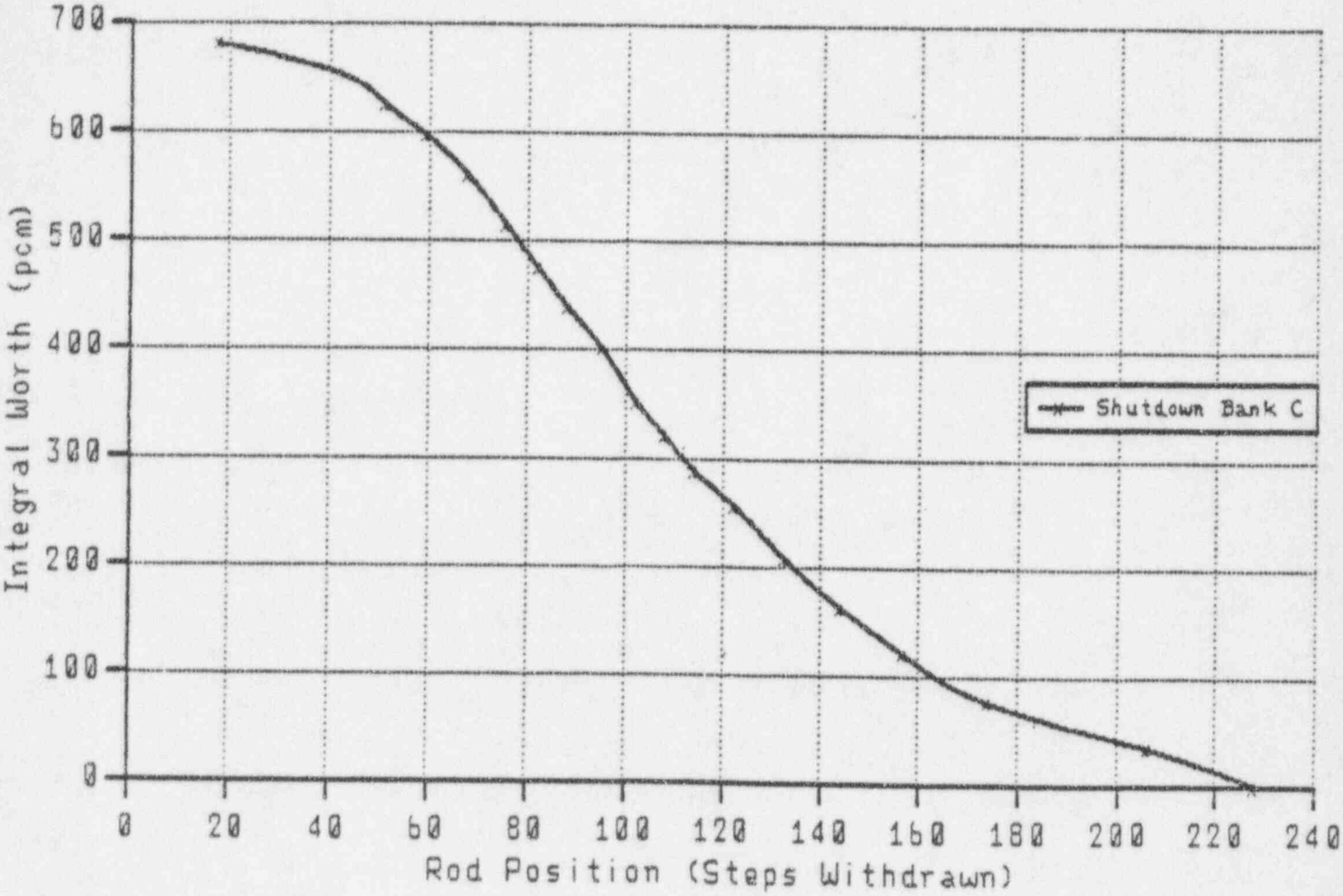
Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH SHUTDOWN BANK A	Figure 7.5-11



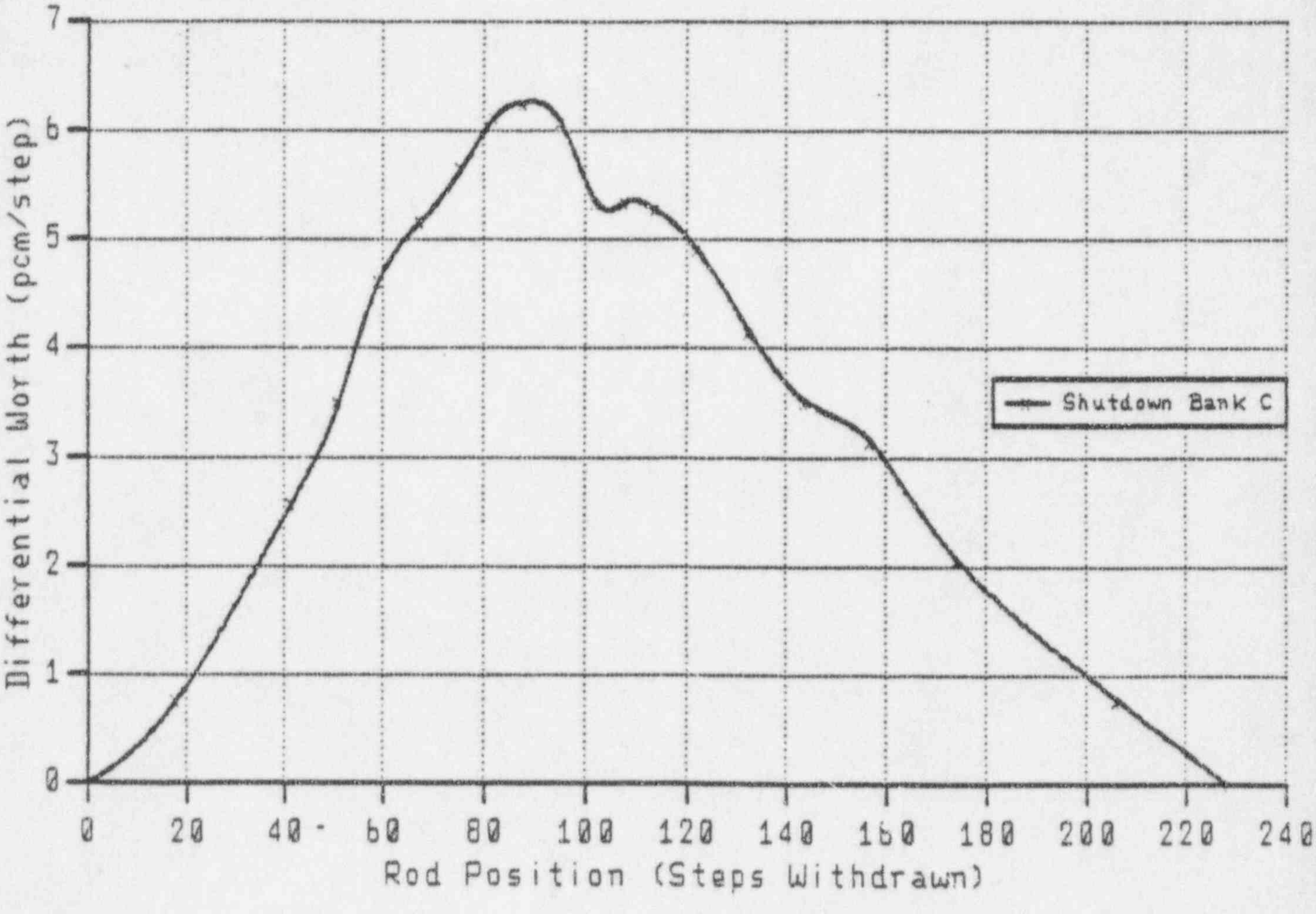
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH SHUTDOWN BANK B	Figure 7.5-12



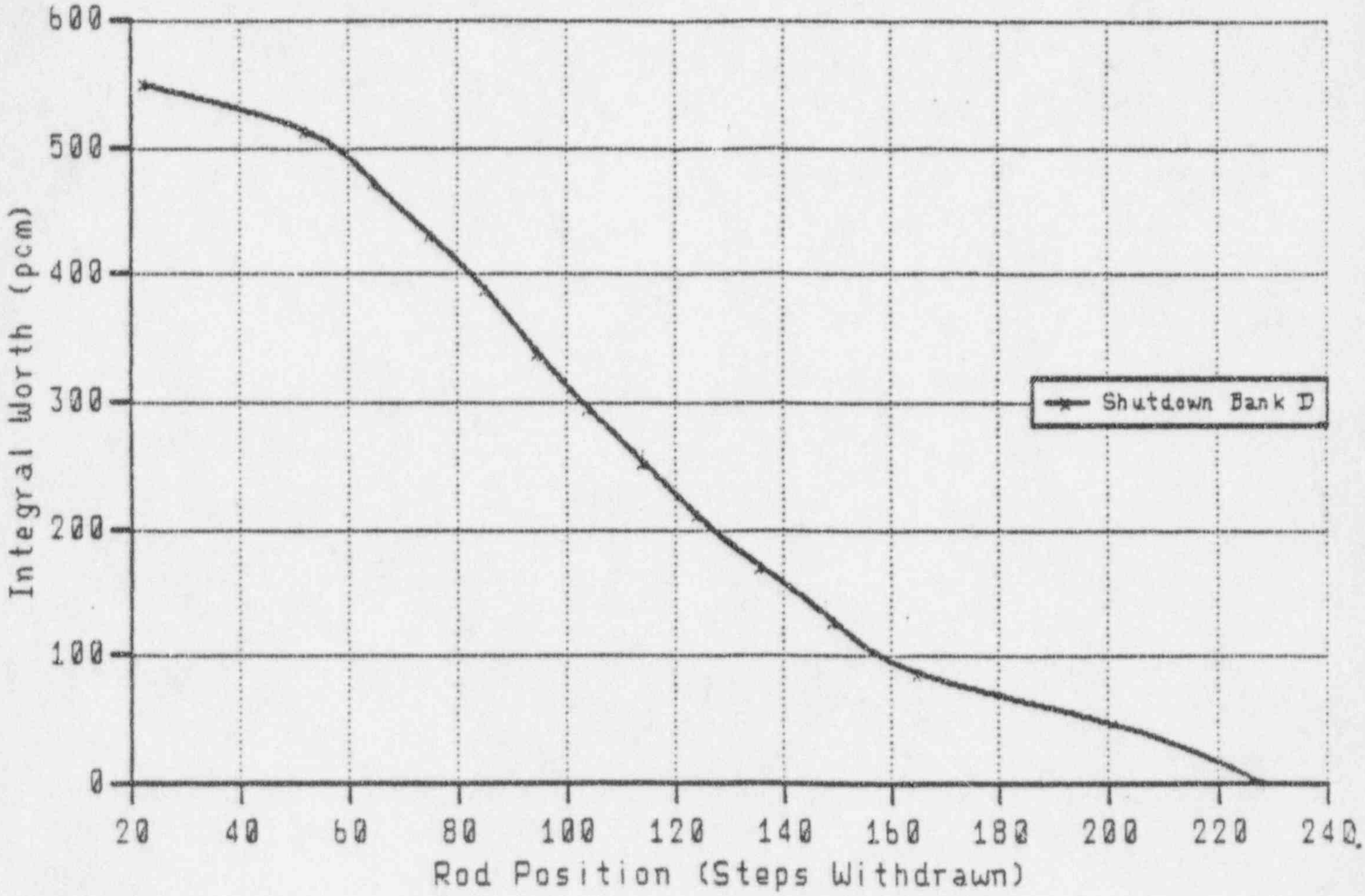
Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH SHUTDOWN BANK B	Figure 7.5-13



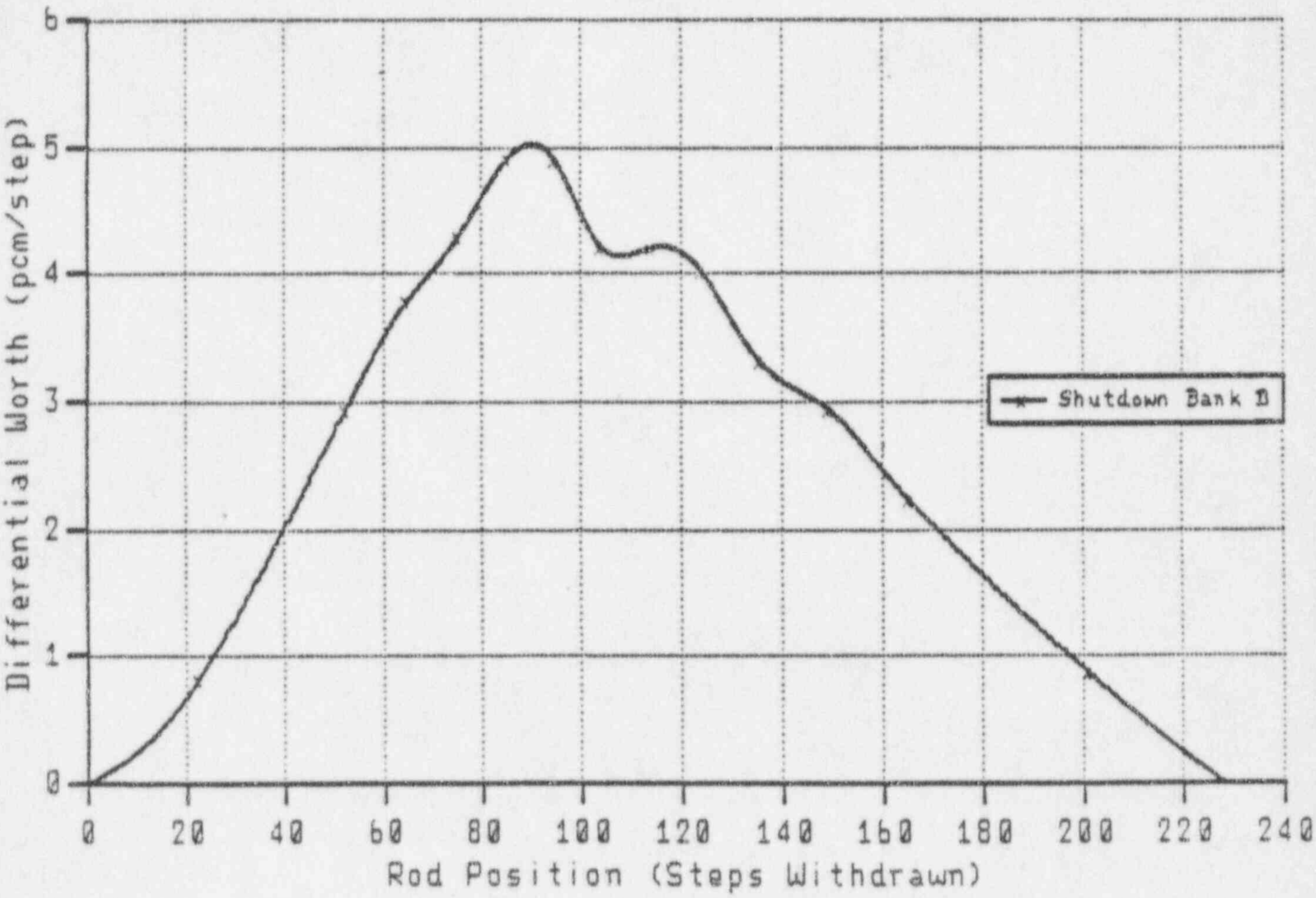
Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH SHUTDOWN BANK C	Figure 7.5-14

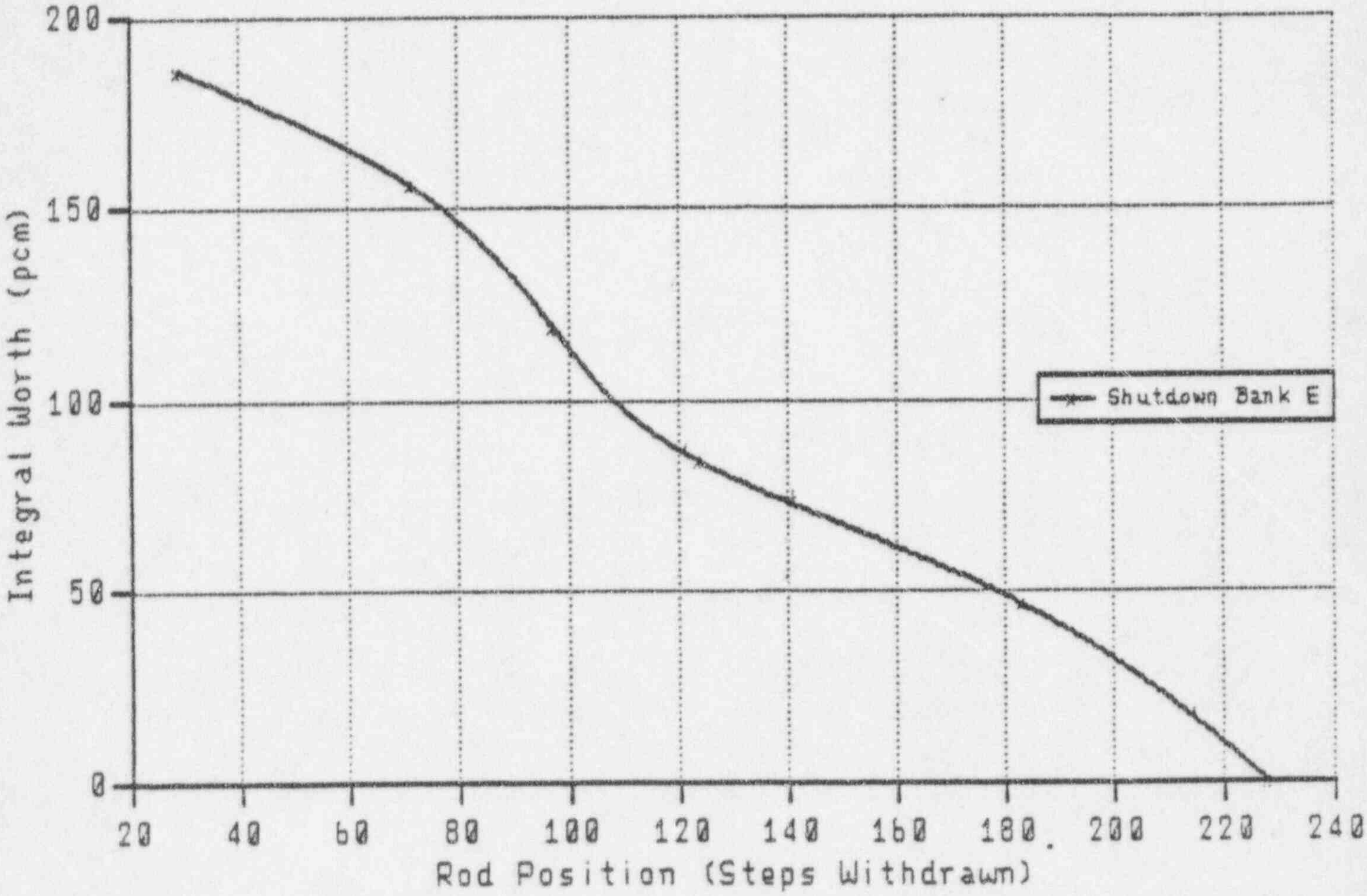


Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH SHUTDOWN BANK C	Figure 7.5-15

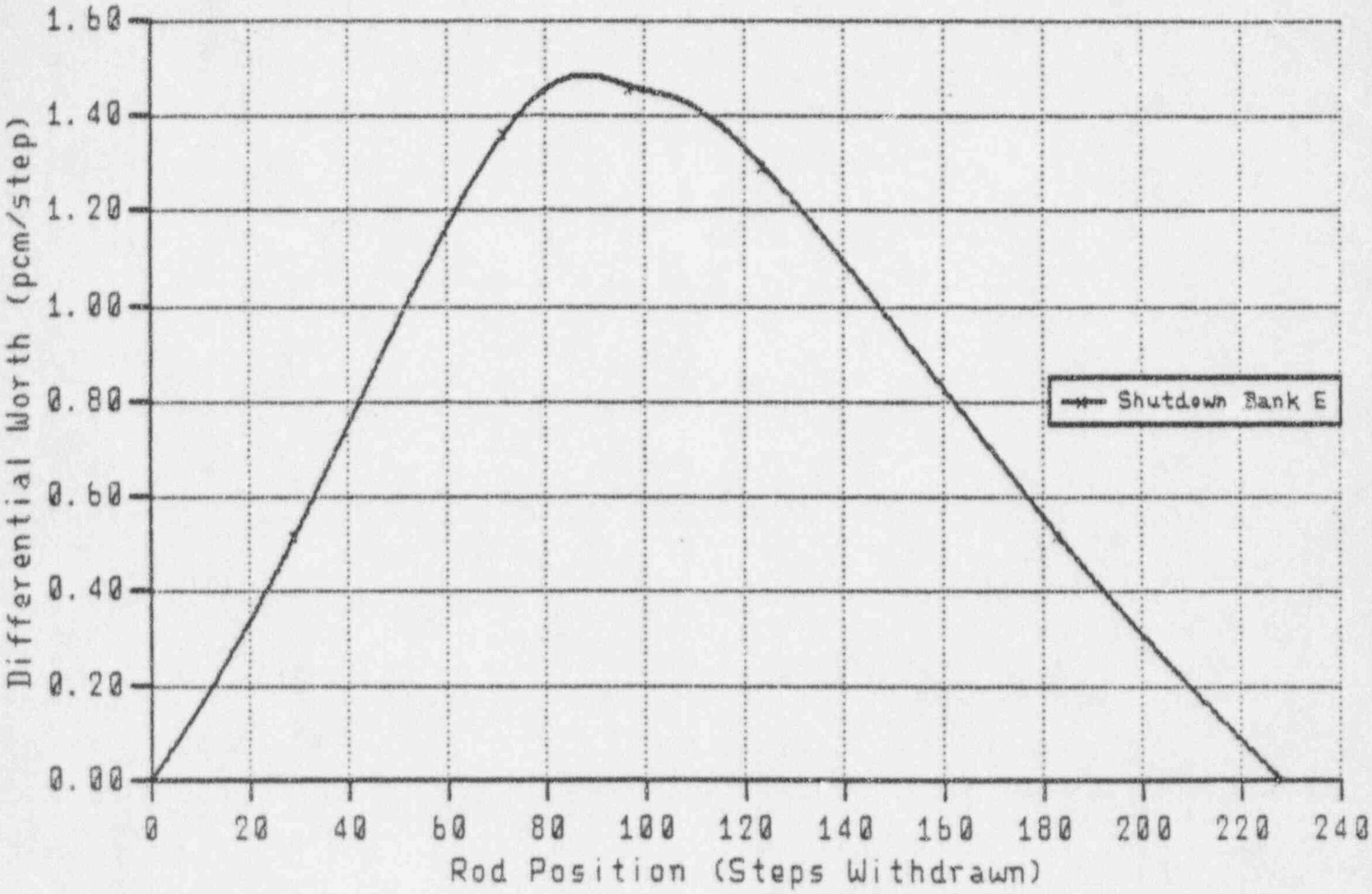


Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH SHUTDOWN BANK D	Figure 7.5-16


Differential Worth (pcm/step)
0
1
2
3
4
5
6
Rod Position (Steps Withdrawn)
0
20
40
60
80
100
120
140
160
180
200
220
240
Shutdown Bank D



Millstone Nuclear Power Station Unit No. 3	INTEGRAL CONTROL ROD WORTH SHUTDOWN BANK E	Figure 7.5-18



Millstone Nuclear Power Station Unit No. 3	DIFFERENTIAL CONTROL ROD WORTH SHUTDOWN BANK E	Figure 7.5-19

7.6 ZERO POWER FLUX MAPS

3-INT-7000 (Testing controlled by Base Procedure)

OBJECTIVE

The objective of the zero power flux maps was to measure the core power distribution at hot zero power conditions and verify that core peaking factors were within the technical specification limits.

DISCUSSION

The zero power flux maps were performed on 01-29-86 and 01-30-86. With control banks at the desired rod position, reactor power was increased to between 1 and 2 percent power and a full core flux map was performed using the moveable incore detector system. During the flux map, data was collected on the plant process computer and later analyzed using the Westinghouse Incore 3.7 computer program. The results of the analysis were compared to the core design and technical specification limits.

Flux maps were performed at the following conditions:

1. Zero Power Rod Insertion Limit (RIL): Control Bank A at 228 steps, Control Bank B at 164 steps, Control Bank C at 50 steps, and Control Bank D at 0 steps.
2. The Zero Power RIL with the control rod in core location D-12 withdrawn to 228 steps (ejected rod measurement).
3. Control Bank D fully inserted with all other Control Banks fully withdrawn.
4. All Rods Out (ARO)

All acceptance criteria were met for the zero power flux maps with the exception of the incore tilt measured in the "ARO" and "D in" flux maps. Both flux maps showed that the incore quadrant power tilt ratio design limit of 1.02 had been exceeded. As the "D in" flux map and the "ARO flux" map had been performed approximately 24 hours after the "D-12 ejected rod" flux map, it was determined that localized xenon due to the simulated ejected rod configuration had caused the tilt. A fifth flux map using 21 symmetric thimbles was performed approximately 48 hours after the "ARO" flux map to check the incore tilt at approximately 2 percent reactor power. The map showed the incore tilt to be less than the design limit of 1.02. For specific test results see Tables 7.6-1 through 7.6-5.

Test Date:	01-29-86 0315 - 0415
Map ID:	HZP RIL
Power Level:	1%
Boron Concentration:	1395 ppm
Rod Position:	CB A: 228, CB B: 165, CB C: 53, CB D: 0
Maximum Measured F_{xy}*:	1.78 @ B7
Maximum F_Q:	2.76 @ F15
Maximum $F^N_{\Delta H}$:	1.54 @ A6
Maximum $F^N_{\Delta H}$ Error (from predicted):	-5.1% @ C12
Total Core Axial Offset:	-36.5%
Maximum Quadrant Power Tilt Ratio:	1.006 { Design Limit: QPTR $\leq$ 1.02; Safety Limit: QPTR $\leq$ 1.04 }

*In locations unexcluded by Technical Specifications. F^L_{xy} = 2.04 at 30% RTP

Test Date:	01-29-86 1100 - 1200
Map ID:	HZP RIL D-12 Ejected
Power Level:	1%
Boron Concentration:	1429 ppm
Rod Position:	CB A: 228, CB B: 165, CB C: 52, CB D:0, D-12: 228
FSAR F_Q Limit:	11.5
Maximum F_Q:	7.00 @ D12
Maximum $F^N_{\Delta H}$:	4.02 @ C13

See Section 7.7 for more information on the Pseudo Ejected Rod Testing

Test Date:	01-30-86 0600 - 0700
Map ID:	CB D In; all other banks Out
Power Level:	1%
Boron Concentration:	1511 ppm
Rod Position:	CB D: 0, all other banks >209
Maximum Measured F_{xy}*:	1.86 @ J-2
Maximum F_Q:	2.81 @ G2
Maximum $F^N_{\Delta H}$:	1.705 @ J2
Maximum $F^N_{\Delta H}$ Error (from predicted):	-7.9 @ D-12
Total Core Axial Offset:	1.467%
Maximum Quadrant Power Tilt Ratio:	1.023** $\left\{ \begin{array}{l} \text{Design Limit: QPTR} \leq 1.02 \\ \text{Safety Limit: QPTR} \leq 1.04 \end{array} \right.$

*In locations unexcluded by Technical Specifications. F^L_{xy} = 2.04 at 30% RTP.

**Design limit exceeded - see text.

Test Date:	01-30-86 1030 - 1130
Map ID:	ARO HZP
Power Level:	1%
Boron Concentration:	1566 ppm
Rod Position:	CB D: 228
Maximum Measured F_{xy}*:	1.578 @ J2
Maximum F_Q:	2.36 @ J2
Maximum $F^N_{\Delta H}$:	1.45 @ J2
Maximum $F^N_{\Delta H}$ Error (from predicted):	-4.7% @ D12
Total Core Axial Offset:	2.37%
Maximum Quadrant Power Tilt Ratio:	1.023** { Design Limit: QPTR $\leq$ 1.02; Safety Limit: QPTR $\leq$ 1.04 }

*In locations unexcluded by Technical Specifications. F^L_{xy} = 1.77 at 30% RTP.
**Design limit exceeded - see text.

Millstone Nuclear Power Station Unit No. 3	CORE POWER DISTRIBUTION MEASUREMENT ARO HZP	Table 7.6-4

Test Date:	02-01-86 2146 - 2210
Map ID:	Six Pass Symmetric Thimble Tilt Check
Power Level:	2%
Boron Concentration:	NA
Rod Position:	CB C: 107
Total Core:	-38.82
Maximum Quadrant Power Tilt Ratio:	1.003

Design Limit: QPTR ≤ 1.02
Safety Limit: QPTR ≤ 1.04

7.7 PSEUDO EJECTED ROD TEST

3-INT-7000 (Testing controlled by Base Procedure)

OBJECTIVE

The objectives of this test were to:

1. Measure the worth of the highest worth inserted rod to verify that the rod worth used in the rod ejection accident analysis was conservative.
2. Verify that the core peaking factors measured by a flux map with the highest worth rod fully withdrawn from the core and the other control rods at the zero power rod insertion limit were less than the value assumed in the accident analysis.

DISCUSSION

The control rods were positioned at the zero power rod insertion limit (RIL). Through control rod motion, reactor power was increased to approximately 1 percent and a flux map was performed. This provided a base line condition for the ejected rod. The power level was then reduced to the zero power testing range and the rods were again repositioned at the zero power RIL.

The lift coils for all control bank D rods, except D-12, were then deenergized. A boration was started, and, to compensate for the negative reactivity addition, control rod D-12 was withdrawn in discrete increments. The reactivity of each withdrawal operation was measured on the reactivity computer. Once rod D-12 was fully withdrawn, core power was increased to approximately 1 percent and a flux map was performed. The power level during the performance of the first flux map in the ejected rod configuration was very unstable due to oscillations in steam generator level. As a result, this flux map was not analyzed and a second flux map was performed. This second map was used in the analysis.

RESULTS

The worth of the ejected rod and the peak FQ for the core were both less than the safety analysis limits. The results of this test are shown on Table 7.7-1.

Flux Map	D-12 Position	Measured FQ	Tech Spec Limit	Safety Analysis Limit	Location / Axial Posit	Measured Maximum $F^N_{\Delta H}$
Zero Power Rod Insertion Limit	D-12 Aligned with Control Bank D at 0 Steps	2.99	4.64	4.64	F-15 24%	1.54 in A-6
Zero Power Rod Insertion Limit D-12 Ejected	D-12 at 228 Steps	7.004	NA	11.5	D-12 37%	4.02 in C-13

Note: D-12 Rod Worth = 365.9 pcm
Predicted D-12 Worth = 491 pcm

Millstone Nuclear Power Station Unit No. 3	PSEUDO EJECTED ROD TEST RESULTS	Table 7.7-1

7.8 NATURAL CIRCULATION

3-INT-7000, Appendix 7006

OBJECTIVE

The objectives of this test were to:

1. Demonstrate plant performance capabilities and provide operators with experience and training in core heat removal by natural circulation with offsite power available. Satisfactory verification of natural circulation shall be confirmed by the establishment of stable reactor coolant loop temperatures subsequent to the initiation of the transient.
2. Verify the ability to bring the reactor to a hot zero power condition using natural circulation and the atmospheric steam dump valves.
3. Determine the length of time necessary to achieve and stabilize natural circulation.
4. Determine reactor core flow distribution.
5. Verify and monitor subcooling margin performance under natural circulation conditions. Through natural circulation, the subcooled margin in the reactor shall be maintained > 30°F. Saturation conditions shall not exist in the RCS with the exception of the pressurizer.

DISCUSSION

The test was performed on 01-30-86 with the reactor initially at slightly less than 5 percent power. The test transient was initiated by tripping all reactor coolant pumps from the control room. Monitoring of temperature indications provided verification of the establishment of natural circulation flow. After steady state conditions were verified, the reactor was brought to hot zero power conditions. Forced circulation was then reestablished.

Data collection was accomplished using a process computer with special programs, a computer trend block with data printer and the use of strip chart recorders.

Verification of satisfactory natural circulation flow was accomplished by monitoring plant parameters and the review of collected data.

RESULTS

The reactor coolant pumps were tripped at 1910. Prior to tripping the pumps, a core exit thermocouple map had been taken to document pre-transient conditions. Refer to Figure 7.8-1. Natural circulation conditions were verified to exist at 1930. This was based on stable core exit thermocouple readings as well as stable T_{hot} and T_{cold} readings. Natural circulation was maintained for approximately 30 minutes. Refer to Figure 7.8-2 for a typical core exit thermocouple map during natural circulation. Plant cooldown was then initiated using the atmospheric dumps. This continued for approximately 40 minutes during which a cooldown rate of 30.7°F/hr was achieved. During the cooldown, the lowest T_{avg} was 552.8°F which was above the test established lower limit of 551°F.

Once the cooldown was completed, the plant response to charging/letdown flow and pressurizer heater/spray valve operation was determined. At all times RCS subcooled margin (except in the pressurizer) was maintained above 30°F. When the plant response testing was completed, the reactor was shut down and forced circulation established.

During the test, the lowest T_{avg} value observed was 552.3°F which was above the limit of 551°F. The lowest subcooled margin observed during the test was approximately 49°F which was above the 30°F limit. No unexpected responses were observed during the test.

A typical plant transient response plots covering the initial phase of the test where natural circulation conditions were being established is provided as Figure 7.8-3 through 7.8-5.

Millstone
Nuclear Power Station
Unit No. 3

PRETEST CORE EXIT THERMOCOUPLE MAP NATURAL CIRCULATION TEST

Figure 7.8-1

1/30/1986
19: 2:35

AVERAGE INCORE T/C TEMP. MAP
(DEG F):
= = = = = = = = = = = = = = = =

			556.2	559.0	559.3			
	* 1.0X	558.3	560.4	556.2	557.2	557.2	556.2	
	559.3	555.1	557.2	557.2	562.5	558.5	559.3	
557.2	558.3	528.4	556.2	560.4	561.5	561.5	561.5X	
	532.6	556.2	556.2	559.3	558.3	557.2	561.5	557.2X
	556.2	556.1	557.2	558.3	557.2	555.1	557.2	
	560.4	* 1.0X	560.4	557.2	560.4	556.2	560.4	
			555.1	561.5	560.4			

A B C D E F G H J K L M N P R

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15

* T/C OUT OF SERVICE

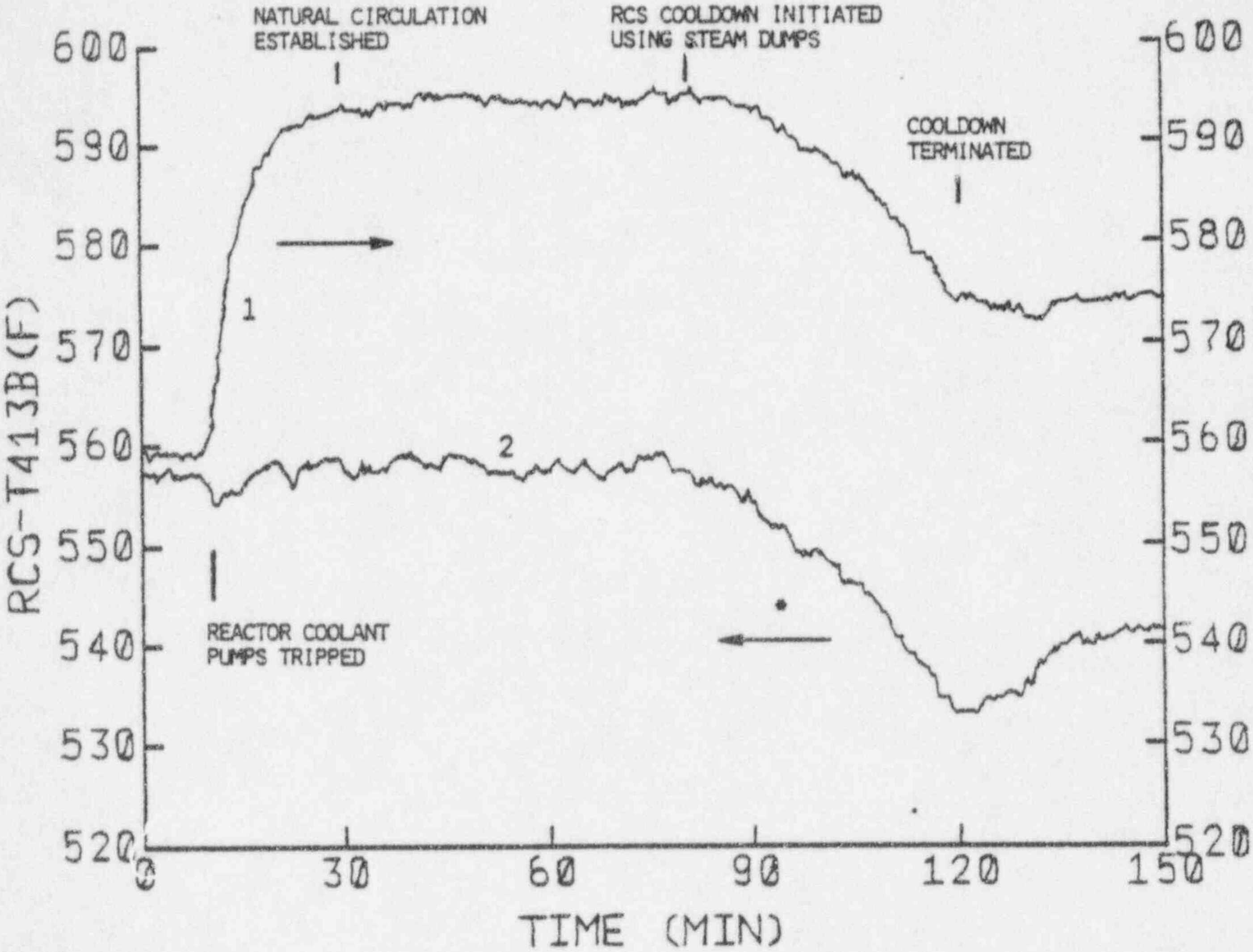
1/30/1986
20:11:52

AVERAGE INCORE T/C TEMP. MAP
(DEG F):
= = = = = = = = = = = = = = =

	1	2	4	6	8	10	12	14	15
A				600.8	597.6	597.6			
C		1.0X*	597.6	599.7	601.9	595.5	595.5	594.5	
E		598.7	593.4	596.6	595.5	600.8	596.6	597.6	
G		596.5	552.9	598.5	592.3	599.7	599.7	601.9X	
H	595.5								595.5X
J		557.2	594.5	594.5	591.2	596.5	597.7	601.9	
L		595.5	594.4	595.5	598.7	595.5	593.4	596.6	
N		598.7	1.0X*	598.7	603.0	599.7	594.5	593.4	
R				599.8	599.7	598.7			

* T/C OUT OF SERVICE

Millstone Nuclear Power Station Unit No. 3	STABLE CORE EXIT THERMOCOUPLE MAP NATURAL CIRCULATION TEST	Figure 7.8-2

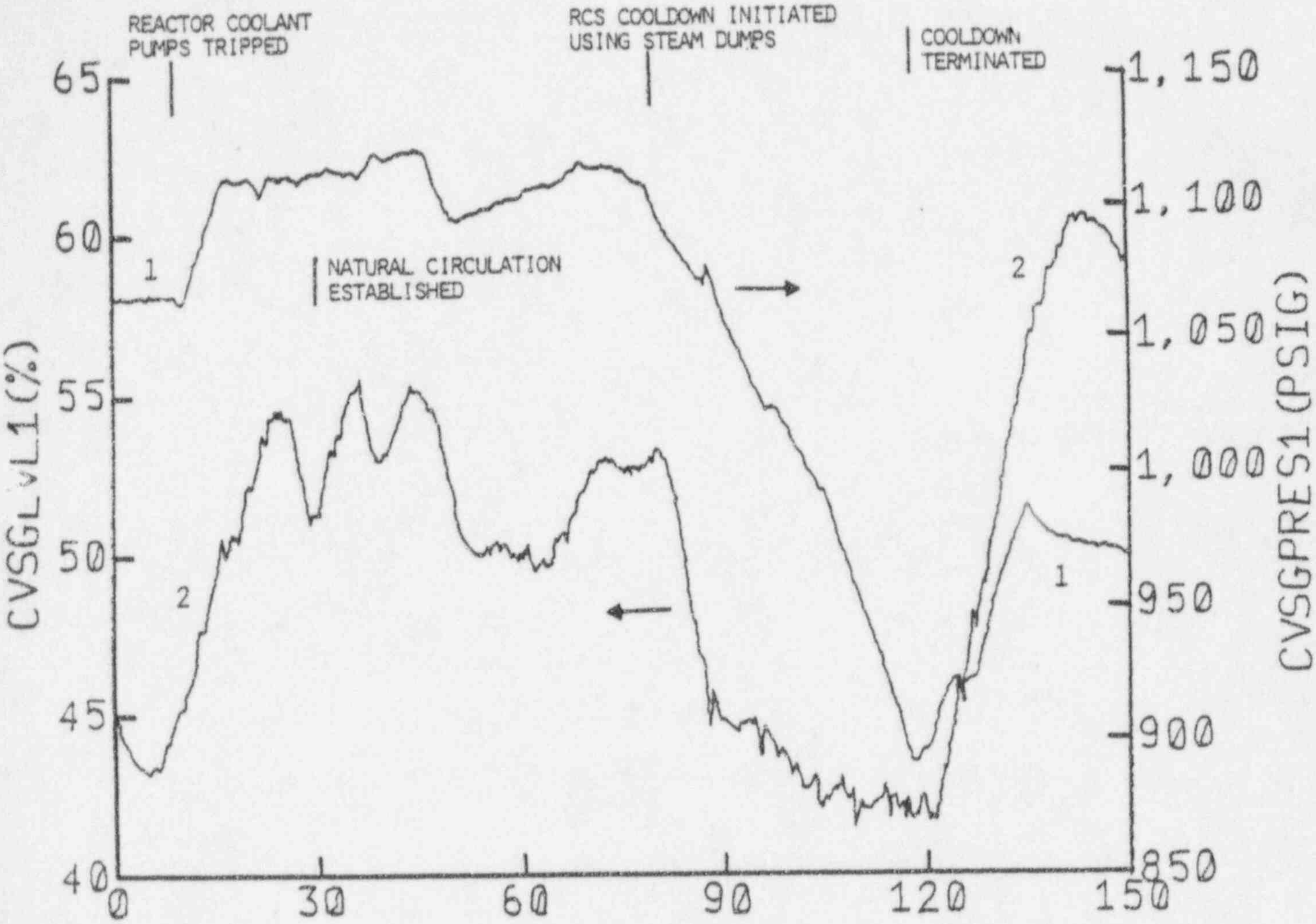


PLOT 1 - RCS LOOP 1 WIDE RANGE T_{HOT}

PLOT 2 - RCS LOOP 1 WIDE RANGE T_{COLD}

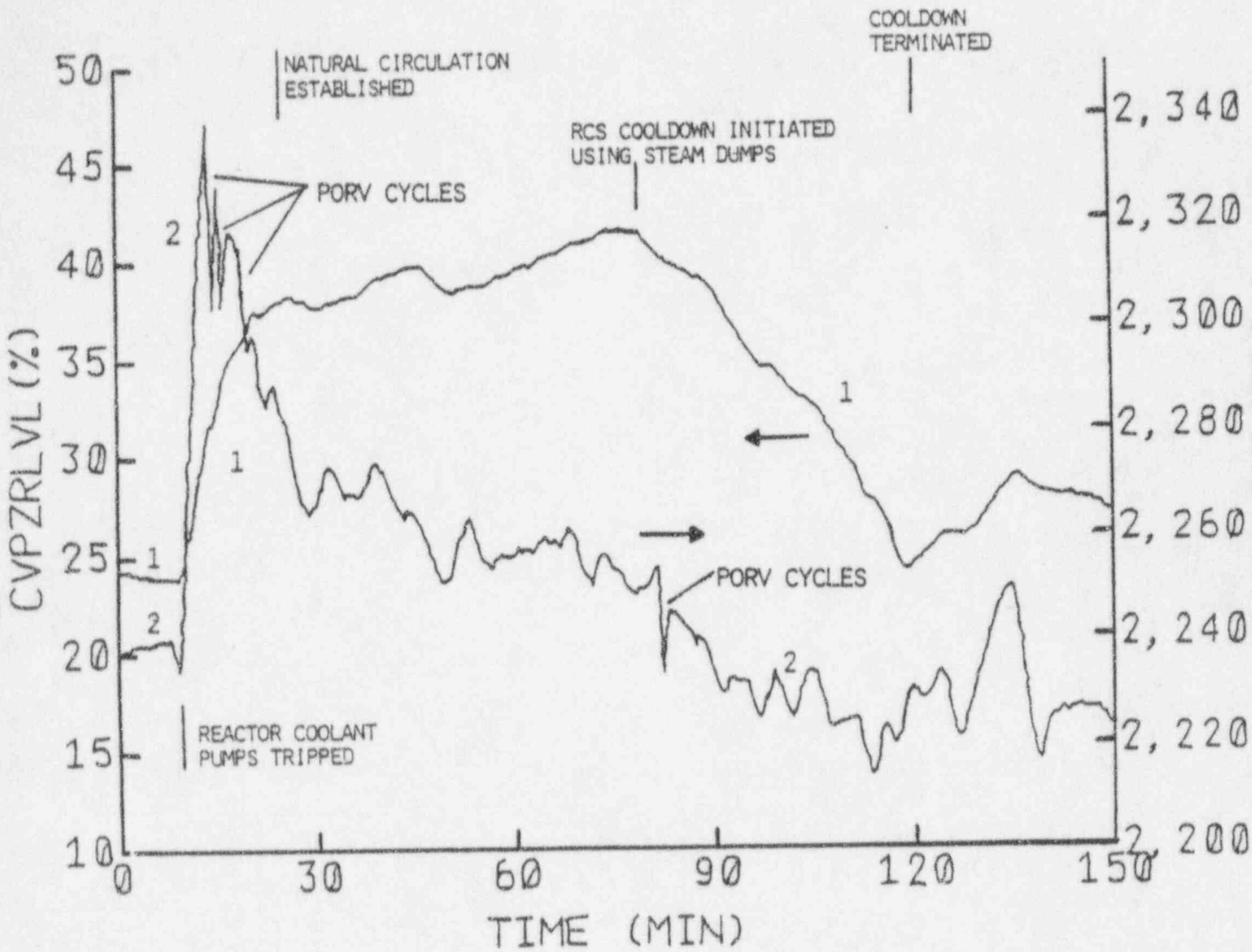
* THE ARROWS ON THIS AND OTHER PLANT TRANSIENT RESPONSE PLOTS INDICATE THE VERTICAL AXIS ASSOCIATED WITH EACH PLOT.

Millstone Nuclear Power Station Unit No. 3	TYPICAL RCS T_{HOT} & T_{COLD} NATURAL CIRCULATION TEST	Figure 7.8-3



PLOT 1 - STEAM GENERATOR 1 PRESSURE
PLOT 2 - STEAM GENERATOR 1 WIDE RANGE LEVEL

Millstone Nuclear Power Station Unit No. 3	PRESSURIZER LEVEL & PRESSURE PLOT NATURAL CIRCULATION TEST	Figure 7.8-4



PLOT 1 - PRESSURIZER LEVEL

PLOT 2 - PRESSURIZER PRESSURE

Millstone Nuclear Power Station Unit No. 3	**TYPICAL S/G LEVEL & PRESSURE NATURAL CIRCULATION TEST**	Figure 7.8-5

8.0 POWER ASCENSION TESTING SUMMARY

3-INT-8000

The base procedure controlled the sequence of events during initial power operation. Most of the testing occurred at power level plateaus of 30, 50, 75, 90, and 100 percent. At each of these power levels, both the primary and secondary systems (plus auxiliaries) were observed for operation within design specifications. Plant and test instruments were used to verify proper operation, not only at steady-state conditions, but also for selected transients. Prior to proceeding from one plateau to another, the test data was reviewed to assure operation at a higher power level was permissible. This test established plant conditions necessary for specific tests, called for individual power ascension tests to be performed, provided direction when in transitory periods between individual tests, and provided restoration requirements as needed. Major testing accomplished included the following:

- Instrumentation and controls systems calibration and grooming
- Plant performance verification (steady-state)
- 10 percent load swing
- Reactor trip and shutdown outside the control room
- Large load reduction
- Loss of power trip
- Generator trip from 100 percent
- MSIV closure

The power ascension test sequence was accomplished over the period from 01-31-86 to 04-21-86.

8.1.1 REACTOR COOLANT SYSTEM FLOW MEASUREMENT
3-INT-8000, Appendix 8015

OBJECTIVE

The objectives of the Reactor Coolant System Flow Measurement were:

1. Determine the Reactor Coolant System (RCS) flow utilizing a precision heat balance.
2. Calculate correction factors for the RCS flow elbow taps in order to correlate their indications of flow with the precision heat balance flow.
3. Ensure that adequate Reactor System flow is present as required by Technical Specifications.

DISCUSSION

With the reactor plant operating at a 50 percent power level, a precision heat balance was performed to determine exact reactor thermal power. Reactor power was measured taking high accuracy readings from the protection cabinets and analyzed in accordance with a flow uncertainty analysis performed for this test. An overall uncertainty of 2.1 percent for reactor coolant flow was achieved with this method. Based on this 50 percent power level, the elbow tap instrumentation was normalized. This test was repeated at 90 percent power. The 50 percent preadjustment data and the post-adjustment flow data taken at 90 percent power are presented in Table 8.1.1-1.

RESULTS

All acceptance criteria were met. RCS flow was verified to be above the Technical Specification required level of 387,500 gpm (T.S. 3.2.3.1.a). Based on the RCS flow data taken at 90 percent power level, no adjustment to the RCS flow instrumentation was required.

LOOP	50% POWER LEVEL		90% POWER LEVEL	
	MEASURED	INDICATED	MEASURED	INDICATED
1	107.26%	F414 : 101.95 F415 : 103.78 F416 : 102.78	110.1%	F414 : 106.9 F415 : 107.28 F416 : 107.25
2	111.0%	F424 : 102.7 F425 : 103.9 F426 : 102.08	109.7%	F424 : 102.7 F425 : 108.7 F426 : 110.0
3	108.5%	F434 : 103.4 F435 : 105.78 F436 : 103.18	108.1%	F434 : 103.4 F435 : 108.35 F436 : 108.5
4	104.5%	F444 : 102.58 F445 : 105.38 F446 : 94.0	104.8%	F444 : 104.25 F445 : 103.68 F446 : 103.8

L.2 POWER COEFFICIENT
3-INT-8000, Appendix 8020

OBJECTIVE

The objective of this test was to verify the Westinghouse Nuclear Design Report prediction of the doppler only power coefficient.

DISCUSSION

At the 30, 50, 75, 90 and 100 percent power plateaus, the reactor was allowed to attain equilibrium xenon. Once steady state conditions were achieved, thermal power was measured and rod control was placed in Manual. Then, using the turbine controller, a series of step load decreases/increases of approximately 40 MWE each were made. During these transients, reactor power, ΔT, and T_{avg} were recorded. This data was used to calculate, at each power level, a doppler only power coefficient verification factor (C^M) which was compared to the Westinghouse Nuclear Design Report predicted doppler only power coefficient verification factor (C^P).

RESULTS

The results of the test are listed in Table 8.1.2-1. The acceptance criteria requiring that the absolute difference between C^M and C^P be less than 0.5°F/% power was met.

POWER LEVEL (%)	C^M (°F/% POWER)	C^P (°F/% POWER)	ABSOLUTE DIFFERENCE C^M vs C^P (°F/% POWER)
30	-2.75	2.66	0.09
50	-1.63	1.66	0.03
75	-1.05	1.13	0.08
90	-0.91	0.96	0.05
100	-0.90	0.90	0

Acceptance Criteria: Absolute difference between C^M and C^P is < 0.5 °F/% Power

Millstone Nuclear Power Station Unit No. 3	DOPPLER ONLY POWER COEFFICIENT VERIFICATION	Table 8.1.2-1

8.1.3 RCS BORON MEASUREMENT

3-INT-8000, Appendix 8031

OBJECTIVE

The objective of this test was to perform a core reactivity balance in order to support comparison of the actual full power equilibrium RCS boron concentration to the Westinghouse Nuclear Design Report predicted value.

DISCUSSION

The test was performed on 04-19-86. With the plant operating in a steady state condition at a 100 percent power level with control bank D at 210 steps and equilibrium xenon, three RCS boron samples were taken. In addition, primary side data necessary to support calculation of a core reactivity balance were also taken. A plant calorimetric was then performed to accurately determine thermal power output. Using this information, a core reactivity balance was performed and used to correct the measured RCS boron concentration for actual T_{ref}, xenon, samarium and rod position. The corrected value was then compared to the predicted value of 1058 ppm.

RESULTS

The corrected RCS boron concentration was required to be within ± 1% ΔK/K of the predicted concentration. The corrected concentation was determined to be 1071 ppm which was within 0.124% ΔK/K of predicted. The acceptance criteria was met.

8.1.4 CORE POWER DISTRIBUTION MEASUREMENT

3-INT-8000 (Testing controlled by Base Procedure)

OBJECTIVE

The objective of this test was to measure the core power distribution at various core power levels in order to verify the measured peaking factors were within the limits specified in Technical Specifications and the Westinghouse Nuclear Design Report predictions.

DISCUSSION

Testing was conducted over the period of 02-17-86 to 04-28-86. A total of seven full core maps were taken and analyzed - one at 30, 50, and 90 percent power and two at 75 and 100 percent power. All flux maps were analyzed using the Westinghouse Incore 3.7 computer program.

RESULTS

The results of the testing is provided in Tables 8.1.4-1 through 8.1.4-7. All the test acceptance criteria were met with the exception of the 30 percent power level measured F_{xy} value of 1.56 which exceeded the stated Technical Specification F_{xy}^{RTP} limit of 1.55. Review by Reactor Engineering indicated that the measured F_{xy} value did not exceed the Technical Specification F_{xy}^{L} limit of 1.768. Considering this and since an additional full core flux map was to be taken prior to increasing power an additional 20 percent as required by Technical Specifications, the F_{xy} was considered acceptable. All subsequent measured F_{xy} values were within the Technical Specification F_{xy}^{RTP} limits.

Test Date:	02-17-86
Map ID:	30% Power Flux Map
Power Level:	1013 MW_T
Boron Concentration:	1303 ppm
Rod Position:	CB D: 184
Maximum Measured F_{xy}*:	1.56 @ B7
Maximum F_Q:	2.115 @ B7
Maximum $F^N_{\Delta H}$:	1.41 @ B7
Maximum $F^N_{\Delta H}$ Error (from predicted):	3.3% @ G11
Total Core Axial Offset:	-3.924

Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core	
Quadrant 1	0.9984	0.9984	
Quadrant 2	0.9900	1.0018	Design Limit: ≤ 1.02
Quadrant 3	1.0077	1.0054	Safety Limit: ≤ 1.04
Quadrant 4	0.9950	0.9941	

*In locations unexcluded by Technical Specifications.

NOTE: The F_{xy}^{RTP} limit of 1.55 was exceeded; however the F_{xy}^{L} limit for 30% RTP of 1.768 was not exceeded. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Millstone Nuclear Power Station Unit No. 3	**CORE POWER DISTRIBUTION MEASUREMENT** **30 PERCENT POWER**	Table 8.1.4-1

Test Date:	03-18-86
Map ID:	50% Power ARO
Power Level:	1700 MW_T
Boron Concentration:	1217 ppm
Rod Position:	CB D: 216
Maximum Measured F_{xy}*:	1.51 @ B7
Maximum F_Q:	2.014 @ B9
Maximum $F^N_{\Delta H}$:	1.386 @ B7
Maximum $F^N_{\Delta H}$ Error (from predicted):	4% @ E8
Total Core Axial Offset:	-2.616

Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core	
Quadrant 1	0.9985	0.9987	
Quadrant 2	0.9979	0.9999	Design Limit: ≤ 1.02
Quadrant 3	1.0106	1.0083	Safety Limit: ≤ 1.04
Quadrant 4	0.9930	0.9931	

*In locations unexcluded by Technical Specifications.

NOTE: F_{xy}^{RTP} limit of ≤ 1.55 was met. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Millstone Nuclear Power Station Unit No. 3	CORE POWER DISTRIBUTION MEASUREMENT 50 PERCENT POWER	Table 8.1.4-2

Test Date:	03-27-86
Map ID:	75% Power ARO
Power Level:	2589.0 MW_T
Boron Concentration:	1125 ppm
Rod Position:	CB D: 222
Maximum Measured F_{xy}*:	1.48 @ B7
Maximum F_Q:	2.008 @ B7
Maximum $F^N_{\Delta H}$:	1.368 @ B7
Maximum $F^N_{\Delta H}$ Error (from predicted):	2.4% @ G7
Total Core Axial Offset:	-4.733

Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core	
Quadrant 1	0.9988	0.9989	
Quadrant 2	1.0024	1.0018	Design Limit: QPTR ≤ 1.02
Quadrant 3	1.0049	1.0048	Safety Limit: QPTR ≤ 1.04
Quadrant 4	0.9937	0.9945	

*In locations unexcluded by Technical Specifications

NOTE: RCS Flow = 104%. F^{RTP}_{xy} limit of < 1.55 was met. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Millstone Nuclear Power Station Unit No. 3	CORE POWER DISTRIBUTION MEASUREMENT 75 PERCENT POWER - ARO	Table 8.1.4-3

Test Date:	04-14-86
Map ID:	75% Power Incore/Excore Cross Calibration
Power Level:	2566.0 MW_T
Boron Concentration:	1125 ppm
Rod Position:	CB D: 210
Maximum Measured F_{xy}*:	1.48 @ B7
Maximum F_Q:	1.988 @ B7
Maximum $F^N_{\Delta H}$:	1.364 @ B7
Maximum $F^N_{\Delta H}$ Error (from predicted):	3.8% @ G7
Total Core Axial Offset:	-3.05

Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core	
Quadrant 1	0.9967	0.9947	
Quadrant 2	0.9984	0.9994	Design Limit: QPTR ≤ 1.02
Quadrant 3	1.0053	1.0055	Safety Limit: QPTR ≤ 1.04
Quadrant 4	0.9996	1.0005	

*In locations unexcluded by Technical Specifications.

NOTE: RCS Flow = 104%. F^{RTP}_{xy} limit of ≤ 1.55 was met. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Millstone Nuclear Power Station Unit No. 3	CORE POWER DISTRIBUTION MEASUREMENT 75 PERCENT POWER	Table 8.1.4-4

Test Date:	04-17-86
Map ID:	90% Power ARO
Power Level:	3050.0 MW_T
Rod Position:	CB D: 202
Maximum Measured F_{xy}*:	1.49 @ B7
Maximum F_Q:	2.04 @ B7
Maximum $F^N_{\Delta H}$:	1.36 @ B7
Maximum $F^N_{\Delta H}$ Error (from predicted):	-2.8% @ C12
Total Core Axial Offset:	-8.89

Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core
Quadrant 1	0.9978	0.9975
Quadrant 2	0.9970	0.9995
Quadrant 3	1.0074	1.0078
Quadrant 4	0.9978	0.9953

Design Limit: QPTR $\leq$ 1.02
Safety Limit: QPTR $\leq$ 1.04

*In locations unexcluded by technical specifications

NOTE: Burnup = 670 MWD/MTU. RCS Flow = 107%. F^{RTP}_{xy} limit of $\leq$ 1.55 was met. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Millstone Nuclear Power Station Unit No. 3	CORE POWER DISTRIBUTION MEASUREMENT 90 PERCENT POWER	Table 8.1.4-5

Test Date:	04-19-86
Map ID:	100% Power ARO
Power Level:	3411.0 MW_T
Boron Concentration:	1078 ppm
Rod Position:	CB D: 213
Maximum Measured F_{xy}*:	1.47 @ B7
Maximum F_Q:	1.99 @ B7
Maximum $F^N_{\Delta H}$:	1.35 @ B7
Maximum $F^N_{\Delta H}$ Error (from predicted):	3.7% @ R11
Total Core Axial Offset:	-7.28

Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core	
Quadrant 1	0.9965	0.9973	
Quadrant 2	0.9973	0.9993	Design Limit: QPTR ≤ 1.02
Quadrant 3	1.0068	1.0080	Safety Limit: QPTR ≤ 1.04
Quadrant 4	0.9995	0.9955	

*In locations unexcluded by Technical Specifications.

NOTE: Burnup = 760 MWD/MTU. F^{RTP}_{xy} limit of ≤ 1.55 was met. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Test Date:	04-28-86	
Map ID:	100% Power ARO	
Power Level:	3410.0 MW_T	
Boron Concentration:	1090 ppm	
Rod Position:	CB D: 212	
Maximum Measured F_{xy}*:	1.47 @ B7	
Maximum F_Q:	1.98 @ B7	
Maximum $F^N_{\Delta H}$:	1.35 @ B7	
Maximum $F^N_{\Delta H}$ Error (from predicted):	4.5% @ R11	
Total Core Axial Offset:	-6.88	
Quadrant Power Tilt Ratios:	Top Half of Core	Bottom Half of Core
Quadrant 1	0.9979	0.9974
Quadrant 2	0.9985	0.9978
Quadrant 3	1.0060	1.0070
Quadrant 4	0.9971	0.9970

Design Limit: QPTR ≤ 1.02
Safety Limit: QPTR ≤ 1.04

*In locations unexcluded by Technical Specifications.

NOTES: Burnup = 977 MWD/MTU. RCS Flow = 107%. F^{RTP}_{xy} limit of ≤ 1.55 was met. $F^N_{\Delta H}$ was less than the Technical Specification limit of 1.49 at RTP.

Millstone Nuclear Power Station Unit No. 3	CORE POWER DISTRIBUTION MEASUREMENT 100 PERCENT POWER - MAP 2	Table 8.1.4-7

8.2.1 OPERATIONAL ALIGNMENT VERIFICATION OF NUCLEAR INSTRUMENTATION 3-INT-8000, Appendix 8002

OBJECTIVE

The objectives of this test were to:

1. Calibrate the excore power range instrumentation utilizing the power level calculation from the plant process computer calorimetric.
2. Determine overlap indication between the Source Range (SR), Intermediate Range (IR) and Power Range (PR) channels.
3. Verify that PR currents versus reactor power exhibit linear response.

DISCUSSION

The test was conducted on 02-15-86, 3-15-86, 3-17-86, 3-26-86, 4-16-86 and 4-18-86 with the plant at 30, 40, 50, 75, 90 and 100 percent power levels, respectively. At each plateau, plant calorimetrics were performed in order to obtain data for PR adjustments. In addition, at 30 percent power, the flux deviation alignment was verified by manually manipulating the output of a single channel and observing the flux level at which the deviation alarm occurred.

Between the 75 and 90 percent test plateaus, PR detectors N42 and N44 were replaced when water was discovered in their wells in the neutron shield tank. When the water was found in the wells, an inservice leak test was performed on the Neutron Shield Tank (NST). No leaks were found and it was therefore postulated that the water entered the wells during NST fill or testing operations. The original N42 and N44 detectors had exhibited higher detector current than those of N41 and N43, due to the additional moderation from the water in the N42 and N44 wells. The original detectors exhibited normal response to power level changes and trips and good overlap with the intermediate range channels. After replacing N42 and N44, the PR checks were again performed at 30, 40, 50 and 75 percent power levels. The initial tests at 90 and 100 percent power levels were then performed.

Throughout the test IR and PR output data was recorded and evaluated to ensure proper detector overlap. SR and IR overlap data taken during initial criticality was reviewed in order to ensure at least one decade of overlap existed.

RESULTS

The required overlap of at least one decade between SR to IR and IR to PR was successfully verified. After adjustments, all PR channels consistently agreed within 2 percent of the secondary calorimetric reactor power level. All PR channels exhibited a linear response in the power range.

8.2.2 OPERATIONAL ALIGNMENT OF PROCESS TEMPERATURE INSTRUMENTATION 3-INT-8000, Appendix 8004

OBJECTIVE

The objective of this test was to acquire data to align the ΔT and T_{avg} process instrumentation such that individual instrumentation channels are consistent with each other and consistent with core thermal power.

DISCUSSION

The test was performed on 02-15-86, 03-17-86, 03-26-86, 04-16-86 and 04-18-86 with the plant at power levels of 30, 50, 75, 90 and 100 percent, respectively. Process control system T_{hot} and T_{cold} data was collected during thermal equilibrium at listed power levels. Using this data, full load T_{avg} and ΔT values were extrapolated and used to align the process control system T_{avg} and ΔT loops at each power level.

RESULTS

The ΔT and T_{avg} process loops were successfully aligned. At 100 percent each channel's average ΔT was within the acceptance criteria of 55°F to 60°F. The ΔT values were 55.00°F, 55.02°F, 56.03°F, and 55.65°F for loops 1, 2, 3 and 4, respectively. In addition, each channel's T_{avg} was below the high limit of 587.1°F. The values were 585.77°F, 584.53°F, 585.40°F, and 585.30°F for loops 1, 2, 3 and 4, respectively.

All acceptance criteria were based on the Westinghouse Precautions, Limitations and Setpoints (PLS) document.

8.2.3 CALIBRATION OF STEAM FLOW AND FEEDWATER FLOW
3-INT-8000, Appendix 8003

OBJECTIVE

To determine recalibration data for Steam Flow Transmitters to conform to actual plant conditions as determined by the calorimetric program.

DISCUSSION

The test was performed on 02-15-86, 03-17-86, 03-26-86 and 04-18-86 with the plant at 30, 50, 75, and 100 percent power levels, respectively. During the test, process control system parameters for feedwater flow, steam flow, and steam pressure were recorded. Using this data, the process control loops were then adjusted so that steam flow matched feedwater flow during steady state conditions.

As a first step, based on test data, corrected steam flow transmitter ranges were calculated and used to recalibrate the steam flow transmitters. Then the process control system was adjusted to its original settings so that its alignment matched the new transmitter calibration. This process was repeated at each of the power plateaus. Since this procedure was strictly a data collection and adjustment evolution, there were no acceptance criteria.

RESULTS

Steam flow, feedwater flow and steam pressure data was collected and used to adjust the steam flow instrumentation at each of the power plateaus. Based on data obtained from the test, the steam flow transmitters were recalibrated following the completion of the Power Ascension testing program. All activities were successfully completed.

8.2.4 INCORE/EXCORE NUCLEAR INSTRUMENTATION CROSS-CALIBRATION
3-INT-8000, Appendix 8028

OBJECTIVE

The objective of this test was to determine the relationship between the axial offset determined by an incore flux map and the axial offset as indicated by the excore power range nuclear instrumentation. Using the measured incore to excore relationship, calibration factors were determined for the excore power range neutron detectors and the Tilting Factors computer program.

DISCUSSION

The test was performed during the period on 03-28-86 and 04-14-86 at a power level of 75 percent. This test consisted of taking a series of incore flux maps over several different axial flux conditions. The measured incore axial offset was then compared to the axial offset determined from the upper and lower excore detector currents which had been measured at the time of the flux maps.

The first calibration was performed at a 50 percent power level. This was to determine the preliminary calibration factors for the excore detectors prior to exceeding 50 percent power and to provide initial calibration of excore detectors. During this time, two full core flux maps and two quarter core flux maps were performed over a 15 percent change in axial offset. The results of the preliminary calibration are shown in Table 8.2.4-1. This data indicated that the excore power range channels were capable of being calibrated. However, the results for channels N42 and N44 were of concern in that they did not produce the expected test results as seen in channels N41 and N43. As the excore detectors sit inside dry wells in a water-filled, natural circulation cooled neutron shield tank, it was felt that the unexpected test results could have been due to temperature variations within the tank. Based on this proposed explanation, the decision was made to increase power to 75 percent and to perform the test at 75 percent power or above as required by technical specifications.

At the 75 percent power plateau, three full core flux maps and five quarter core flux maps were performed over a 23 percent swing in axial offset. The plot of axial offset versus time is shown in Figure 8.2.4-1. The results of the test are shown in Table 8.2.4-2 and in Figures 8.2.4-2 through 8.2.4-5. These results once again showed that the detectors were capable of being calibrated but the data for detectors N42 and N44 did not produce the expected results in that the current for detector N42 Bottom was approximately twice the current of N42 Top and the current for detectors N44 Top and Botton were approximately 10 times higher than the current found on channels N41, N42, and N43.

Based on this anomalous data, a decision was made to check the excore detectors in containment. This was performed during a cold shutdown for steam generator water chemistry cleanup prior to increasing power above 75 percent. A series of electronic checks had already been made on the excore detector channels from the instrument racks. No problems had been noted. During the cold shutdown, the detectors were checked for loose connections and general detector condition inside the detector wells. Inspection of the detector wells indicated that the well for channel N42 contained approximately 1.5 feet of water and the N44 well contained approximately 3.5 feet of water. In addition, the aluminum can that houses the detectors for channel N44 was full of water. The other six excore detector wells were examined and found to be dry.

After this discovery, the detectors for channels N42 and N44 were removed from the detector wells and a leak test was performed on the neutron shield tank. The leak test applied a pressure of 15 psig to the tank and was held for 24 hours. The test results showed no leakage of water into the detector wells and it was subsequently decided that, during the initial fill of the neutron shield tank, water had spilled out of the tank manways on the top of the tank and into the detector wells. Although the detector wells were inspected after the initial fill, the water was evidently not noticed. The detector wells were pumped out, dried and two new power range detectors were installed. As channel N44 was used as the input channel to the reactivity computer during Low Power Physics Tests (LPPT), an evaluation was done on the acceptability of the physics test results. Since testing of channel N44 indicated no damage had been done to the detector, and since previous incore/excore cross-calibration test results showed the detector to be capable of being calibrated, it was determined that LPPT results were still valid.

The third incore/excore cross-calibration was performed during the power ascension following the outage. Prior to startup, the two new detectors which had been installed were adjusted using the calibration factors determined in the previous incore/excore cross-calibration using symmetrically opposite detectors. Channel N42 was adjusted using channel N41's calibration factors and channel N44 was adjusted using N43's calibration factors. At 50 percent power a check of Quadrant Power Tilt Ratio (QPTR) and excore axial flux difference was performed. The indicated QPTR was less than the technical specification limit of 1.02 and greatest difference between the highest and lowest indicated axial flux difference channel was less than 2 percent. Power was then increased to 75 percent and the third set of incore/excore cross-calibration measurements were taken.

The third calibration consisted of two full core flux maps and two quarter core flux maps over an 18 percent change in axial flux offset. The plot of axial offset versus time is shown in Figure 8.2.4-6. Additional quarter core flux maps and one full core flux map had been planned; however, it became necessary to reduce power after the second quarter core flux map due to an oil leak in the turbine generator electro-hydraulic control system. The data from the four flux maps was analyzed. The results for the two detectors which had not been replaced was consistent with the results of the previous calibration and the results for the two new detectors was consistent with the expected results. The results of the third calibration are shown on Table 8.2.4-3 and Figures 8.2.4-7 through 8.2.4-10.

RESULTS

The objectives of the test were met. As discussed above, problems with power range detectors N42 and N44 were corrected. The performance of the excore detector system has been satisfactory with the original N41 and N43 detectors and the replacement N42 and N44 units.

Detector 41 Calibration Curves:

$$Incore_{AO} = .934(Excore_{AO}) + 6.35$$

$$Upper_{CURR} = .575(\Delta q) + 113.65$$

$$Lower_{CURR} = -.785(\Delta q) + 123.95$$

Detector 42 Calibration Curves:

$$Incore_{AO} = 1.318(Excore_{AO}) + 43.5$$

$$Upper_{CURR} = .910(\Delta q) + 105.04$$

$$Lower_{CURR} = -1.74(\Delta q) + 209.15$$

Detector 43 Calibration Curves:

$$Incore_{AO} = 1.357(Excore_{AO}) + 4.6$$

$$Upper_{CURR} = .979(\Delta q) + 117.13$$

$$Lower_{CURR} = -.746(\Delta q) + 125.64$$

Detector 44 Calibration Curves:

$$Incore_{AO} = 1.357(Excore_{AO}) + 0.416$$

$$Upper_{CURR} = 9.023(\Delta q) + 1157.9$$

$$Lower_{CURR} = -7.868(\Delta q) + 1165.8$$

Notes: Number of data points	4
Axial Flux Difference swing	8.2%
Duration	3-18-86 to 3-24-86 @ 50% RTP

Detector 41 Calibration Curves:

$$Incore_{AO} = 1.355(Excore_{AO}) + 6.1$$

$$Upper_{CURR} = .833(\Delta q) + 111.5$$

$$Lower_{CURR} = -.812(\Delta q) + 122.4$$

Detector 42 Calibration Curves:

$$Incore_{AO} = 1.420(Excore_{AO}) + 43.1$$

$$Upper_{CURR} = .800(\Delta q) + 108.77$$

$$Lower_{CURR} = -1.79(\Delta q) + 203.45$$

Detector 43 Calibration Curves:

$$Incore_{AO} = 1.380(Excore_{AO}) + 3.78$$

$$Upper_{CURR} = .894(\Delta q) + 118.47$$

$$Lower_{CURR} = -.852(\Delta q) + 125.30$$

Detector 44 Calibration Curves:

$$Incore_{AO} = 1.520(Excore_{AO}) + 2.50$$

$$Upper_{CURR} = 7.15(\Delta q) + 1126.76$$

$$Lower_{CURR} = -7.94(\Delta q) + 1165.04$$

Notes: Number of data points	9
Axial Flux Difference swing	25%
Duration	17 hours

Detector 41 Calibration Curves:

$$Incore_{AO} = 1.340(Excore_{AO}) + 6.35$$

$$Upper_{CURR} = .882(\Delta q) + 107.9$$

$$Lower_{CURR} = -.768(\Delta q) + 118.95$$

Detector 42 Calibration Curves:

$$Incore_{AO} = 1.350(Excore_{AO}) + 8.72$$

$$Upper_{CURR} = .834(\Delta q) + 106.6$$

$$Lower_{CURR} = -.838(\Delta q) + 121.5$$

Detector 43 Calibration Curves:

$$Incore_{AO} = 1.340(Excore_{AO}) + 3.85$$

$$Upper_{CURR} = .834(\Delta q) + 114.64$$

$$Lower_{CURR} = -.795(\Delta q) + 121.76$$

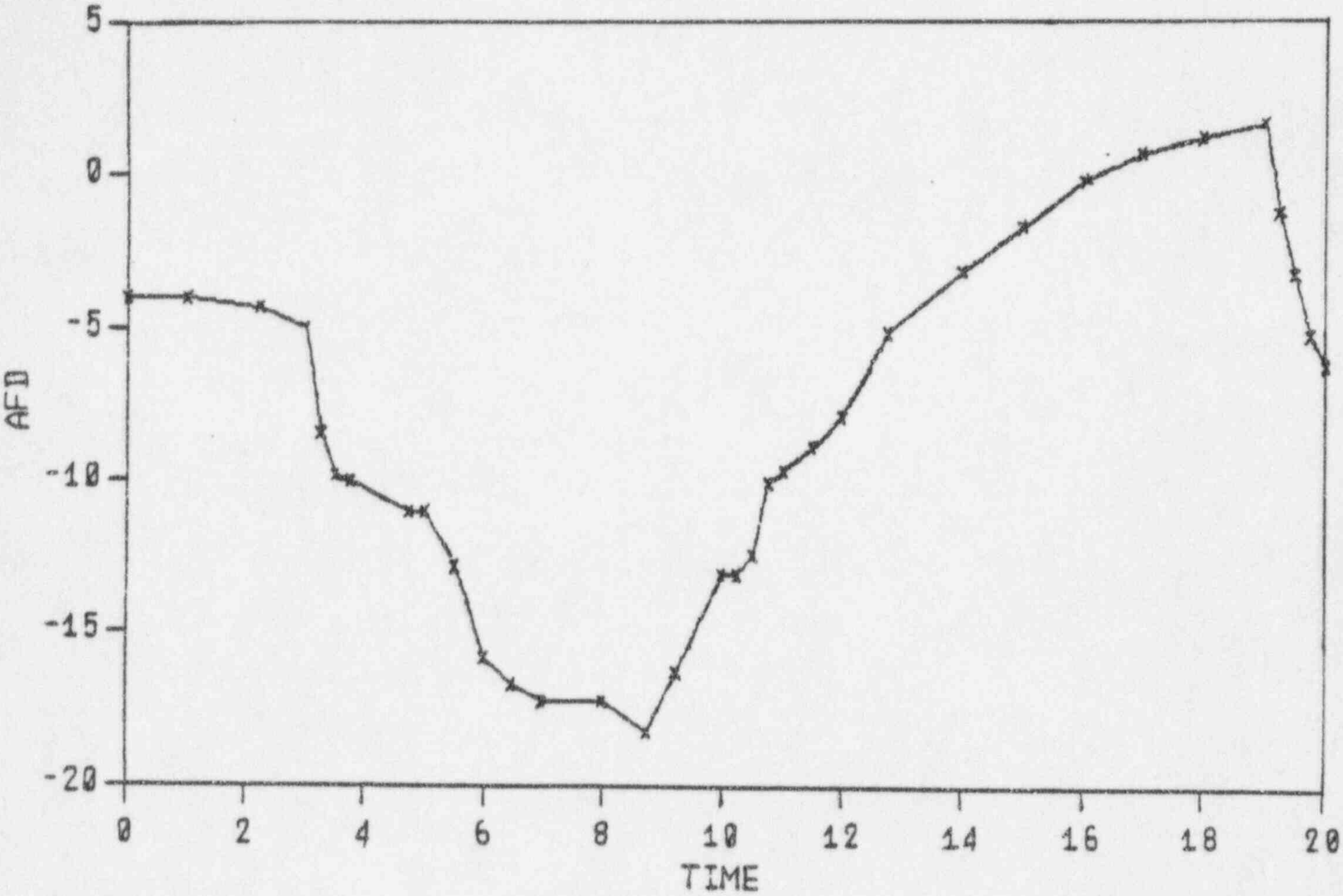
Detector 44 Calibration Curves:

$$Incore_{AO} = 1.340(Excore_{AO}) + 19.8$$

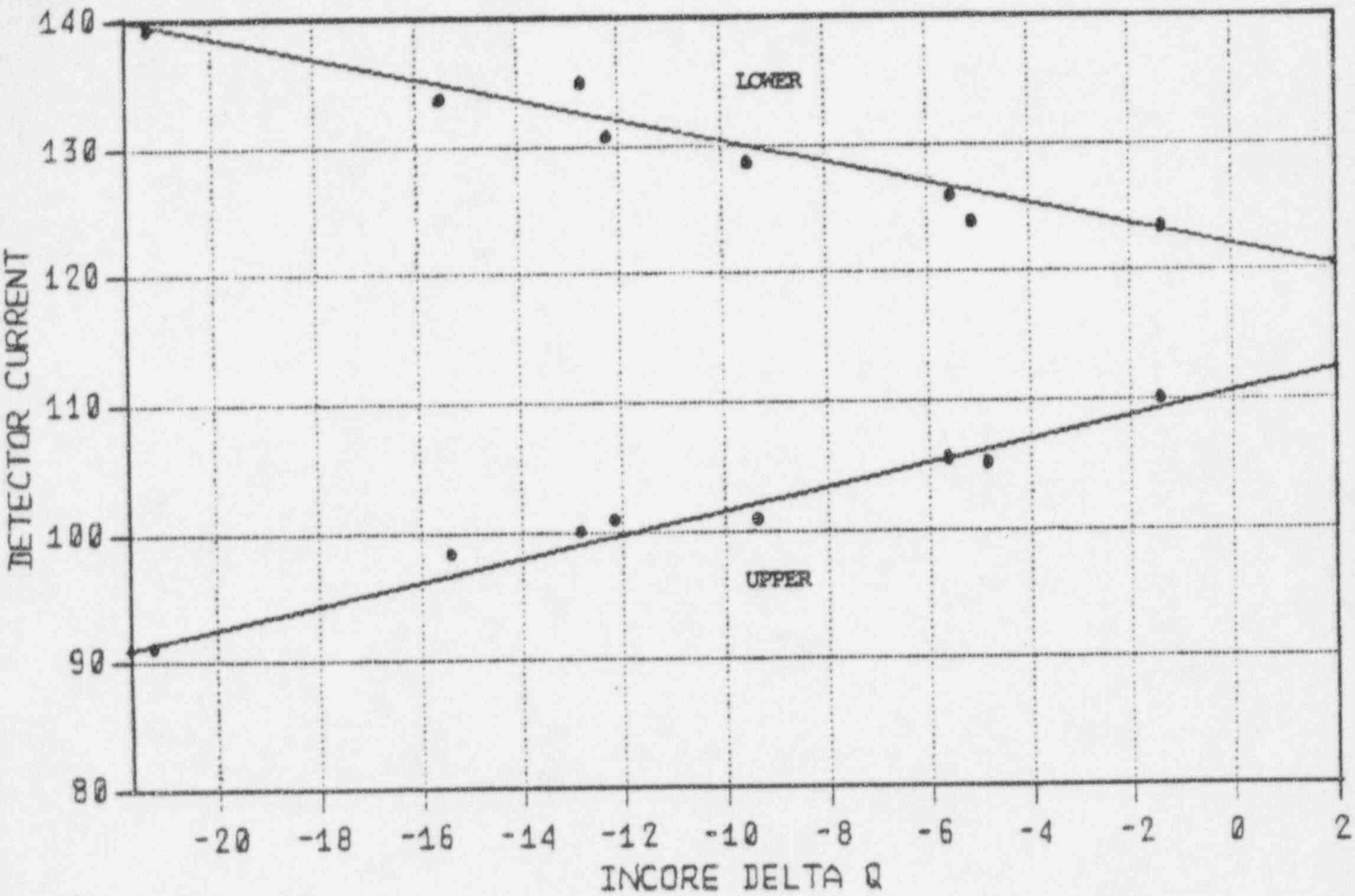
$$Upper_{CURR} = .902(\Delta q) + 112.94$$

$$Lower_{CURR} = -1.13(\Delta q) + 152.24$$

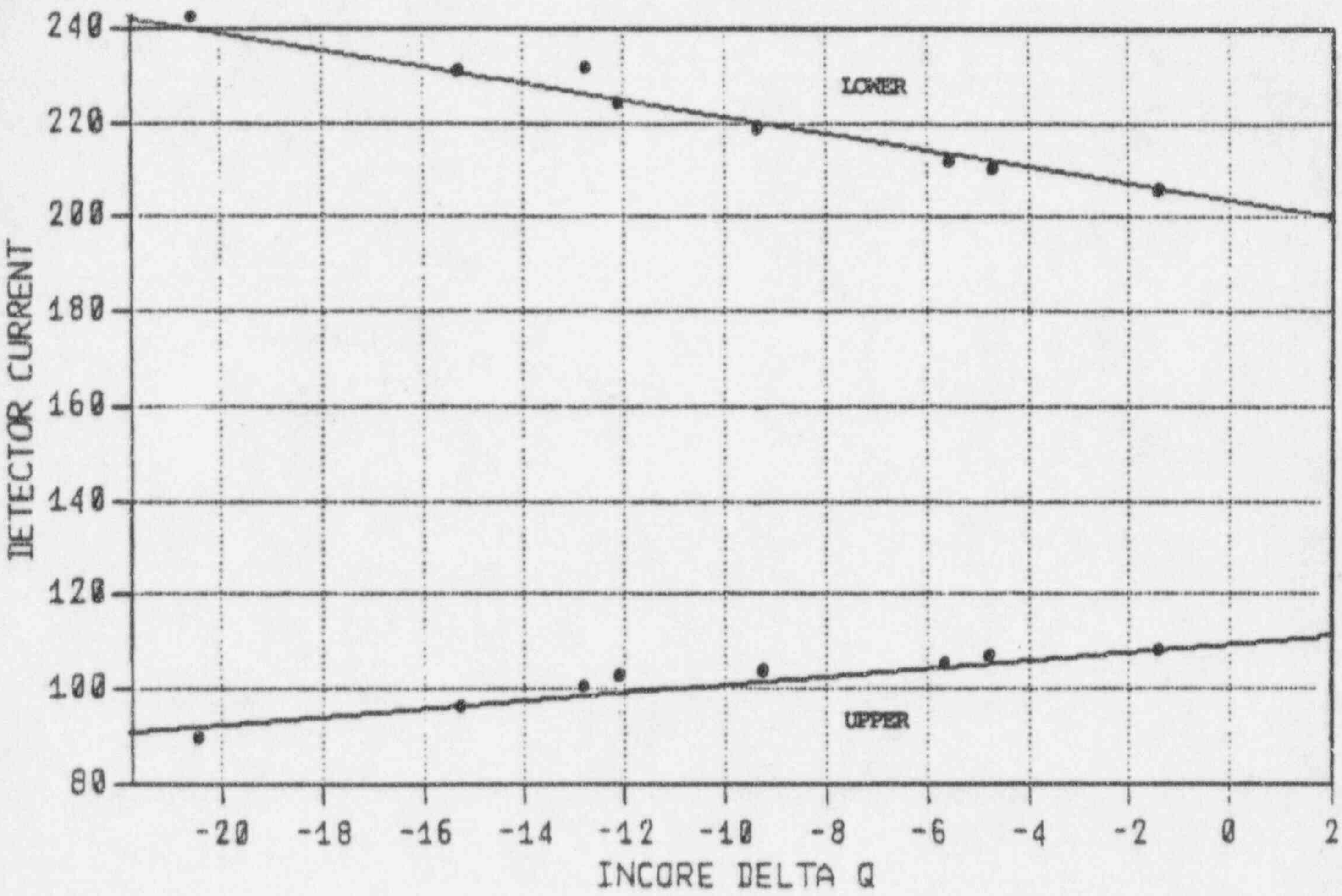
Notes: Number of data points	4
Axial Flux Difference swing	14.2%
Duration	7 hours



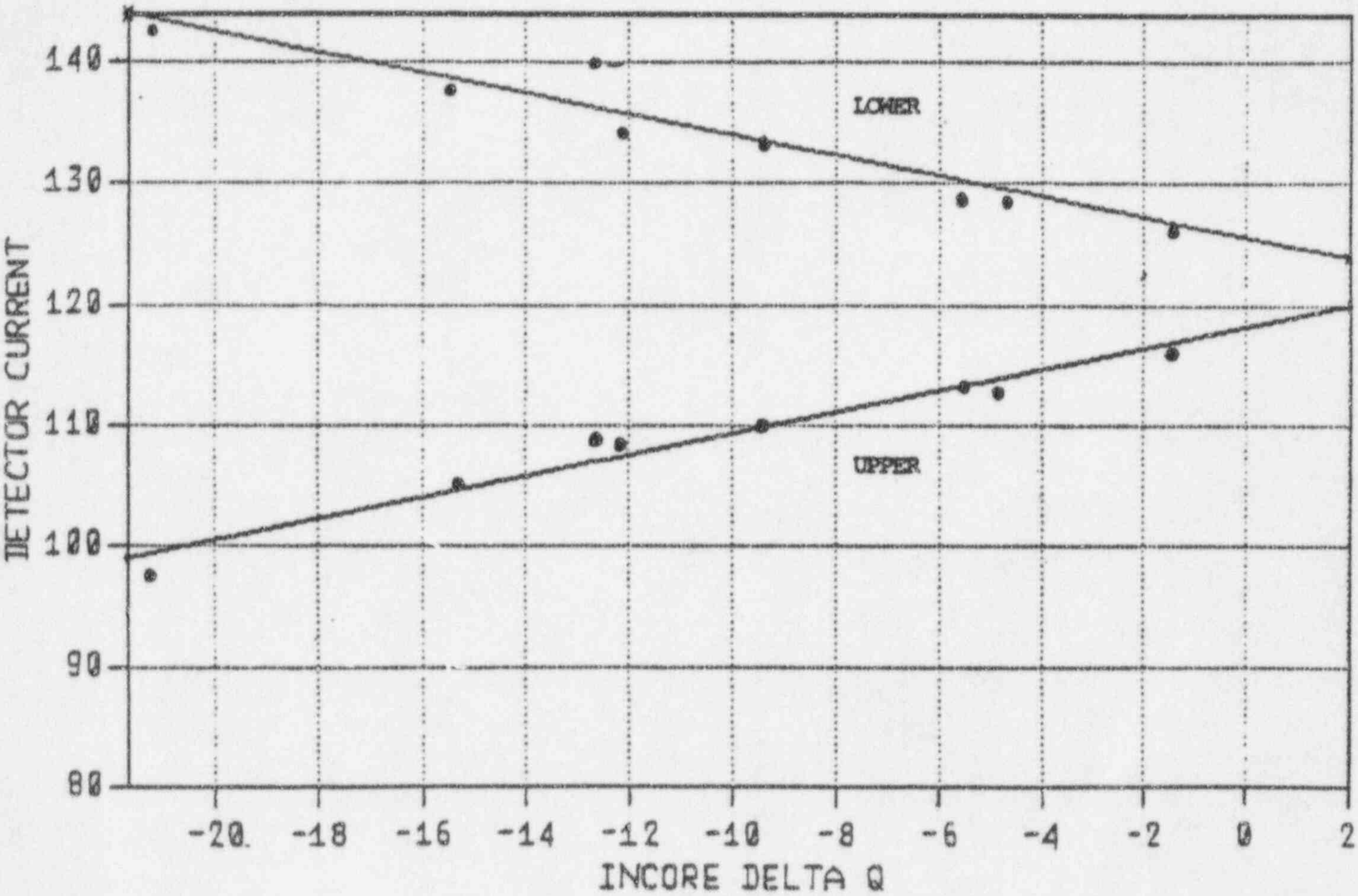
Millstone Nuclear Power Station Unit No. 3	AXIAL FLUX DIFFERENCE VERSUS TIME TEST 1 – 75 PERCENT POWER	Figure 8.2.4-1



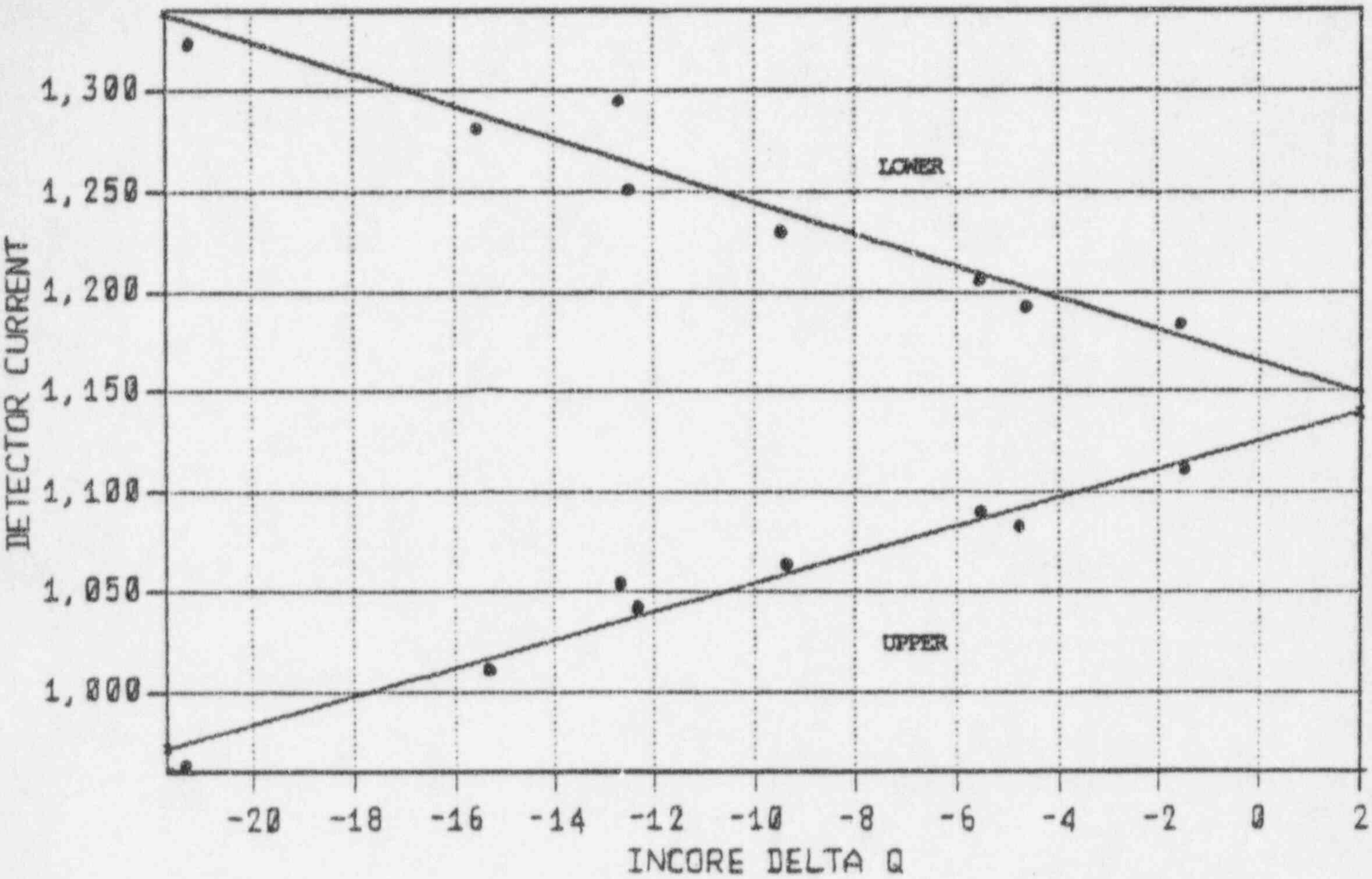
Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 41 CURRENT TEST 1 - 75 PERCENT POWER	Figure 8.2.4-2



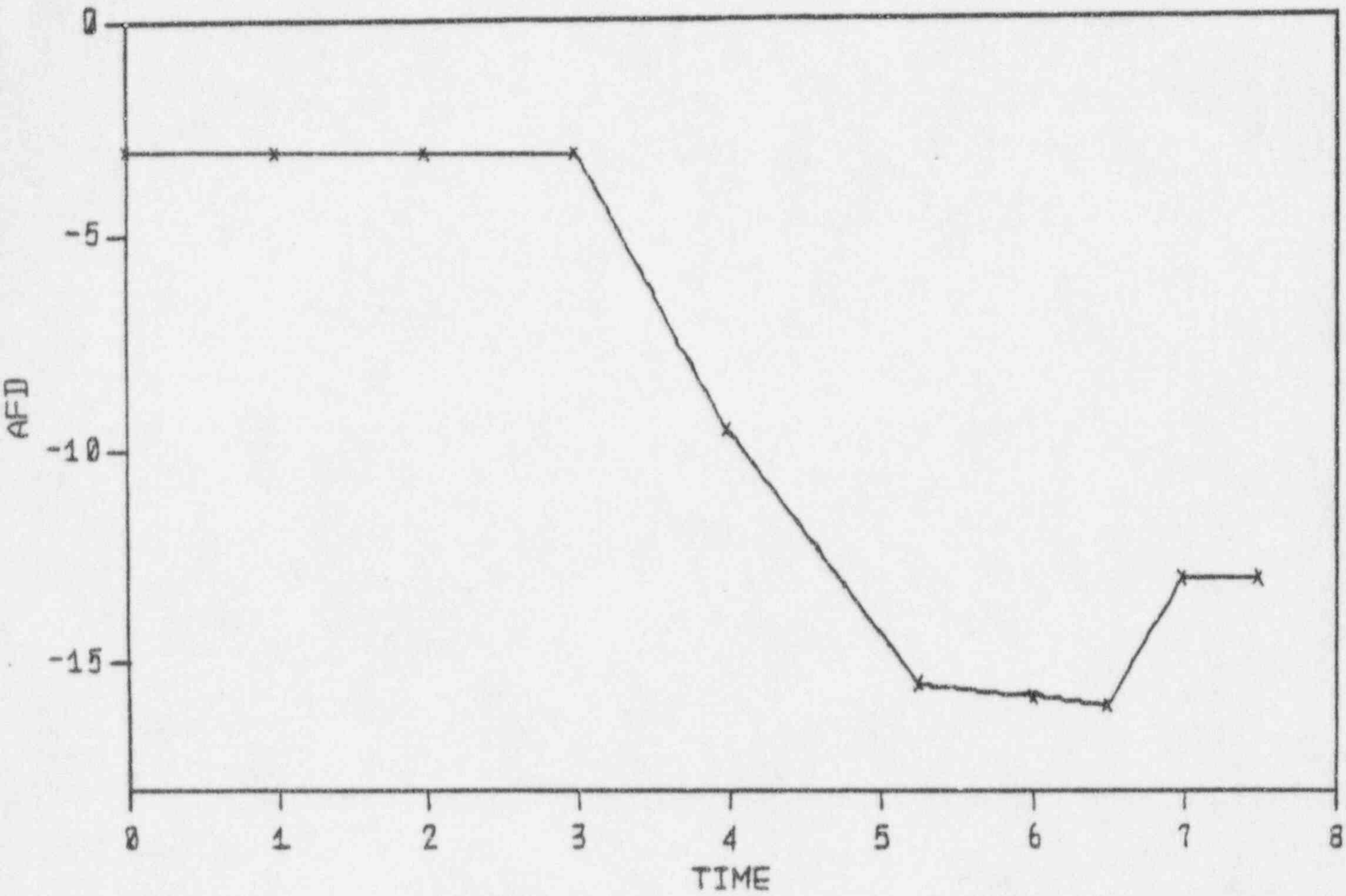
Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 42 CURRENT TEST 1 - 75 PERCENT POWER	Figure 8.2.4-3



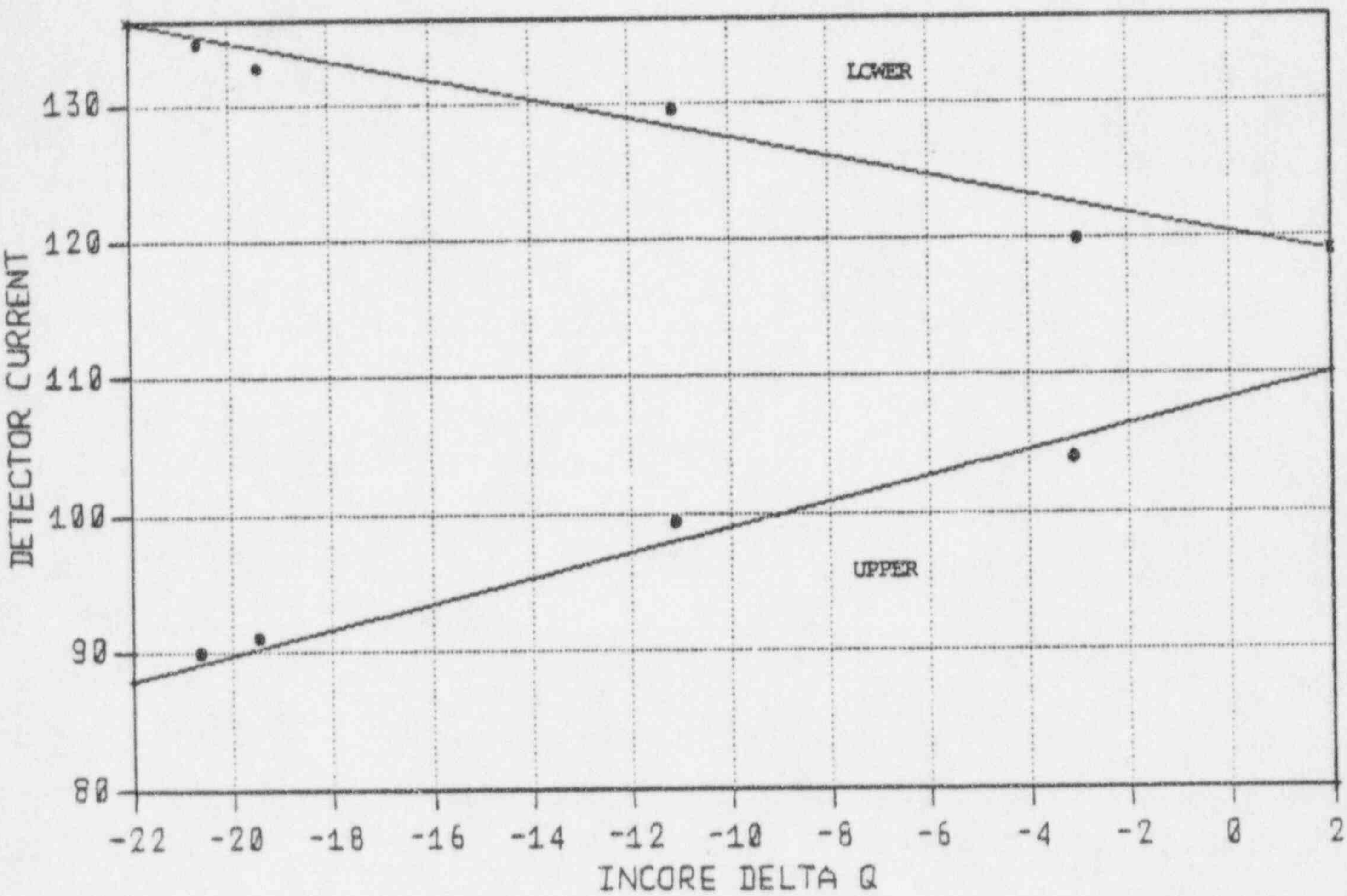
Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 43 CURRENT TEST 1 – 75 PERCENT POWER	Figure 8.2.4-4



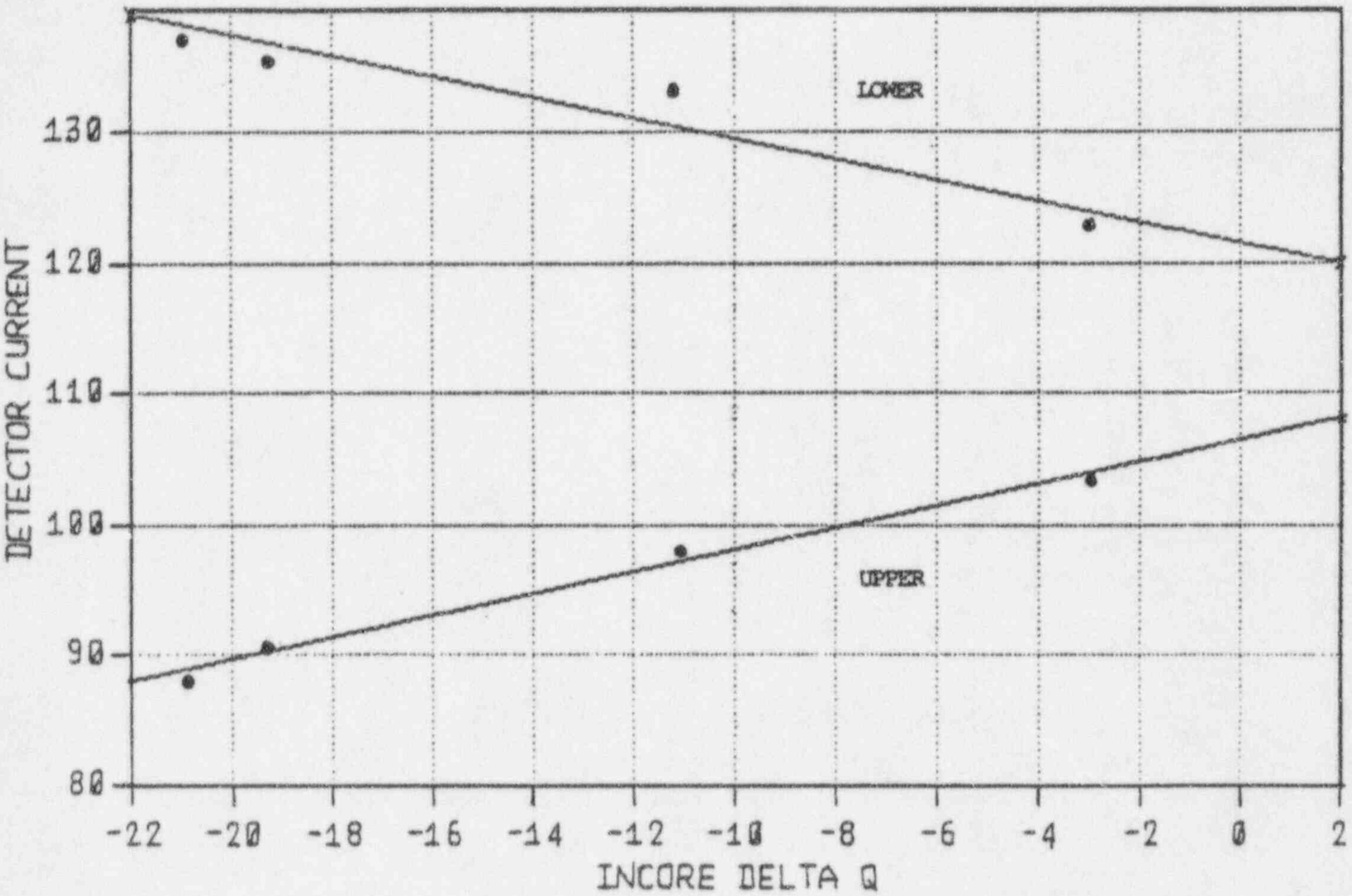
Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 44 CURRENT TEST 1 - 75 PERCENT POWER	Figure 8.2.4-5



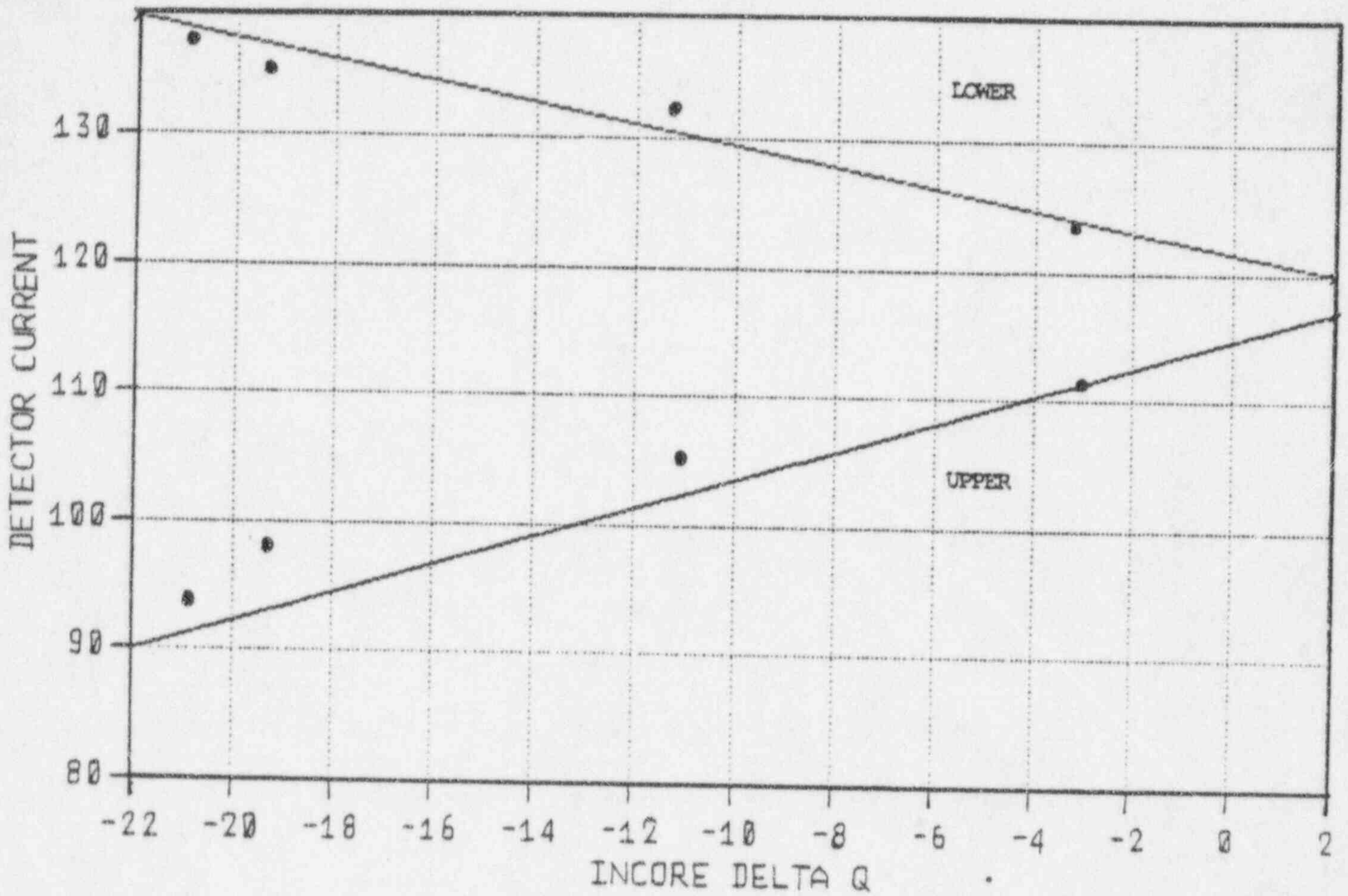
Millstone Nuclear Power Station Unit No. 3	AXIAL FLUX DIFFERENCE VERSUS TIME TEST 2 - 75 PERCENT POWER	Figure 8.2.4-6



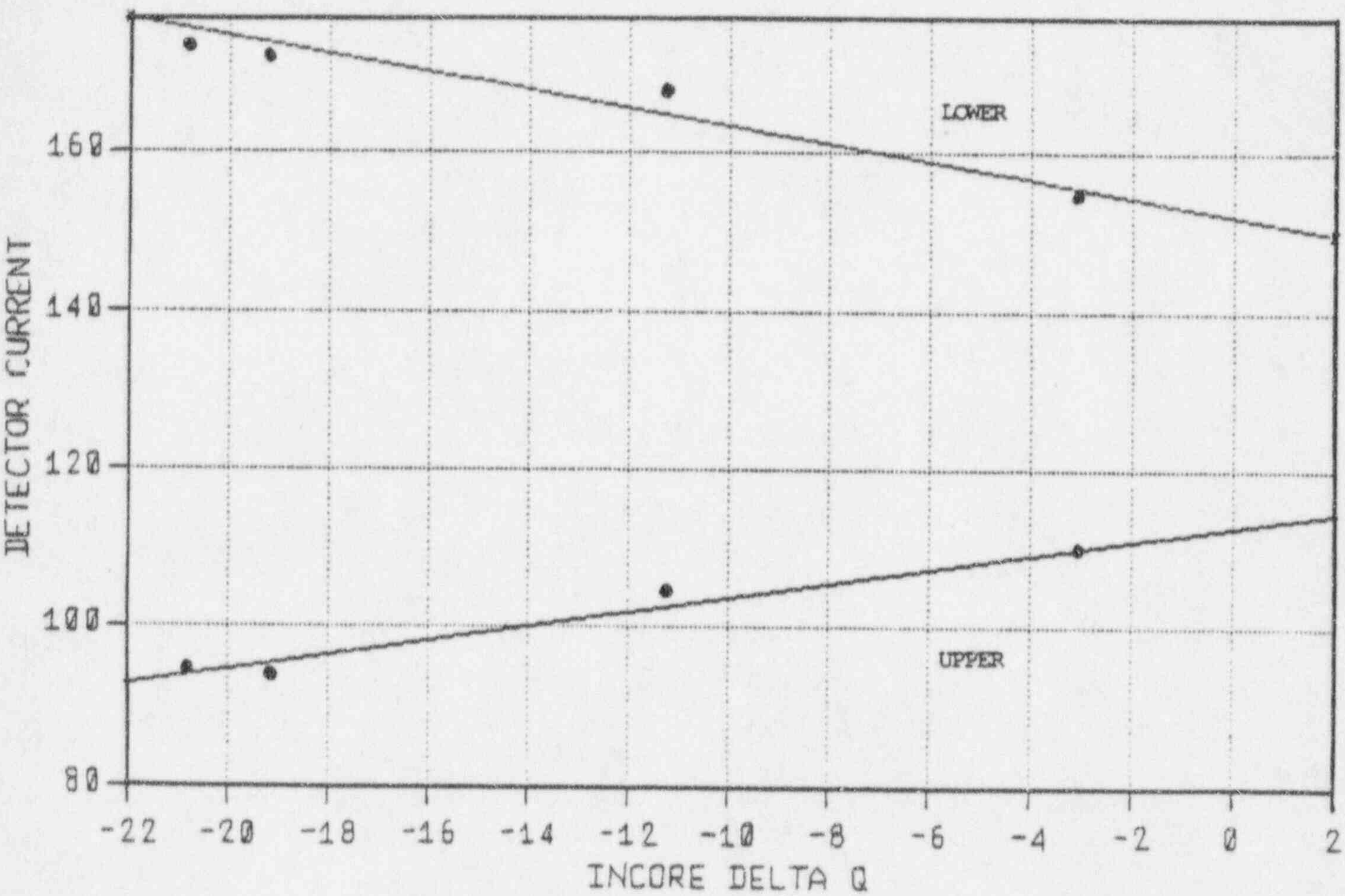
Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 41 CURRENT TEST 2 - 75 PERCENT POWER	Figure 6.2.4-7



Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 42 CURRENT TEST 2 - 75 PERCENT POWER	Figure 8.2.4-8



Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 43 CURRENT TEST 2 – 75 PERCENT POWER	Figure 8.2.4-9



Millstone Nuclear Power Station Unit No. 3	INCORE ΔQ VERSUS CHANNEL 44 CURRENT TEST 2 - 75 PERCENT POWER	Figure 8.2.4-10

8.3.1 REACTOR AND TURBINE CONTROL

3-INT-8000, Appendix 8005

OBJECTIVE

The objectives of this test were:

1. To determine the T_{avg} program resulting in the highest possible steam pressure and optimum plant efficiency without exceeding pressure limitations for the turbine, or the maximum allowable T_{avg}.
2. To obtain primary system temperatures, steam pressures and reactor thermal power data at steady-state conditions for zero, 30, 50, 75, 90 and 100 percent power levels.

DISCUSSION

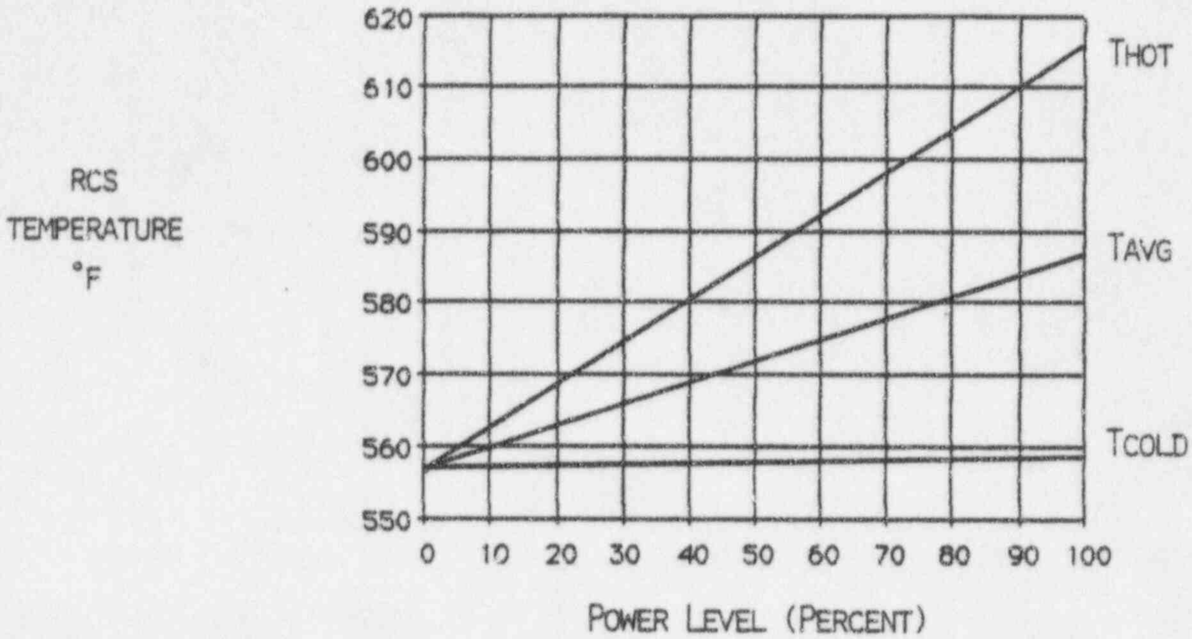
The test was performed on 02-01-86, 02-15-86, 03-17-86, 04-13-86, 04-15-86, and 04-18-86 with the plant at power levels of zero, 30, 50, 75, 90 and 100 percent, respectively. Plant performance data, including loop T_{hot}, T_{cold}, T_{avg}, feedwater flow, feedwater temperature, steam pressure, turbine inlet pressure, turbine impulse chamber pressure and plant gross electrical output, was collected at each power plateau. This data was then analyzed and compared to the design T_{avg} and steam pressure. Based on this comparison, adjustments to the T_{avg} control program were to be made to achieve the design steam pressure for each power level while still maintaining parameters within design limitations.

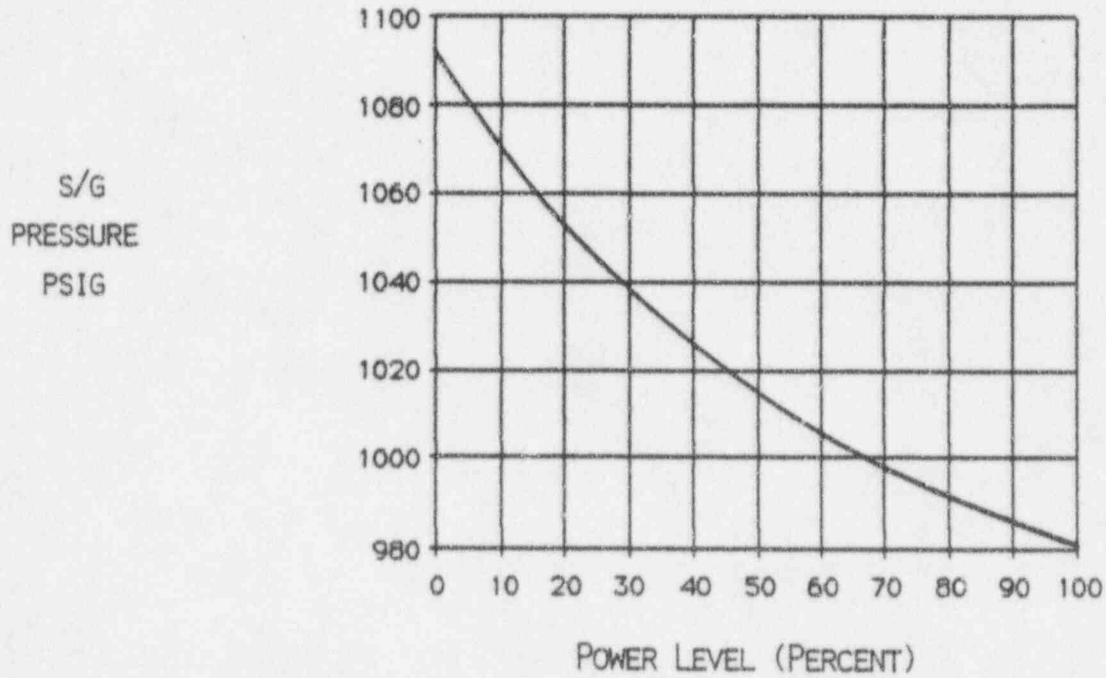
At the zero, 50, and 75 percent power level plateaus, data was taken twice - once with steam supplied to the moisture separator/reheaters (MSR) and once with steam isolated. The tests with steam supplied to the MSRs were intended to closely approximate actual plant performance conditions. Steam was continuously supplied to the MSRs during the 90 and 100 percent power level data collection periods.

During the 100 percent power testing, required plant conditions included full load steam generator pressures between 980 and 1000 psia, and T_{avg} less than the upper design limit of 587.1°F. This was to verify that the T_{avg} control program was properly adjusted.

RESULTS

The T_{avg} control program was verified to function properly in that T_{avg} and full load steam pressures were within design limits. No adjustments to the control program were required. Figure 8.3.1-1 provides the T_{avg} and average steam generator pressure as a function of power level, determined during the test.





Note: The above graphs are averaged representations of numerous data points taken during testing and should not be considered official test results.

Millstone Nuclear Power Station Unit No. 3	RCS TEMPERATURE AND STEAM GENERATOR PRESSURE AS A FUNCTION OF REACTOR POWER	Figure 8.3.1-1

8.3.2 DYNAMIC AUTOMATIC STEAM DUMP CONTROL TEST
3-INT-8000, Appendix 8013

OBJECTIVE

The objective of this test was to verify the proper closed loop response of the steam dump control system in the T_{avg} and steam pressure modes of operation. The T_{avg} mode was tested in both the plant trip and load reject submodes.

DISCUSSION

The test was performed on 02-11-86.

The plant trip submode was tested by increasing T_{avg} to 567°F with power maintained at 15 percent by manual rod control. A reactor trip was then simulated to the steam dump system so as to control T_{avg} on the plant trip controller. The steam dump was then placed in T_{avg} mode and data collected for 10 minutes to ensure the plant trip controller achieved and maintained a stable T_{avg}. The acceptance criteria was for T_{avg} to be maintained within 1°F of the program value of 562°F with no divergent oscillations in temperature.

The load reject submode was tested by maintaining power at 15 percent and T_{avg} at approximately program level (562°F) in manual rod control with a high rate of load rejection and zero impulse pressure simulated (load reject to 0 percent). The steam dump was placed in T_{avg} mode and data collected for 10 minutes to ensure that the load reject controller achieved and maintained a stable T_{avg}. The acceptance criteria was the T_{avg} to be maintained 1.5 to 4°F above the 557°F no load value with no divergent temperature oscillation.

The steam pressure mode was tested by setting the steam header pressure controller to 1078 psig at 15 percent power, placing the dump valve controller in automatic, and monitoring plant pressure response for 10 minutes following a slight increase in reactor power. The acceptance criteria was that the steam generator pressure controller response could maintain a stable 1078 psig pressure.

RESULTS

All test acceptance criteria were met. In the plant trip submode, T_{avg} was maintained at 561°F which was within the acceptance criteria of 562°F ±1°F. For the load reject submode, T_{avg} was maintained at 561°F which was within the acceptance criteria band of 558.5°F to 561°F. In the steam pressure mode, steam header pressure was maintained at 1078 psig which was as required by the acceptance criteria. No divergent oscillations were observed during any of the transient testing of the steam dump system.

8.3.3 AUTOMATIC REACTOR CONTROL

3-INT-8000, Appendix 8017

OBJECTIVE

The objective of this test was to verify the performance of the automatic reactor control system in maintaining reactor coolant average temperature, T_{avg}, within acceptable steady-state limits.

DISCUSSION

The test was performed on 02-18-86 with the reactor and turbine generator at a steady-state power level of 30 percent. The pressurizer level and pressure control, steam generator water level control, and turbine driven feed pump speed control systems were all in automatic. The steam dump system was in automatic in the T_{avg} mode. The rod control system was in manual. The following plant parameters were monitored: auctioneered nuclear flux, power mismatch, compensated power mismatch, auctioneered hi T_{avg}, compensated T_{avg}, T_{error}, compensated T_{ref}, rod speed demand, steam header pressure, turbine impulse pressure, and pressurizer pressure.

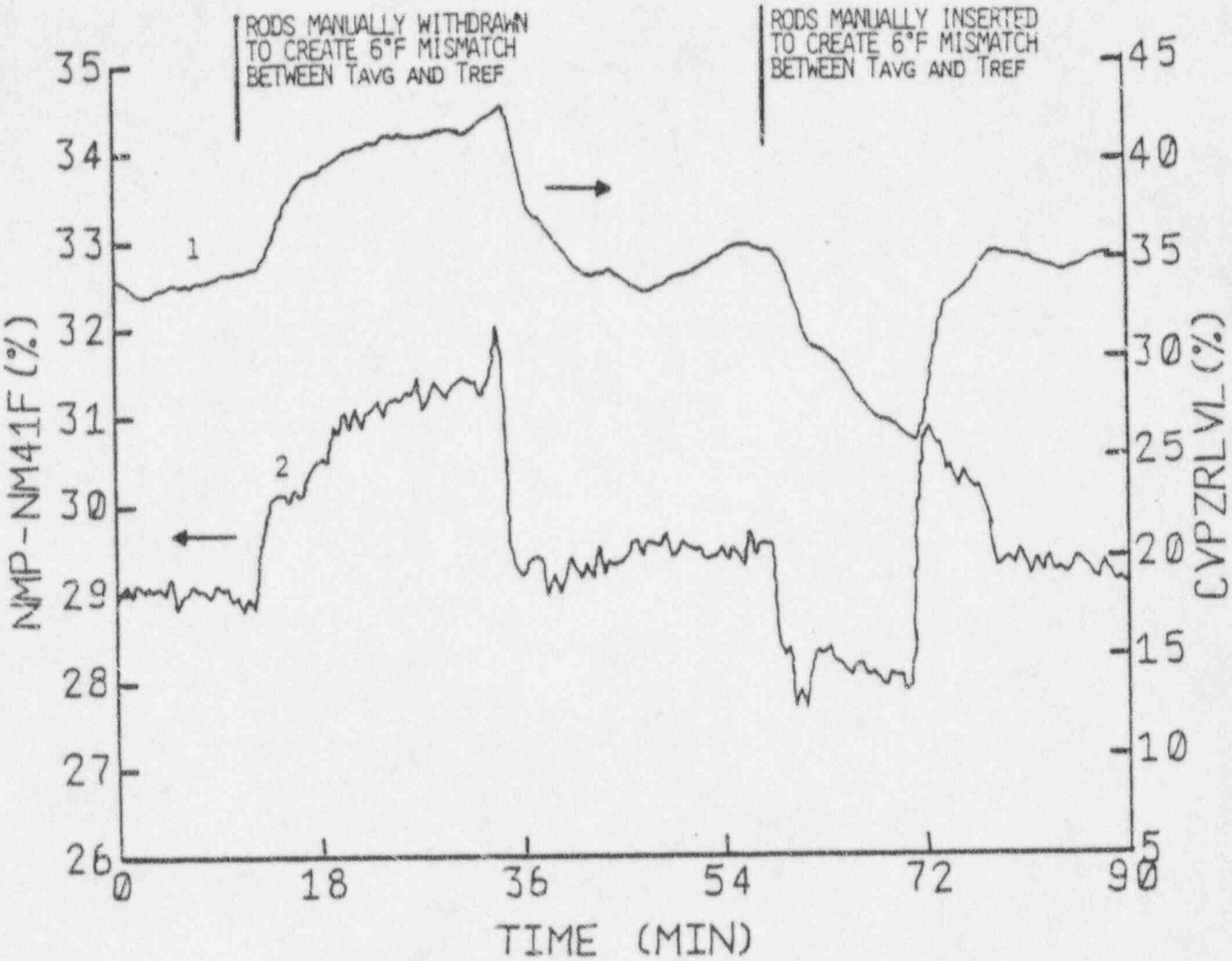
The test consisted of switching the rod control system to automatic and monitoring the plant response. Rods were then shifted to manual and withdrawn to create a 6°F mismatch between T_{avg} and T_{ref}. The rods were shifted to automatic to allow T_{avg} to return to T_{ref}. This step was then repeated with rods driven in to create the 6°F mismatch.

The acceptance criteria was that no manual intervention was required and that T_{avg} returned to within 1.5°F of T_{ref}.

RESULTS

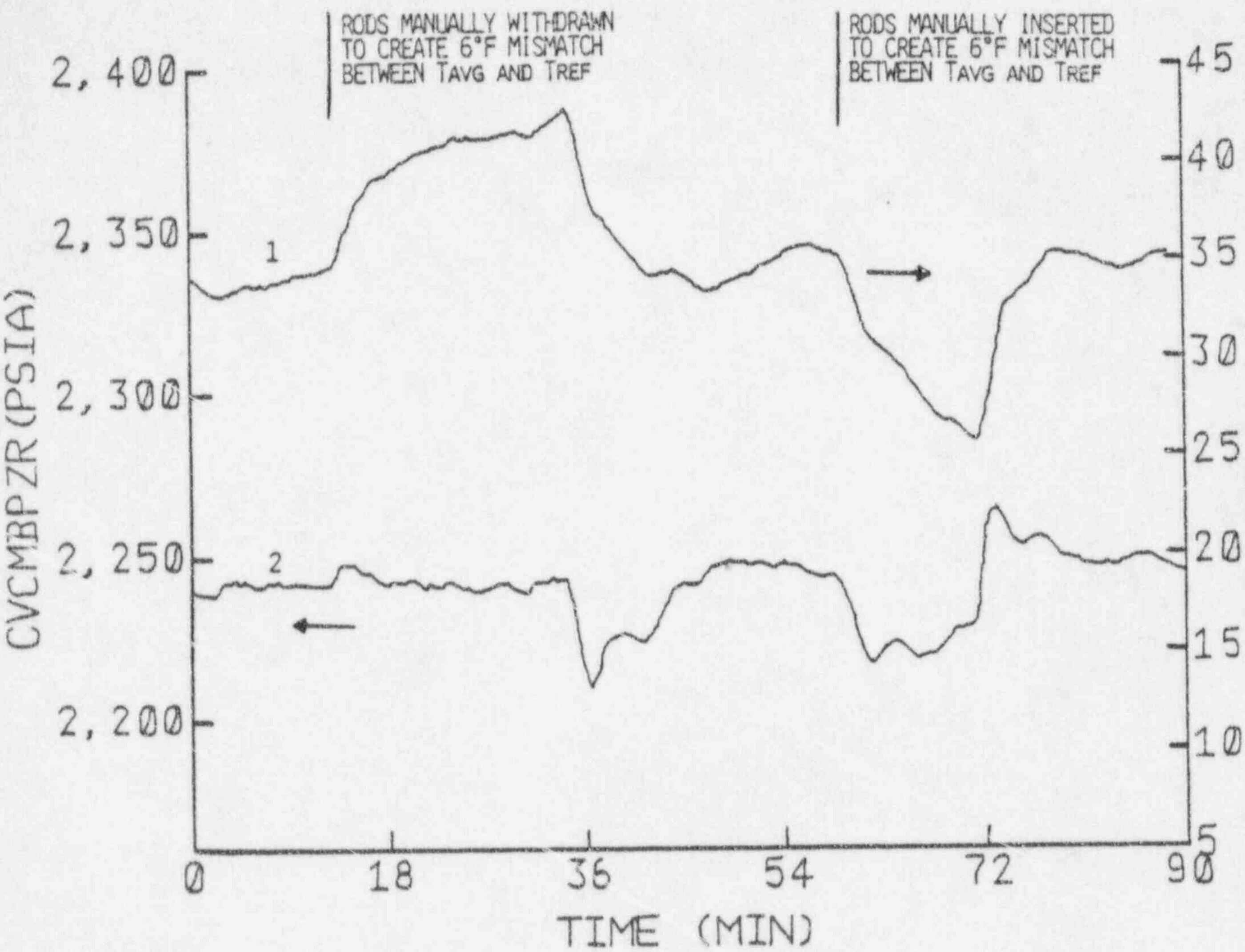
The plant responded as expected. The rod control system controlled T_{avg} in a stable manner. No adjustments were required to fine tune the instrumentation. Following rod withdrawal, T_{ref} was at 566°F and T_{avg} was at 572°F. Once automatic control was established, T_{avg} returned to 566°F within 398 seconds. Following rod insertion, T_{ref} was at 566°F and T_{avg} was at 560°F. Once automatic control was established, T_{avg} returned to 566°F within 259 seconds. At no time was manual intervention required.

The transient response of T_{hot}, T_{cold}, pressurizer level and pressure during this test is illustrated in Figure 8.3.3-1.



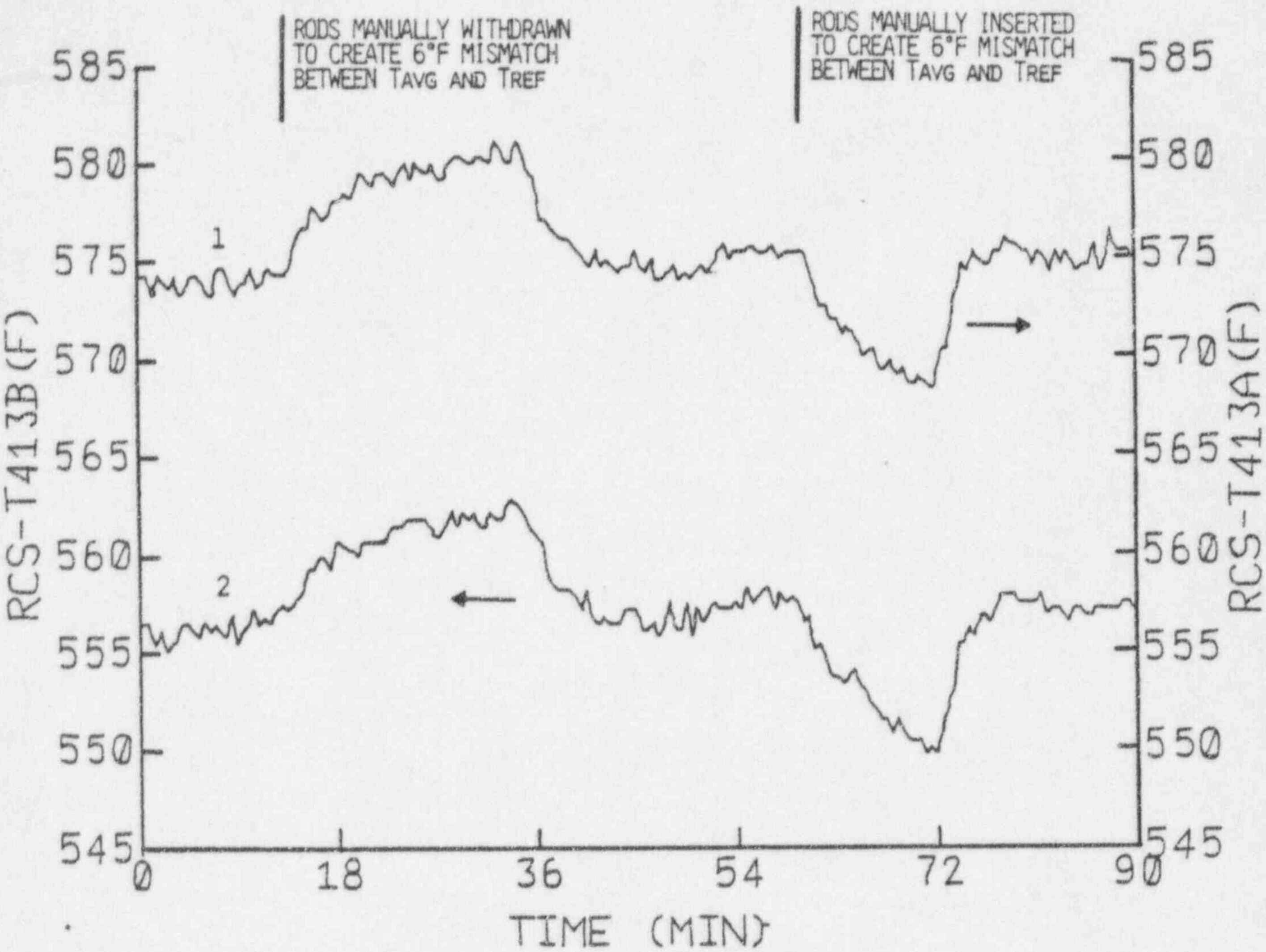
PLOT 1 - PRESSURIZER LEVEL
PLOT 2 - REACTOR POWER-POWER RANGE CHANNEL 41

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT AUTOMATIC REACTOR CONTROL TEST	Figure 8.3.3-1 Page 1



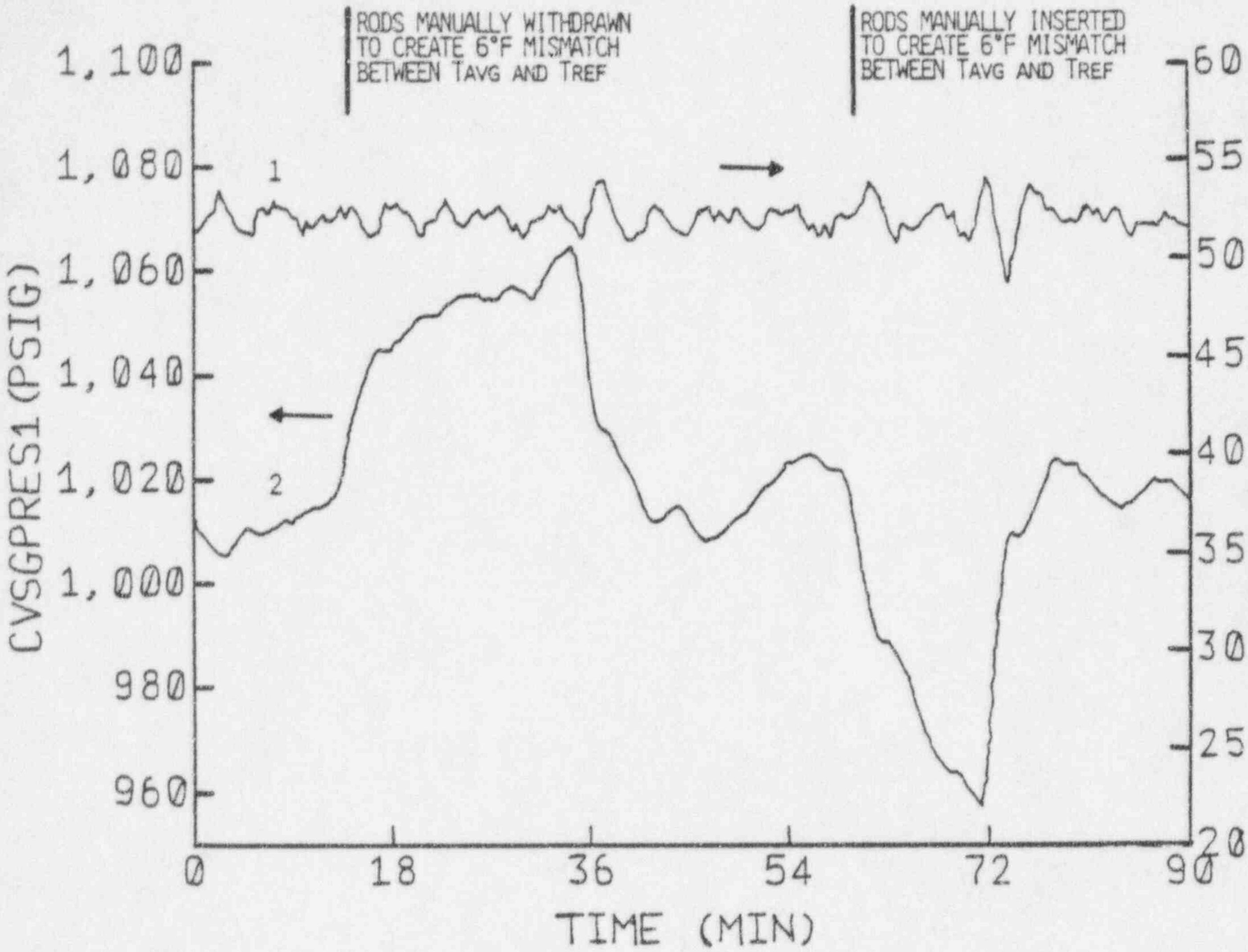
PLOT 1 - PRESSURIZER LEVEL
PLOT 2 - PRESSURIZER PRESSURE

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT AUTOMATIC REACTOR CONTROL TEST	Figure 8.3.3-1 Page 2



PLOT 1 - RCS LOOP 1 WIDE RANGE T_{HOT}
PLOT 2 - RCS LOOP 2 WIDE RANGE T_{COLD}

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT AUTOMATIC REACTOR CONTROL TEST	Figure 8.3.3-1 Page 3



PLOT 1 - STEAM GENERATOR 1 NARROW RANGE LEVEL
PLOT 2 - STEAM GENERATOR 1 PRESSURE

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT AUTOMATIC REACTOR CONTROL TEST	Figure 8.3.3-1 Page 4

8.3.4 AUTOMATIC STEAM GENERATOR WATER LEVEL CONTROL
3-INT-8000, Appendix 8018

OBJECTIVE

The objectives of this test were to:

1. Demonstrate the level control stability of the steam generator feedwater bypass valves in automatic control at low power.
2. Demonstrate the stability of the steam generator water level control system when transferring control from the feedwater bypass valves to the main feedwater valves.
3. Demonstrate proper response of the automatic steam generator level control system during plant transients at power levels of 50, 75, and 100 percent with adjustments being made as required to optimize system performance.
4. Demonstrate proper operation of the turbine driven feedwater pump speed control during power escalation.
5. Verify proper automatic programming of the steam generator level during power escalation.

DISCUSSION

The test was performed over the periods of 2-10-86 to 02-15-86, 02-16-86 to 03-23-86, 03-18-86 to 03-23-86, 03-28-86 to 03-30-86, and 04-20-86 to 04-21-86 at power levels of <5, 30, 50, 75 and 100 percent, respectively.

With the unit operating at less than 5 percent power and on the feedwater control bypass valves, a set of +5 percent and -5 percent narrow range steam generator level deviations were imposed on the plant. The system response was recorded as steam generator water level control was switched from manual to automatic. This verified the bypass valve control system before proceeding to higher power levels.

Testing the transfer of steam generator water level control from the feedwater control bypass valves to the main feedwater control valves was performed at 20 percent power. During this operation, the main feedwater control valves were slowly opened in manual while observing the feedwater control bypass valves closing in automatic.

At 30 percent power, the steam flow and feedwater flow instrument calibration was conducted in accordance with Appendix 8004. Level deviations of +5 percent and -5 percent were then used to observe the steam generator water level control system's transient response. At the 50, 75, and 100 percent power levels, tests consisted of repeating the steam flow and feedwater flow transmitter calibrations, followed by recording the system response to the 10 percent load swing test (Appendix 8022). The 75 percent power level test included system performance throughout a 50 percent load reduction (Appendix 8026). The plant parameters monitored during the tests included:

Steam Generator Programmed Level Setpoint
Narrow Range Steam Generator Water Level
Level Controller Output
Nuclear Instrumentation Power Level
Feedwater Flow
Steam Flow
Flow Error
Flow Valve Controller Output

Data was collected on strip chart recorders during the tests below 30 percent power. A computer was used as a data-logger for the 30, 50, 75, and 100 percent power tests.

During each test the process control loops for feedwater control valves and feed pump speed control were adjusted as required to achieve optimum performance. In addition, data on the control loop settings and the actual feedwater control valve differential pressure was recorded so that the scaling of the control valves could be adjusted to match plant performance. Though separate from this test, steam generator water level oscillations were observed at 58 percent power, and additional adjustment was performed to optimize system response before increasing power level. The feedwater control valve position was increased and feedwater pump speed decreased to stabilize the levels, and then further testing and control system adjustment resumed.

RESULTS

Automatic steam generator water level control demonstrated the ability to meet the established acceptance criteria:

1. Level overshoot/undershoot was less than ±4.0 percent following a level increase/decrease.
2. Level returned to within 2 percent of reference level, within 10 minutes following a transfer of level control, or within 20 minutes following a change in level or level setpoint.

Automatic feedwater pump speed control was demonstrated to meet the established acceptance criteria:

1. Feedwater pump discharge pressure oscillations were less than ±3 percent following a steam flow change.
2. Main feedwater control valve stem position was:

Steam Flow (%)	Valve Position (%)
30	10-30
50	20-40
75	40-60
100	60-85

8.3.5 MAIN STEAM ISOLATION VALVE CLOSURE TEST
3-INT-8000, Appendix 8037

OBJECTIVE

The objectives of this test were to:

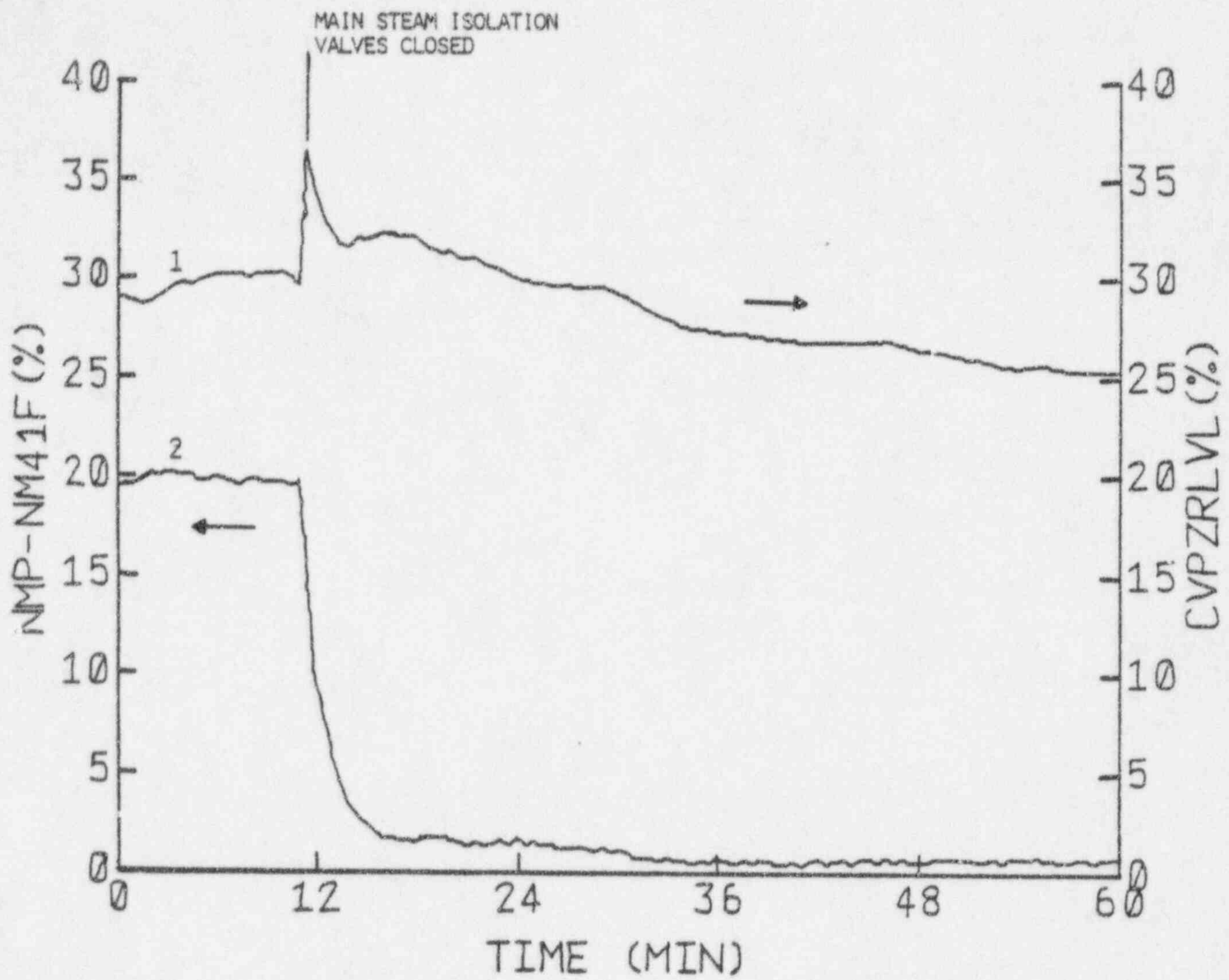
1) Verify, under dynamic steam flow conditions, the ability of the valves to close in less than 5 seconds.
2) Verify the ability of the primary plant, secondary plant, and plant automatic control systems to sustain the simultaneous closure of all MSIVs and bring the plant to stable hot standby conditions without initiating safety injection or lifting primary/secondary safety valves.

DISCUSSION

The test was performed on 03-31-86 with plant power being maintained at 20 percent. The test was initiated by the simultaneous manual closure of all four main steam isolation valves. The plant was brought to hot standby conditions by use of the atmospheric steam dumps. Final steam generator pressure was 1092 psig. Plant conditions were monitored using installed instrumentation, the plant computer, and a high speed data logger.

RESULTS

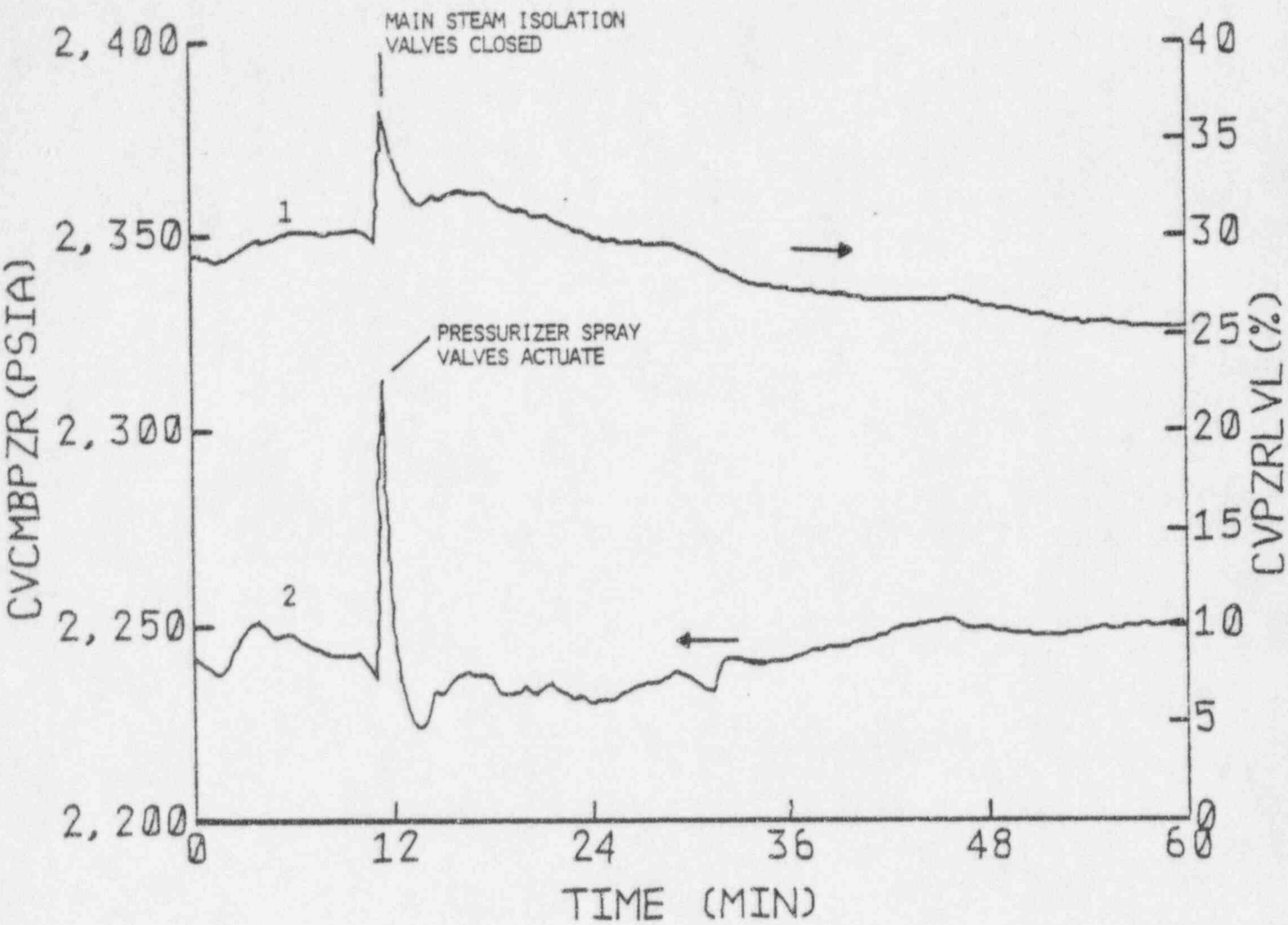
All MSIVs closed in less than 5 seconds with A, B, C, and D closing in 3.11, 2.76, 3.05, and 3.20 seconds respectively. During the test, neither the pressurizer safety valves nor main steam safety valves lifted, nor did safety injection initiate. All acceptance criteria were met. Plant performance following closure was as expected. The transient response of various plant parameters during this test is illustrated in Figure 8.3.5-1.



PLOT 1 - PRESSURIZER LEVEL

PLOT 2 - REACTOR POWER-POWER RANGE DETECTOR CHANNEL 41

Millstone Nuclear Power Station Unit No. 3	PLANT TRANSIENT RESPONSE PLOT MSIV CLOSURE TEST	Figure 8.3.5-1 Page 1



PLOT 1 - PRESSURIZER LEVEL
PLOT 2 - PRESSURIZER PRESSURIZER

Millstone Nuclear Power Station Unit No. 3	PLANT TRANSIENT RESPONSE PLOT MSIV CLOSURE TEST	Figure 8.3.5-1 Page 2

8.4.1 TURBINE OVERSPEED TEST
3-INT-8000, Appendix 8016

OBJECTIVE

The objective of this test was to demonstrate the capability of the turbine generator to consistently trip at acceptable speeds during an overspeed condition.

DISCUSSION

The test was performed on 02-15-86 with the plant, initially, at a 15 percent power level. Prior to performing the actual overspeed tests, the electrohydraulic control (EHC) system was put through a series of electrical and mechanical tests. After these were successfully performed, the unit's backup overspeed trip feature was tested by running the turbine generator up to 105 percent of rated speed and observing the trip. This was performed three times. The mechanical overspeed mechanism was then functionally checked at a reduced speed.

With the backup overspeed system and elements of the mechanical overspeed system tested, the turbine generator was then set to overspeed in order to perform a functional check of the mechanical overspeed trip and verify that the unit tripped at an acceptable level. This was also performed three times.

RESULTS

All checks and trips were successfully performed. During the 105 percent trip of the backup overspeed trip feature, the unit tripped consistently at 1894 RPM during each of the three runs. This was well within the acceptance criteria range of 1845 to 1935 RPM. During the mechanical overspeed trip portion, the unit tripped at 1962, 1963, and 1963 RPM. This compared well to the acceptance criteria of $\leq$ 1998 RPM.

8.4.2 10 PERCENT LOAD SWING TESTS

3-INT-8000, Appendix 8022

OBJECTIVE

The objective of this test was to verify proper plant transient response, including automatic control system performance, when 120 MWE step load changes were introduced at the turbine generator.

DISCUSSION

The test was performed on 2-18-86, 3-23-86, 3-29-86 and 4-21-86 at reactor power levels of 30, 50, 75 and 100 percent, respectively. The test consisted of rapidly lowering the generator load by approximately 120 MWE by adjusting the EHC load limiter to a predetermined target value. When the plant had stabilized at the new power level, the generator load was rapidly increased to its original level using the EHC standby load set potentiometer.

During and after each transient, the following plant parameters were monitored:

- Auctioneered nuclear flux
- Loop 1 T_{hot} narrow range
- Loop 1 T_{cold} narrow range
- Loop 1 T_{avg}
- Loop 1 ΔT
- T_{ref}
- SG 1 feed flow
- Steam flows
- Steam generator levels
- Steam header pressure
- Feed pump discharge pressure
- Pressurizer pressure
- Pressurizer level

Auctioneered T_{avg}

Loop 1 overpower ΔT trip setpoint

Loop 1 overtemperature ΔT trip setpoint

Generator output (MWE)

Feedwater temperature

Acceptance criteria for this test were:

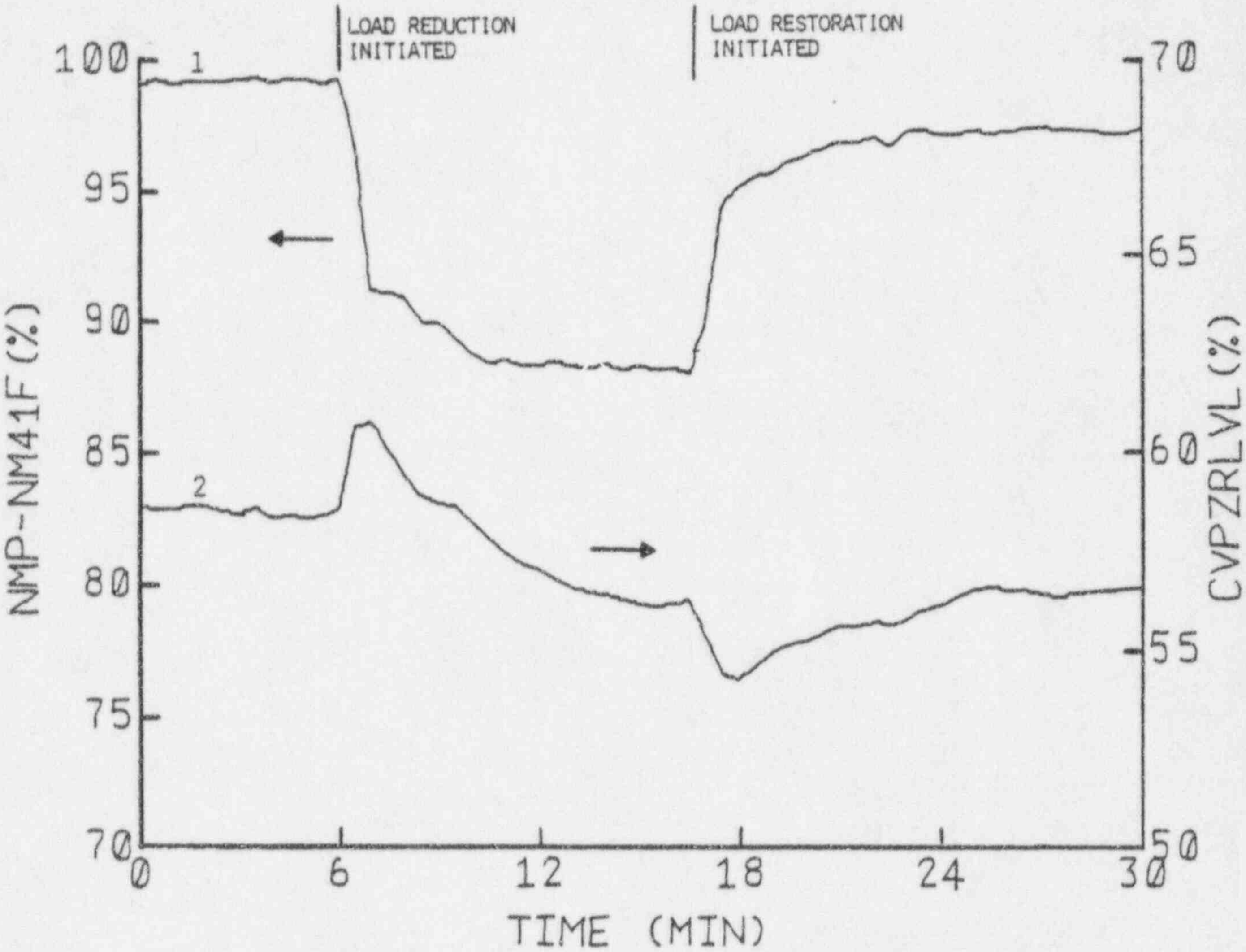
1. Reactor trip does not occur
2. Turbine trip does not occur
3. Steam generator atmospheric dump valves do not lift
4. Steam generator code safety valves do not lift
5. Pressurizer power operated relief valves do not lift
6. Pressurizer code safety valves do not lift
7. Unexpected manual operator intervention is not required
8. Plant parameters do not incur sustained or divergent oscillations
9. Nuclear power overshoot or undershoot is < 3 percent

RESULTS

The test was successfully performed with the following exceptions:

1) On the 10 percent decrease from 75 percent, the atmospheric dump valve for steam generator A lifted. The setpoint selected on the main board hand-indicating controller for that valve was set too low. The setpoint was readjusted by Operations personnel.
2) On the 10 percent decrease from 100 percent power, feedwater flow started oscillating. Manual intervention was required to stop the oscillation. I&C personnel investigated and determined that the steam generator water level controller characteristics had been changed by a recent repacking of feedwater regulating valves. The valves had been made less responsive due to tighter packing. The steam generator level control system was adjusted to compensate for the tighter packing.

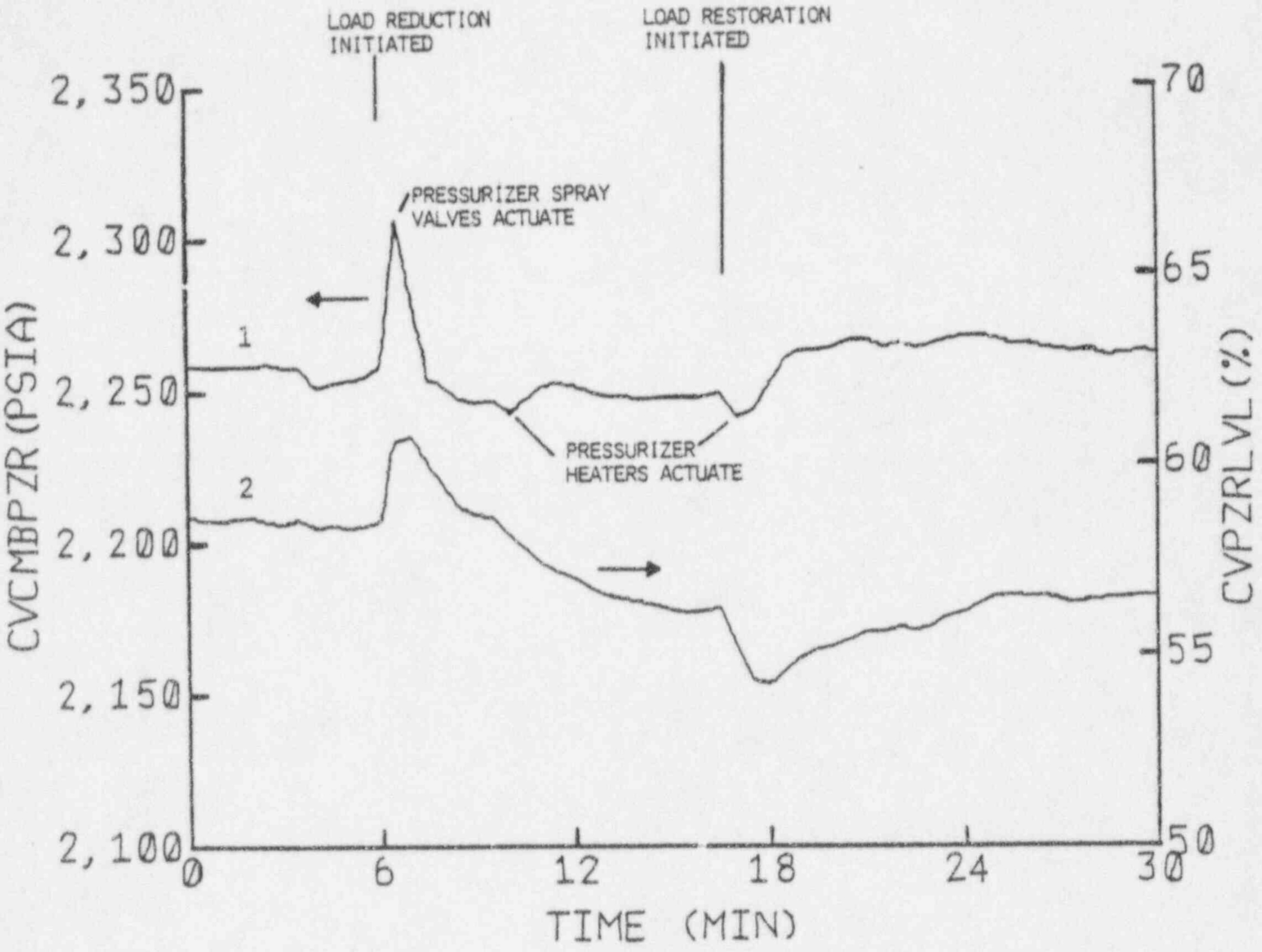
The above discrepancies were corrected as noted or evaluated to be acceptable. During each induced transient, undershoot/overshoot was within the 3°F acceptace criteria. The maximum value observed was approximately 2°F undershot during the increase to 100 percent power. Figure 8.4.2-1 provides a representation of typical plant response to a load change. The information was taken during the testing at 100 percent power.



PLOT 1 - REACTOR POWER-POWER RANGE DETECTOR CHANNEL 41

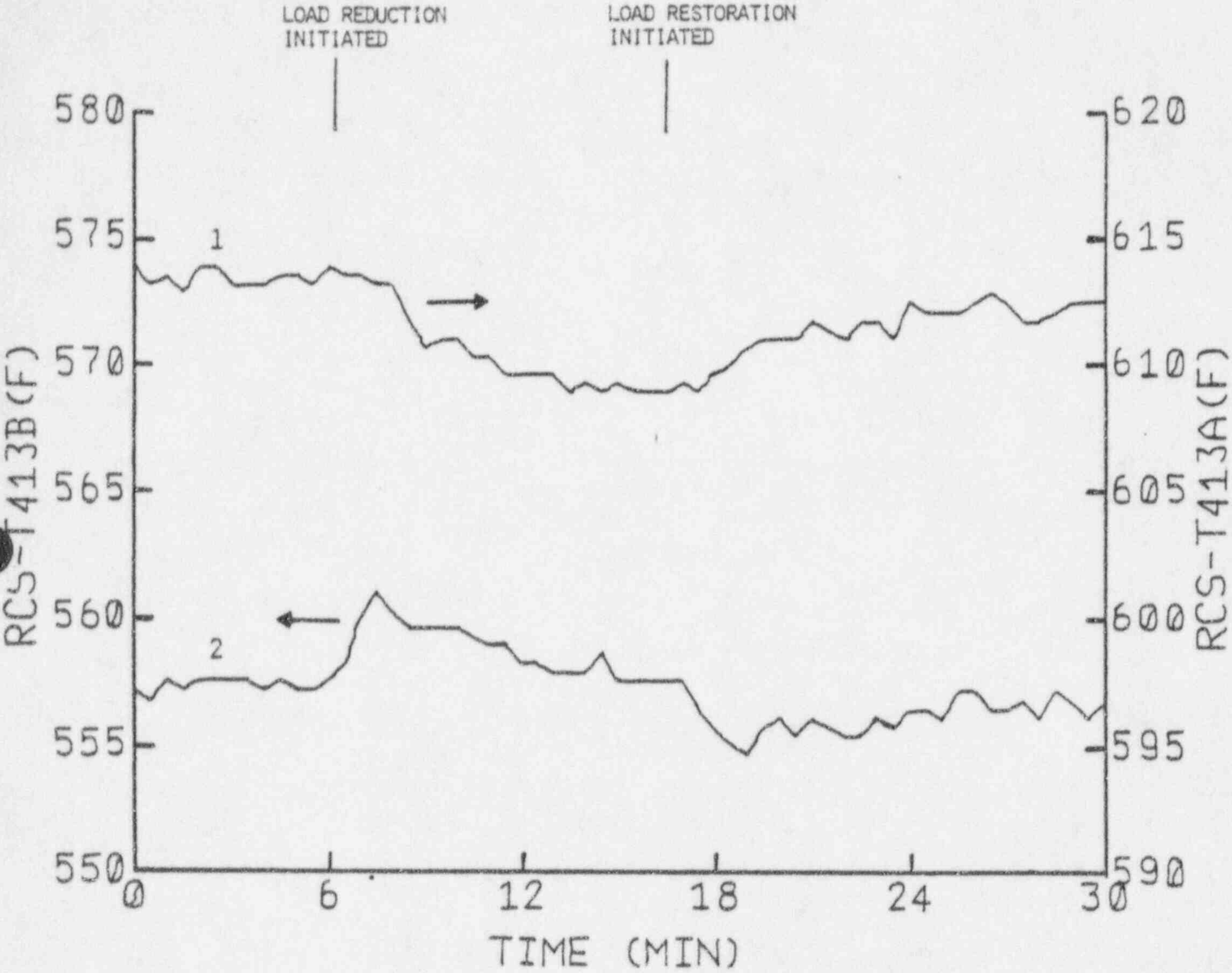
PLOT 2 - PRESSURIZER LEVEL

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT 10 PERCENT LOAD SWING	Figure 8.4.2-1 Page 1



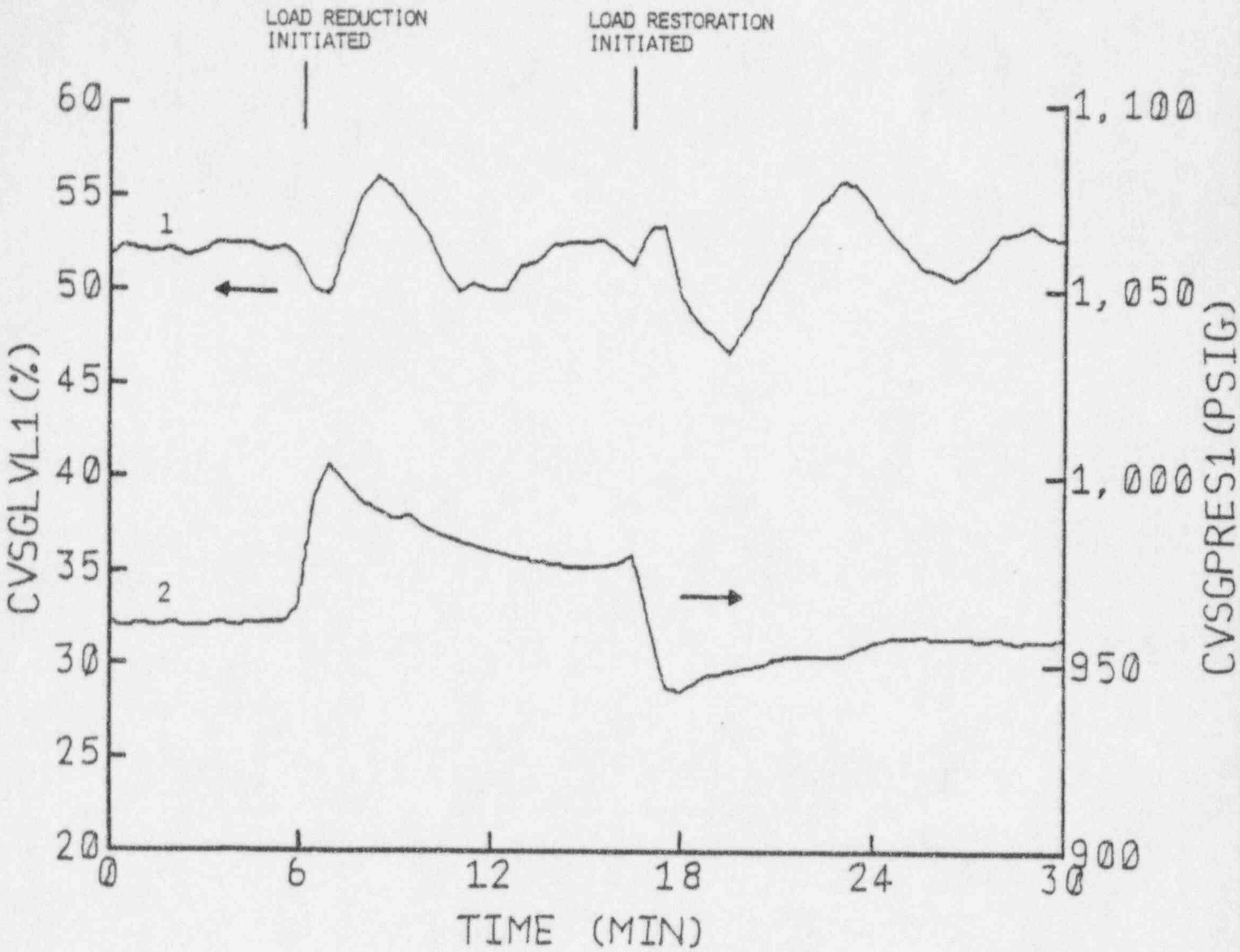
PLOT 1 - PRESSURIZER PRESSURE
PLOT 2 - PRESSURIZER LEVEL

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT 10 PERCENT LOAD SWING	Figure 8.4.2-1 Page 2



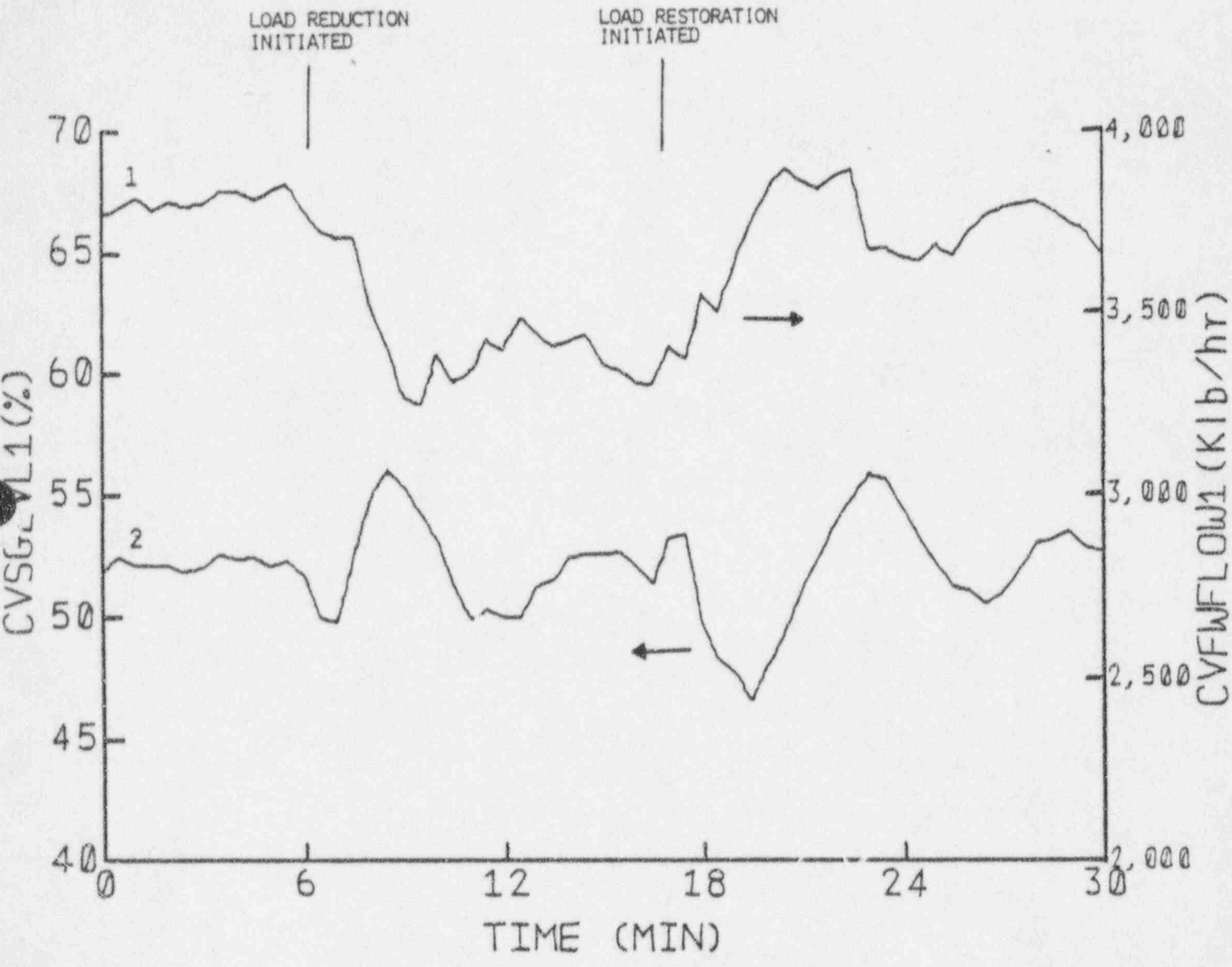
PLOT 1 - RCS LOOP 1 WIDE RANGE T_{HOT}
PLOT 2 - RCS LOOP 1 WIDE RANGE T_{COLD}

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT 10 PERCENT LOAD SWING	Figure 8.4.2-1 Page 3



PLOT 1 - STEAM GENERATOR 1 NARROW RANGE LEVEL
PLOT 2 - STEAM GENERATOR 1 PRESSURE

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT 10 PERCENT LOAD SWING	Figure 8.4.2-1 Page 4



PLOT 1 - STEAM GENERATOR 1 PRESSURE
PLOT 2 - STEAM GENERATOR 1 NARROW RANGE LEVEL

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT 10 PERCENT LOAD SWING	Figure 8.4.2-1 Page 5

8.4.3 REACTOR TRIP AND SHUTDOWN OUTSIDE CONTROL ROOM
3-INT-8000, Appendix 8023

OBJECTIVES

The objectives of this test were:

1. To demonstrate plant trip and shutdown capability from outside the control room, resulting in hot standby condition, utilizing the Technical Specification minimum shift crew.
2. To demonstrate that the plant can be maintained in hot standby condition from outside the control room.
3. To demonstrate that plant control can be transferred back to the control room from the remote control location.

As an initial condition of the test, reactor power level was required to be greater than 10 percent.

DISCUSSION

With the reactor operating at a power level of approximately 15 percent, the test commenced at 1630 on 02-18-86 by initiating a remote reactor trip from the reactor trip breakers located on the 43'6" level of the Auxiliary Building. Turbine trip occurred automatically following the reactor trip. Plant control was then transferred to the Auxiliary Shutdown Panel located on the 4'6" level of the Control Building. A Hot Standby condition (Mode 3) was achieved at 1635. After Mode 3 had been maintained for more than thirty minutes, control was transferred back to the Control Room. Reactor startup commenced at 1730 hours.

No abnormal conditions occurred during the test. System, equipment, and instrument response was as expected for a normal plant trip.

RESULTS

The acceptance criteria for the reactor trip and shutdown outside the control room test were:

1. The plant can be remotely tripped with transfer to the Auxiliary Shutdown Panel. Hot standby condition (Mode 3) can be achieved from outside the control room per plant Emergency Operating Procedures.
2. Plant Hot standby condition (Mode 3) can be maintained for at least 30 minutes from outside the control room.
3. With stable plant conditions, control can be transferred back to the control room from the remote control location.

All acceptance criteria for the test were demonstrated satisfactorily.

In addition to the above test, the ability to take the plant to Hot Shutdown (Mode 4) from outside the Control Room was successfully demonstrated during the precore hot functional test.

8.4.4 LARGE LOAD REDUCTION

3-INT-8000, Appendix 8026

OBJECTIVE

The objectives of the test were to:

1) Verify the ability of the primary plant, secondary plant and the automatic reactor control system to sustain a 50 percent step load reduction from a 75 percent power level.
2) To obtain transient response data for the evaluation of the interaction of plant systems.
3) To obtain transient response data for determination if control system setpoint changes were required to improve transient response based on actual plant operation.

DISCUSSION

The test was performed on 03-30-86. Prior to the start of the test, the plant was operating in steady-state conditions at 75 percent power. Additionally, the reactor rod control system, the turbine bypass system, steam generator water level control system, pressurizer pressure and level control systems and the feedwater pump speed control system were in automatic control.

The reduction in power was accomplished by a rapid lowering of the setpoint of the turbine control load limiter to a previously determined target value.

Acceptance criteria for the test was that the plant could sustain the transient without a reactor or turbine trip, safety injection, lifting of steam generator or pressurizer safety valves or unexpected manual intervention. In addition to these acceptance criteria, there also were predicted values for the extreme values of several plant parameters during the transient. These included T_{avg}, steam generator and pressurizer levels, pressurizer pressure and time duration of maximum rod speed and steam dump actuation.

RESULTS

The plant responded as expected. The transient was successfully performed and all acceptance criteria were met. The plant electrical load was reduced from 861 MWE to 214 MWE, a drop of 56.3 percent. Of this reduction, 550 MWE were shed in the first 25 seconds of the transient. Figure 8.4.4-1 indicates the reduction in generator output during the performance of the test.

The only operator involvement in the establishment of stable conditions after the transient was to place the feed pump speed controller to manual. This was to minimize the interaction between the two pumps at low power levels.

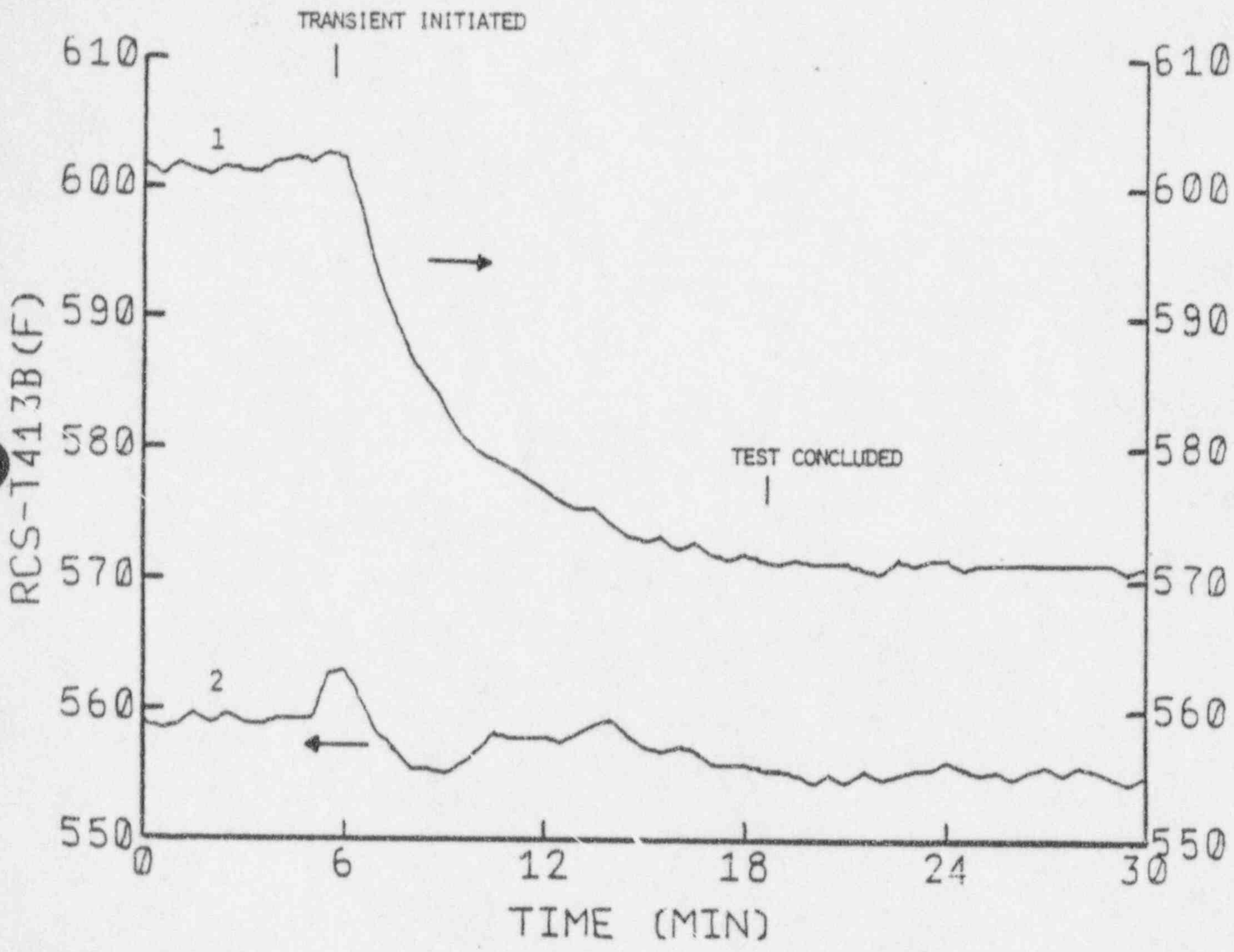
During the duration of the test, the predicted transient extremes of several parameters were exceeded. This was due to the load reduction being larger than 50 percent and were not deviations from the acceptance criteria. The predicted extreme and actual extreme values are shown in Table 8.4.4-1. The primary system pressure transient was controlled by pressurizer spray and a 4.5 second opening of the PORV.

After completing the test, the plant was returned to a 75 percent power level to permit the continuation of the testing program.

The transient response of various plant parameters during this test is illustrated in Figure 8.4.4-2.

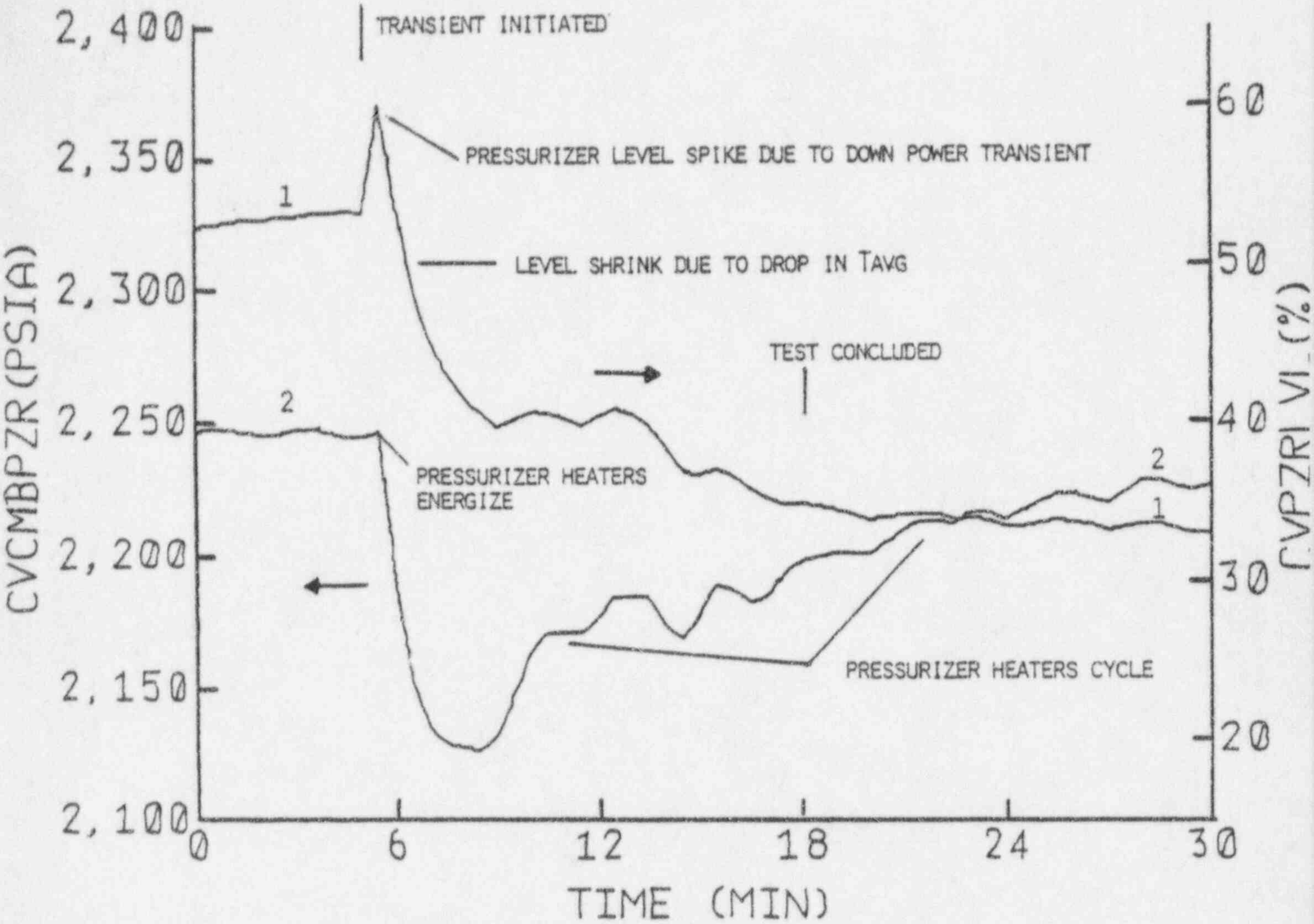
PARAMETER	EXPECTED EXTREME	ACTUAL EXTREME
Tave peak	<5° F above initial steady state value	7°F
Tave undershoot	<5° F below final steady state value	2°F
Tave oscillation	<5° F during steam dump	0°F
Primary pressure	≤ +80 psi -100 psi	+75 psi -125 psi
Steam Generator Level	≤+15%	S/G A -22.5% + 19% S/G B -14% + 13% S/G C -25% + 22% S/G D -25% + 18%
Maximum Rod Speed	≤ 30 seconds	1 minute 16 seconds
Steam Dump Actuation	≤ 8 minutes	8 minutes 30 seconds

Note: The above values are expected results and do not represent acceptance criteria.



PLOT 1 - RCS LOOP 1 WIDE RANGE THOT
PLOT 2 - RCS LOOP 1 WIDE RANGE TCOLD

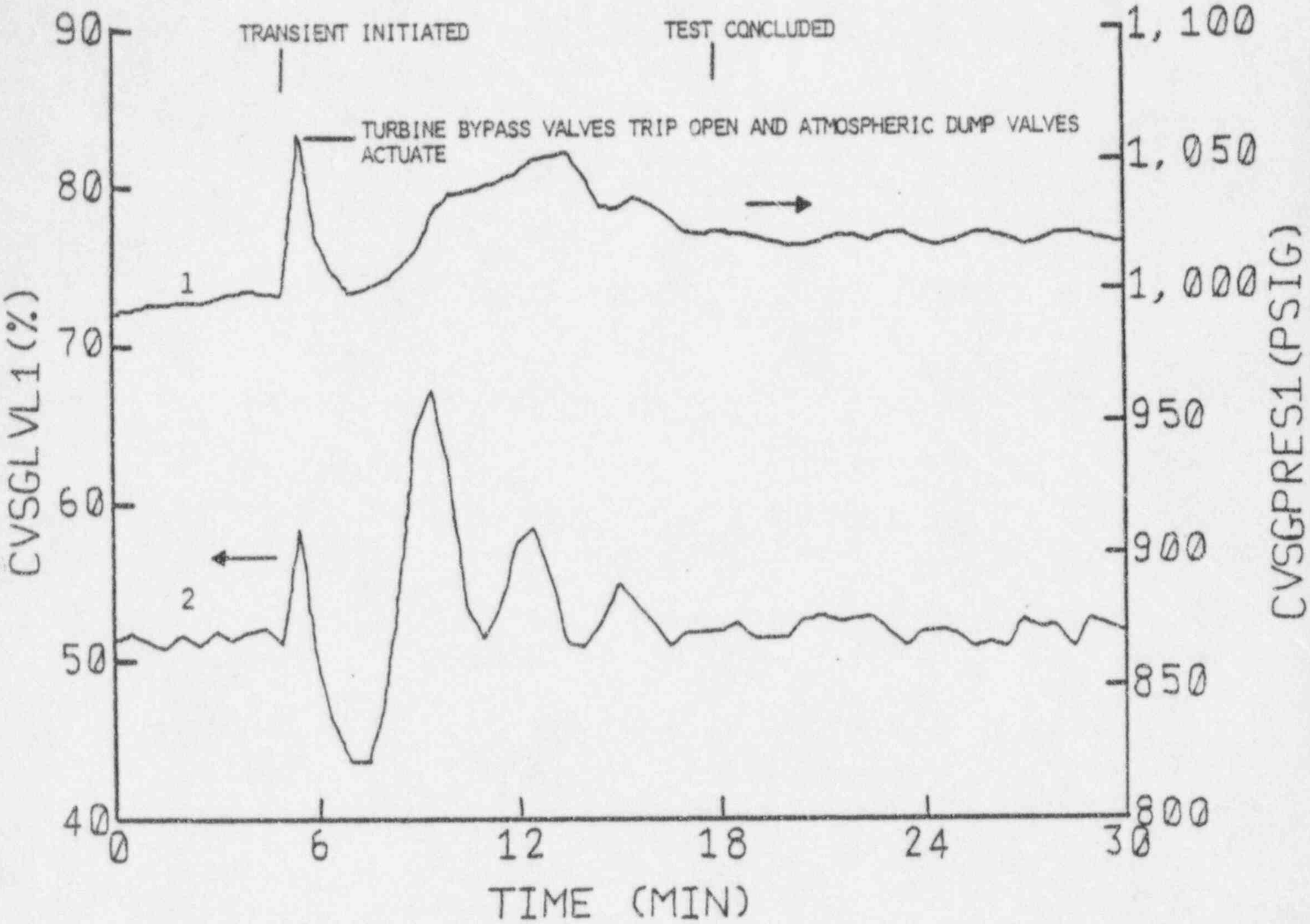
Millstone Nuclear Power Station Unit No. 3	PLANT TRANSIENT RESPONSE PLOT LARGE LOAD REDUCTION	Figure 8.4.4-1 Page 1



PLOT 1 - PRESSURIZER LEVEL

PLOT 2 - PRESSURIZER PRESSURE

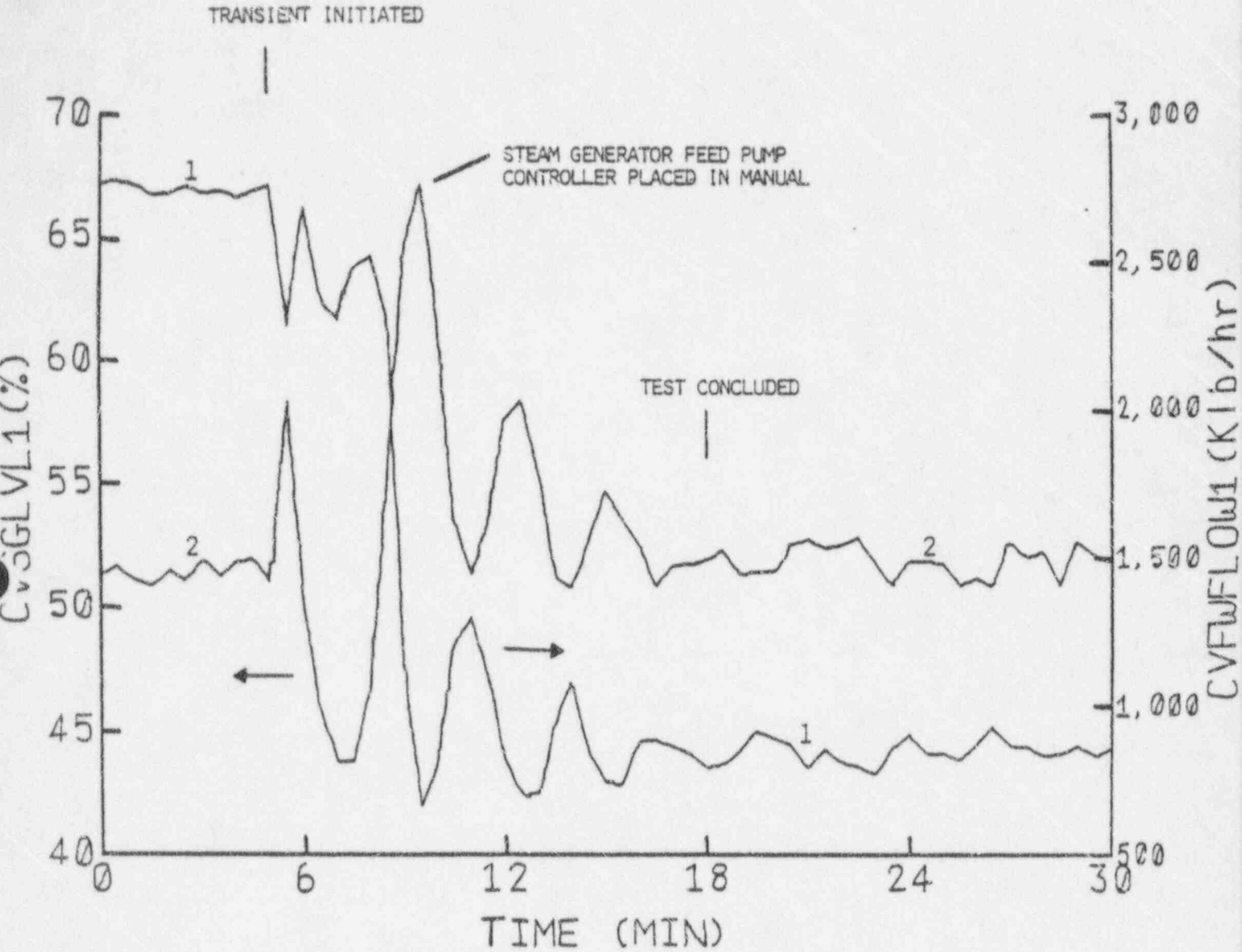
Millstone Nuclear Power Station Unit No. 3	PLANT TRANSIENT RESPONSE PLOT LARGE LOAD REDUCTION	Figure 8.4.4-1 Page 2



PLOT 1 - STEAM GENERATOR 1 PRESSURE

PLOT 2 - STEAM GENERATOR 1 NARROW RANGE LEVEL

Millstone Nuclear Power Station Unit No. 3	PLANT TRANSIENT RESPONSE PLOT LARGE LOAD REDUCTION	Figure 8.4.4-1 Page 3



PLOT 1 - STEAM GENERATOR 1 FEED FLOW
PLOT 2 - STEAM GENERATOR 1 NARROW RANGE LEVEL

Millstone Nuclear Power Station Unit No. 3	**PLANT TRANSIENT RESPONSE PLOT LARGE LOAD REDUCTION**	Figure 8.4.4-1 Page 4

8.4.5 LOSS OF POWER TEST (20 PERCENT POWER)
3-INT-8000, Appendix 8030

OBJECTIVE

The objectives of this test were to:

1. Demonstrate that the plant responds as designed following a plant trip with no offsite power.
2. Demonstrate that the turbine driven auxiliary feedwater pump (TDAFP) will maintain adequate steam generator levels for a minimum of two hours with the motor driven auxiliary feedwater pumps (MDAFP) and the auxiliary feedwater pump cubicle ventilation system out-of-service.
3. Demonstrate the capability of the batteries to provide vital power without any AC support (battery chargers and AC power to the inverters out-of-service) for a minimum of two hours.

DISCUSSION

The test was performed on 03-31-86. Just prior to initiating a loss-of-power, the MDAFP and the auxiliary feedwater pumps' cubicle ventilation system were removed from service by placing applicable switch controls in pull-to-lock. This ensured that only the TDAFP would be available to provide feedwater to the steam generators and that it would run without any ventilation. Also, AC power breakers to the battery chargers and inverters were opened.

The test was initiated with the plant at 16 percent power level. The turbine was off-line and steam was being dumped to the condenser through the condenser dump/turbine bypass valves. The transient was begun by manually tripping the reactor and then opening all off-site feeder breakers for the 4.16KV and 6.9KV buses. The emergency diesel generators started and sequenced on vital loads. Plant response was monitored with the computer and control board indications. Natural circulation was established in the primary system. The TDAFP and atmospheric dump valves were used to remove heat for a period of two hours.

Following the test, a plant startup was performed to support further testing.

RESULTS

All acceptance criteria for this test were met, with exceptions noted, as follows:

1. The diesel generators started and sequenced on loads as required except that that the auxiliary building filter fan 3HVR*FN6B and cold shutdown air compressor 3IAS-C2B failed to start. In addition, control building chiller 3HVK*CHL1B started as designed, but tripped shortly thereafter. See Appendix D for a discussion of problems encountered during LOP and their resolution.
2. The TDAFP operated well within established design limits throughout the two-hour run as indicated below.

ITEM	MAXIMUM READING	LIMIT
Bearing Temperature	134°F	≤200°F
Bearing Supply Oil Temperature	94°F	≤150°F
Bearing Return Oil Temperature	106°F	≤180°F
Turbine Rotor Vibration (peak-to-peak)	.6 mils	≤1.5 mils

ITEM	MAXIMUM READING	LIMIT
Pump Shaft Vibration 4400 RPM (peak-to-peak)	.90 mils	≤1.2 mils
Pump Shaft Vibration 3400 RPM (peak-to-peak)	.95 mils	≤1.5 mils

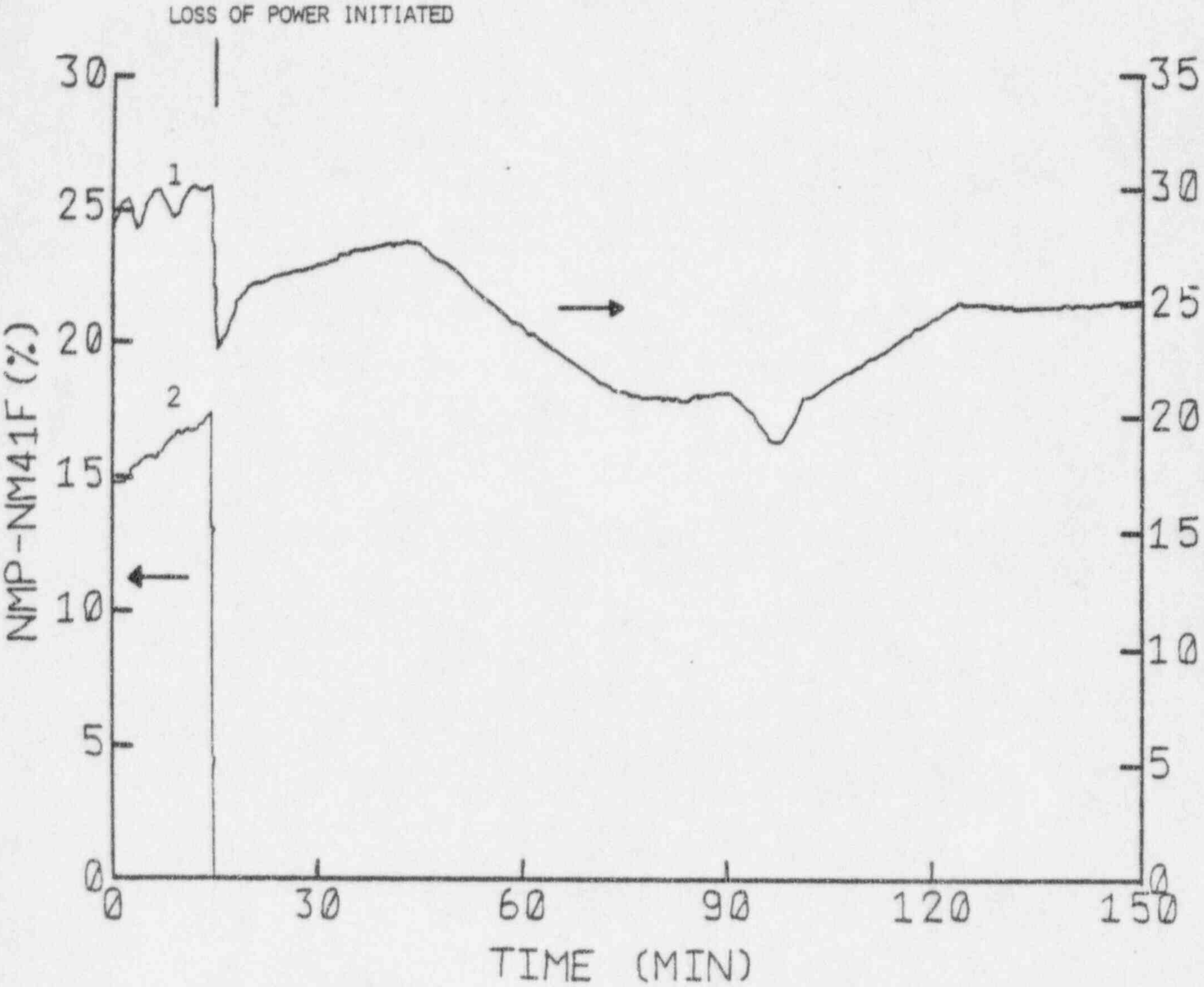
The maximum readings for Bearing Temperature (134°F) and Turbine Rotor Vibration (.6 mils) were recorded immediately after startup. Within 15 minutes, both readings were down to 120°F and .45 mils respectively. As the pump operated, the vibration continued to decrease with all bearing vibrations stabilizing between .18 and .24 mils.

3. The TDAFP cubicle temperature steadied out at a maximum of 97°F, well within the 50 to 120°F normal temperature range. The EEQ Design Basis maximum abnormal excursion, the transient considered for the TDAFP cubicle on a loss of all AC power, is a 58°F increase from 104 to 162°F. Relative Humidity (RH) reached a maximum of 58.6 percent approximately 80 minutes into the run, and then decreased to 53 percent at the end of the two-hour run. The design range is from 10 percent to 75 percent RH.

   The transient response of various plant parameters during this test is illustrated in Figure 8.4.5-1.

LOP PROBLEM SUMMARY

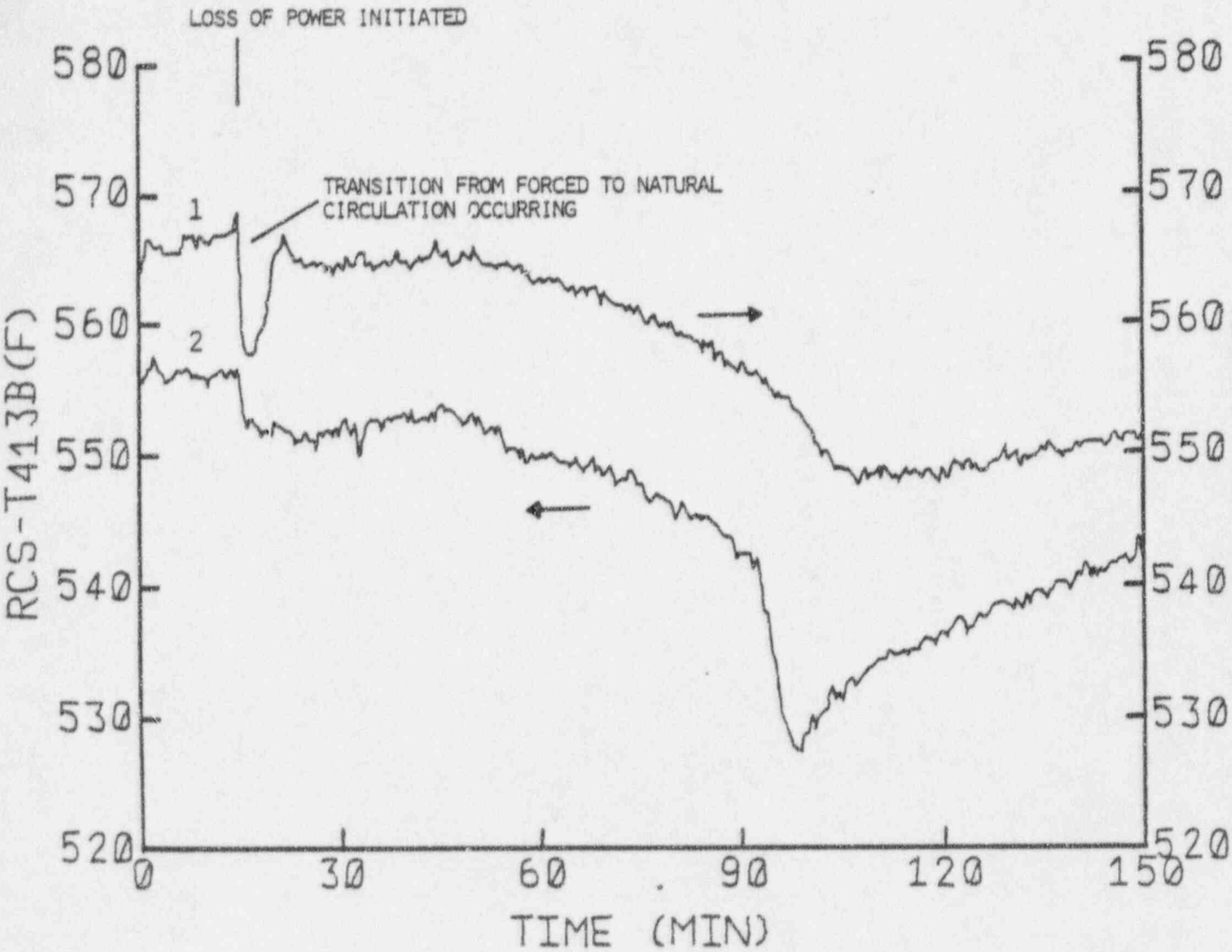
Refer to Appendix D for a summary of problems encountered during the LOP test.



PLOT 1 - PRESSURIZER LEVEL

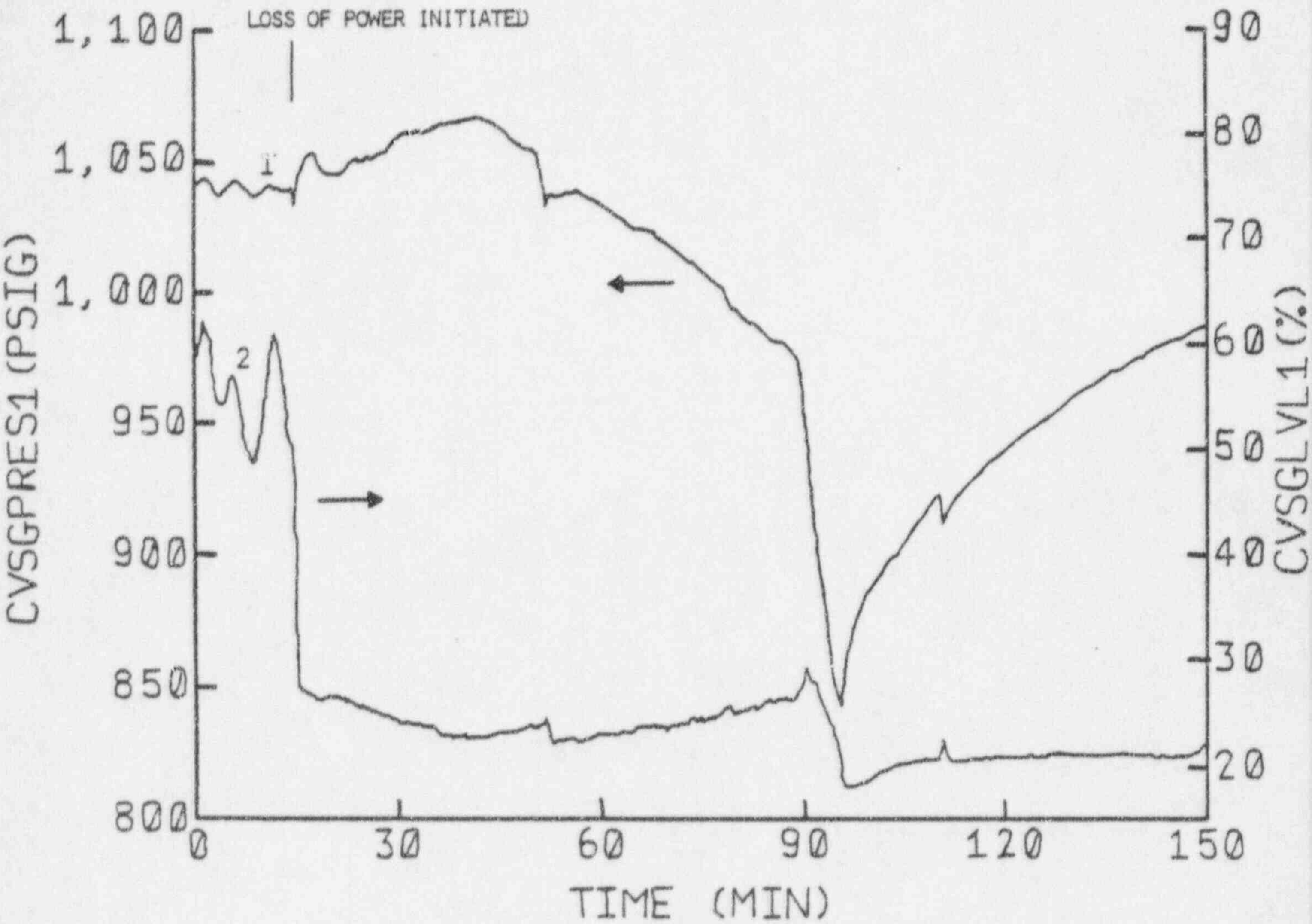
PLOT 2 - REACTOR POWER--POWER RANGE CHANNEL 41

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT LOSS OF POWER TEST	Figure 8.4.5-1 Page 1



PLOT 1 - RCS LOOP 1 WIDE RANGE THOT
PLOT 2 - RCS LOOP 1 WIDE RANGE TCOLD

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT LOSS OF POWER TEST	Figure 8.4.5-1 Page 2



PLOT 1 - STEAM GENERATOR 1 PRESSURE
PLOT 2 - STEAM GENERATOR 1 NARROW RANGE LEVEL

8.4.6 GENERATOR TRIP FROM 100 PERCENT POWER
3-INT-8000, Appendix 8032

OBJECTIVE

The objectives of this test were to:

1. Verify the ability of the primary and secondary plants to sustain a trip from 100 percent load.
2. Verify the ability of control systems to bring the plant to a stable Hot Zero Power (HZP) condition.

DISCUSSION

The test was performed on 04-21-86. Plant load was established at approximately 100 percent. Prior to initiating the trip, rod control, steam generator water level control, pressurizer pressure/level control, and steam generator feed pump speed control were all placed in AUTO. In addition, RCS Tavg, ΔT, steam generator levels, and pressurizer pressure and level were verified to be within the normal full power operating bands. Test personnel were stationed to observe the Main Control Boards, pressurizer safety valves, and steam generator safety valves. A high speed data acquisition system was set up to record key plant parameters. With the plant operating at 100 percent power the test transient was initiated when the generator output breaker was opened by jumpering contacts on the Reverse Power Relay. The generator output breaker opened at 0513 on 04-21-86. Recovery from the resulting turbine trip and reactor trip was in accordance with plant procedures.

The following acceptance criteria applied to the test:

1. All rods fully inserted and nuclear power decreased to less than 15 percent in two seconds.
2. Safety injection did not occur.
3. Pressurizer safety valves did not lift.
4. Steam generator safety valves did not lift.
5. RCS T_{avg} remained above the P12 setpoint of 551°F.

6. Pressurizer pressure remained above 1925 psia.
7. Pressurizer level remained above 17 percent.
8. A reactor trip resulted from the turbine trip.
9. Turbine speed remained less than 1980 rpm.
10. The overall RCS T_{hot} response time was less than 6.0 seconds.

RESULTS

All test acceptance criteria were met:

1. Nuclear power was observed to decrease to less than 15 percent in two seconds.
2. Safety injection did not occur.
3. Pressurizer safety valves did not lift.
4. Steam generator safety valves did not lift.
5. The lowest observed RCS T_{avg} was 552.9°F which was above the acceptance criteria of 551°F.
6. The lowest observed pressurizer pressure was 2003 psia which was above the acceptance criteria of 1925 psia.
7. The lowest observed pressurizer level was 24.6 percent which was above the acceptance criteria of 17.0 percent.
8. A reactor trip resulted from the turbine trip.
9. Peak turbine speed was 1868 rpm which was less than the acceptance criteria of 1980 rpm.
10. The acceptance criteria for the overall RCS hot leg response time was 6.0 seconds. This response time was calculated by measuring the time interval between the point where neutron flux had decreased to 50 percent of its original value to the point where T_{hot} started to decrease.

    This method of calculating the loop response times yielded a 4.0 second response time for loops 1 and 2. Loops 1 and 2 were the two RCS loops where hot leg response time was measured during this test.

After review of the test results with Westinghouse, it was determined that the method used for determination of the overall hot leg response time should have been the time interval between the point where neutron flux had decreased to 50 percent of its original value to the point where the hot leg temperature had decreased by 33 1/3 percent of the initial delta T.

Using this new method to calculate overall hot leg response time resulted in the following:

		New Acceptance Criteria
Loop 1 (w/o pressurizer)	6.7 seconds	$\leq$ 6.8 seconds
Loop 2 (w/ pressurizer)	8.7 seconds	$\leq$ 8.4 seconds

Westinghouse reviewed the failure of the loop 2 hot leg transit time and based on a sensitivity study concluded that the additional 0.3 seconds did not impact the conclusions in the FSAR. However, a reanalysis of five accidents in the FSAR which rely on the overpower and overtemperature delta T reactor trips was determined to be required.

The following five accidents being reanalyzed are:

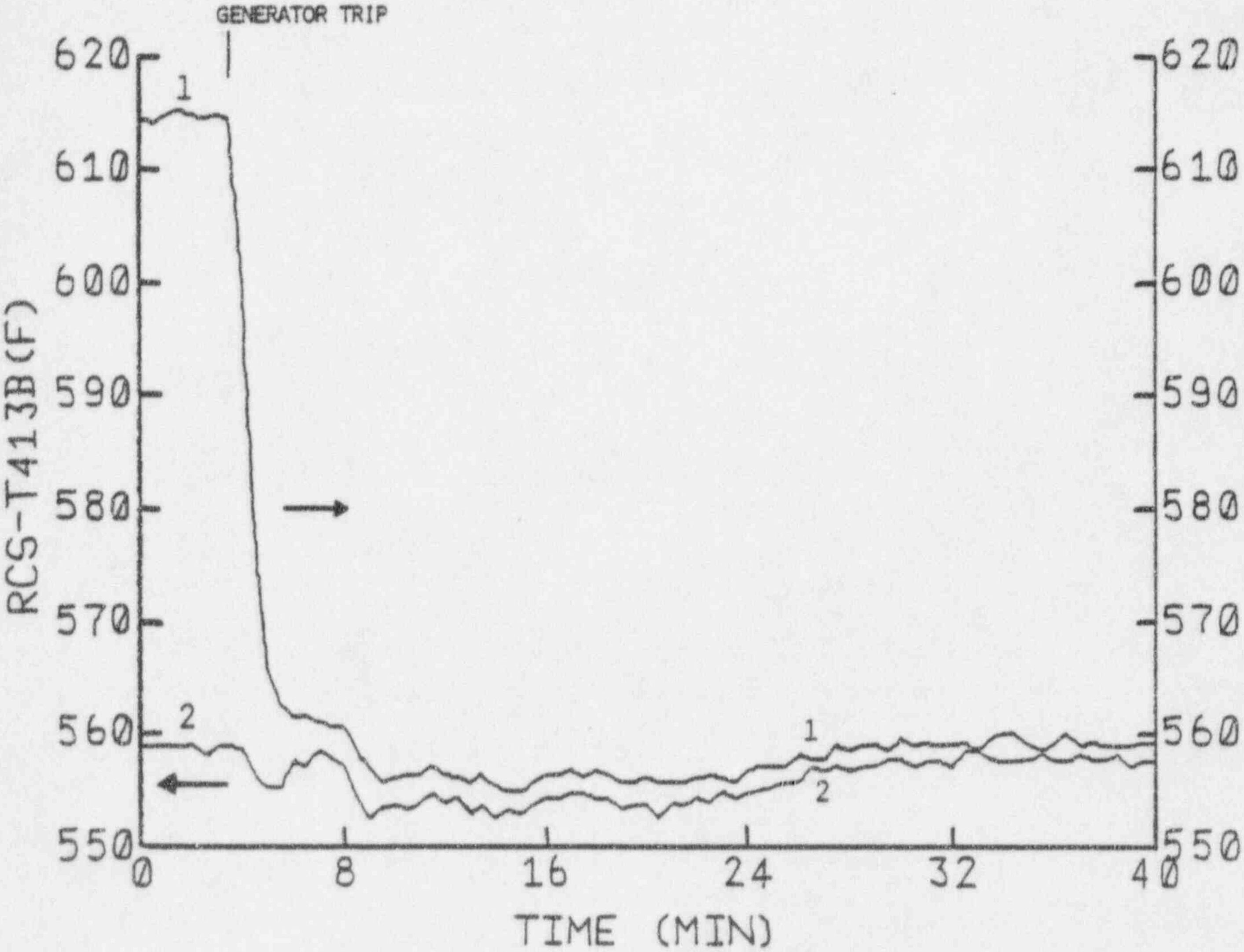
1. Loss of Load
2. Rod Withdrawal at Power
3. RCS Depressurization
4. Steam Line Break at Power
5. Steam Generator Tube Rupture

It is anticipated this reanalysis will be complete on or about 09-01-86.

Figure 8.4.6-1 illustrates the response of various plant parameters to the transient. Table 8.4.6-1 details the response of various plant parameters during the test.

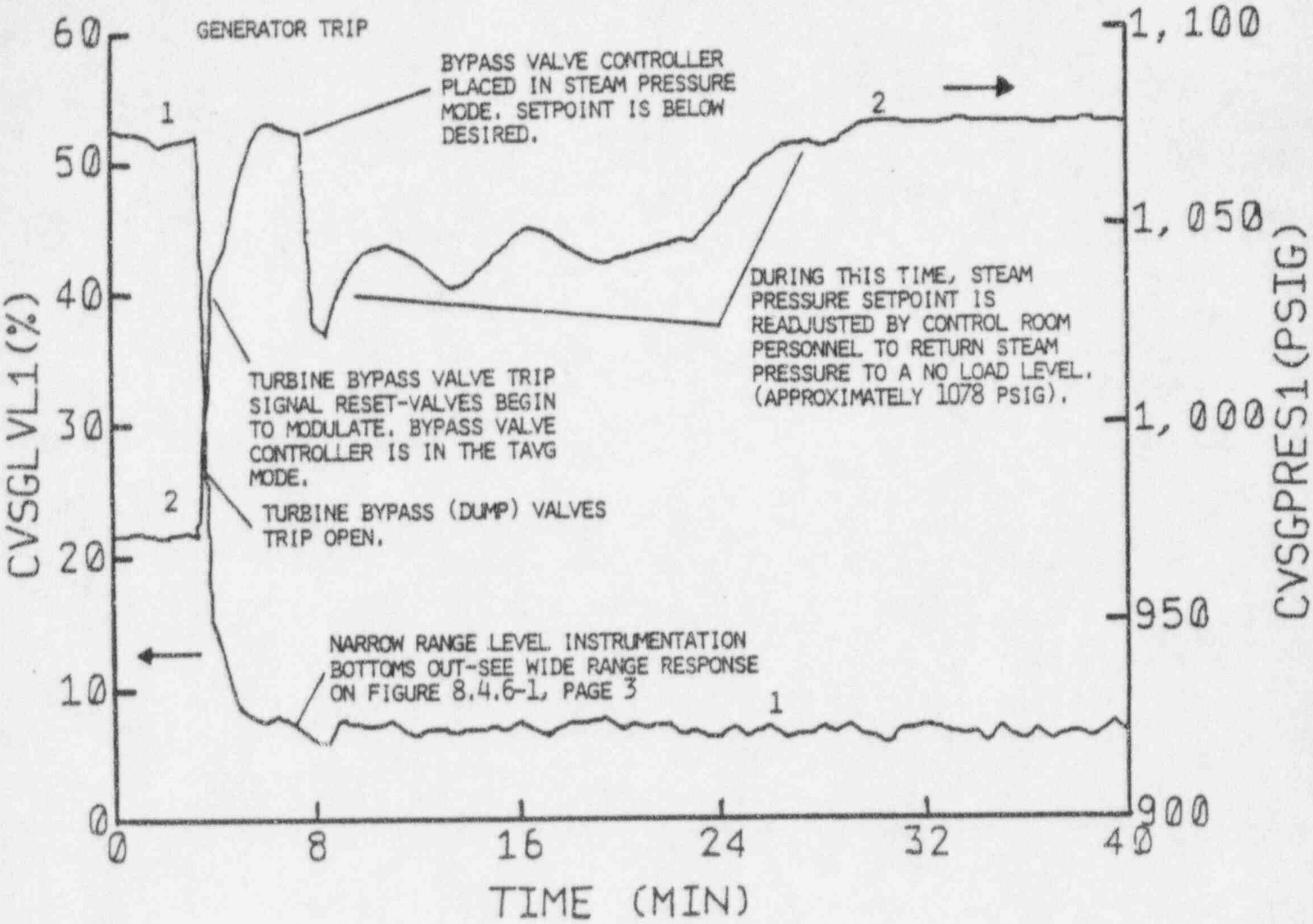
Parameter (Units)	Initial	Minimum	Maximum	Final
Nuclear Power, Channel 41 (%)	99.9	0	99.9	0
T_{avg}, Loop 1 (°F)	587.0	552.9	587.0	558
T_{ref} (°F)	587.5	557.6	587.5	557.7
▲T, Loop 1 (%)	100.6	1.46	100.6	1.46
OP▲T, Loop 1 (%)	109.7	108.6	109.9	109.4
OT▲T, Loop 1 (%)	112.1	109.2	149.9	145.8
Pressurizer Pressure (psia)	2261.3	2003.0	2261.3	2206.3
Pressurizer Level (%)	61.6	24.6	61.9	27.6
Steam Generator NR Level (%)				
Loop 1	51.1	1.8	51.1	5.4
Loop 2	47.7	0	47.7	2.1
Loop 3	50.0	2.4	50.0	3.0
Loop 4	50.3	2.9	50.3	2.9
Steam Flow (MPPH)				
Loop 1	3684	0	3691	0
Loop 2	3684.1	0	3686.5	0
Loop 3	3754.9	0	3754.9	0
Loop 4	3671.9	0	3696.3	0
Steam Generator Pressure (psig)				
Loop 1	978.2	978.2	1082.3	1082.3
Loop 2	976.3	975.6	1079.7	1079.7
Loop 3	970.0	970.0	1075	1075
Loop 4	971.0	971.0	1076	1076
Main Feedwater Flow (MPPH)				
Loop 1	3833	0	3847	0
Loop 2	3725.6	0	4045.4	0
Loop 3	3898.9	0	4826.7	0
Loop 4	3781.7	0	3781.7	0

Note: The above data was taken from a combination of direct indicator observation, data trends, and the temporarily installed high speed data acquisition system.



PLOT 1 - RCS LOOP 1 WIDE RANGE T_{HOT}
PLOT 2 - RCS LOOP 1 WIDE RANGE T_{COLD}

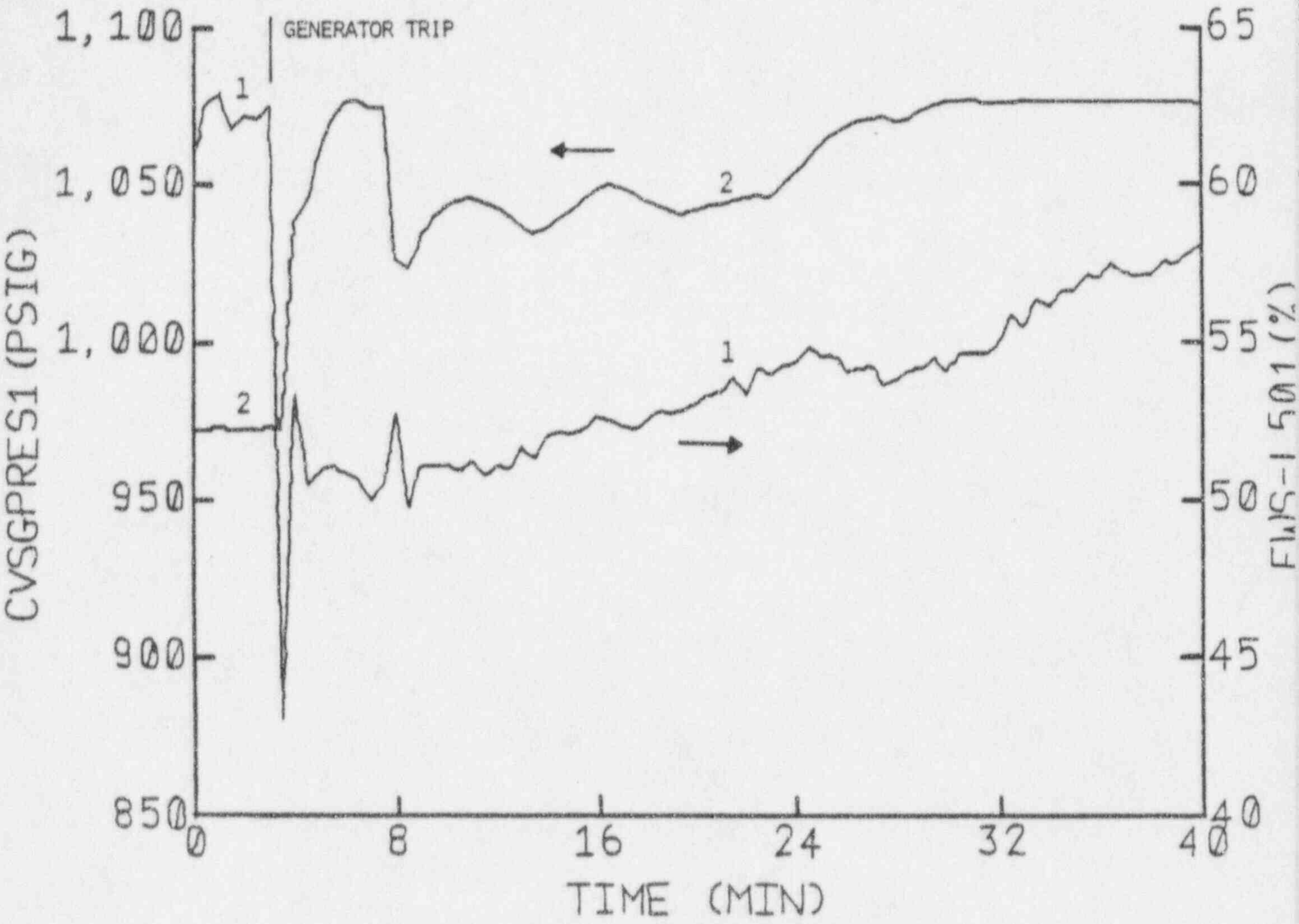
Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT GENERATOR TRIP FROM 100% POWER	Figure 8.4.6-1 Page 1



PLOT 1 - STEAM GENERATOR 1 NARROW RANGE LEVEL

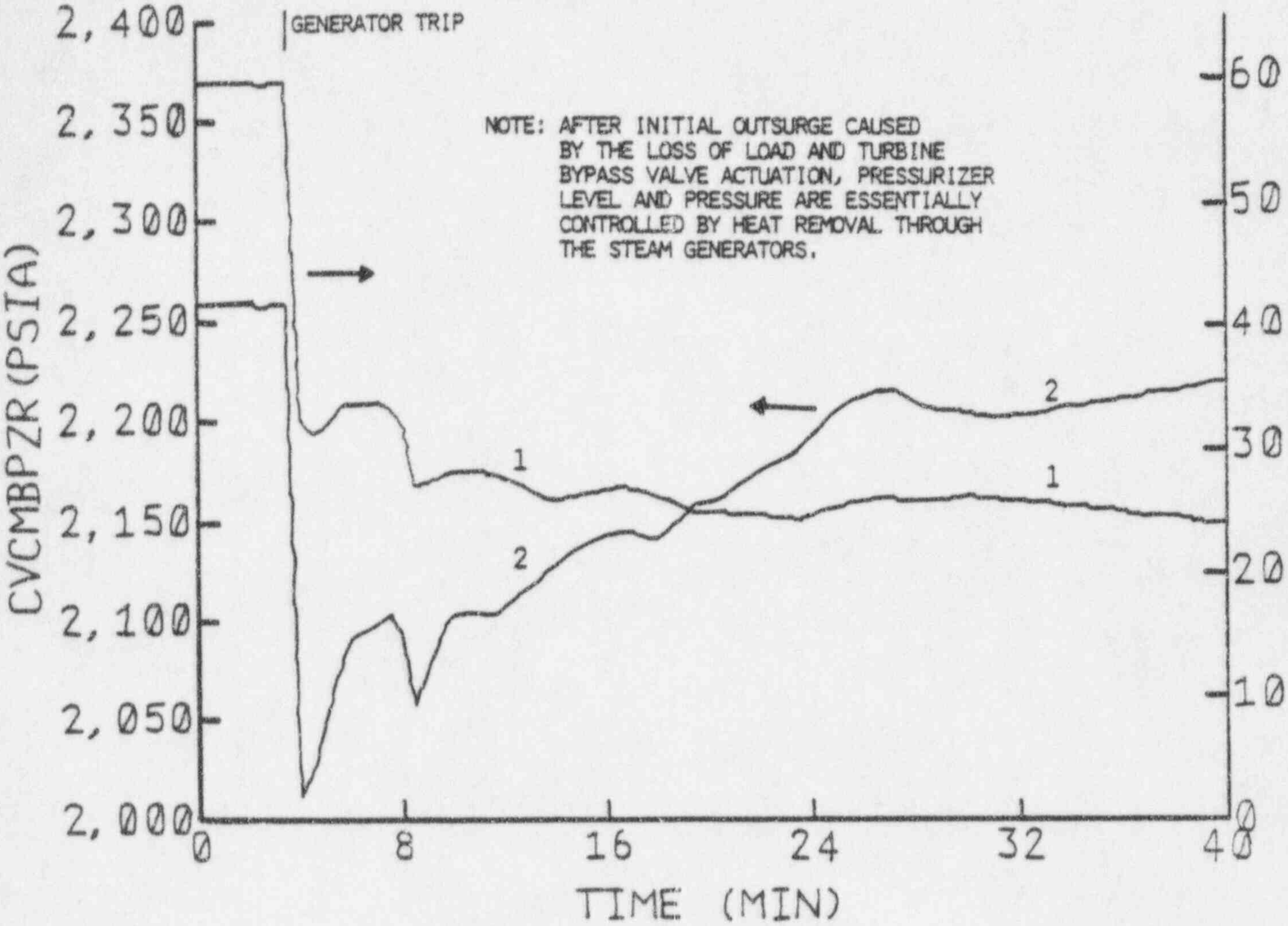
PLOT 2 - STEAM GENERATOR 1 PRESSURE

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT GENERATOR TRIP FROM 100% POWER	Figure 8.4.6-1 Page 2



PLOT 1 - STEAM GENERATOR 1 WIDE RANGE LEVEL
PLOT 2 - STEAM GENERATOR 1 PRESSURE

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT GENERATOR TRIP FROM 100% POWER	Figure 8.4.6-1 Page 3



PLOT 1 - PRESSURIZER LEVEL

PLOT 2 - PRESSURIZER PRESSURE

Millstone Nuclear Power Station Unit No. 3	TYPICAL PLANT TRANSIENT RESPONSE PLOT GENERATOR TRIP FROM 100% POWER	Figure 3.4.6-1 Page 4

8.5.1 CALORIMETRIC

3-INT-8000, Appendix 8001

OBJECTIVE

The objective of this test was to determine, at selected power levels, plant thermal power by means of a manually calculated calorimetric. These calculated values were used as input to the readjustment of the power range (PR) instrumentation. In addition, the manually calculated values were compared against the values from the plant process computer calorimetric program (3P3) as a validation process.

DISCUSSION

The test was conducted at the 25, 30, 40, 50, 75, 90, 98, and 100 percent power plateaus. Once stable plant conditions were established, data was collected on selected plant parameters. In each case, data was taken for a 1 hour period at 5 minute intervals. This data was then reduced and the plant power level calculated.

RESULTS

The results of this test are summarized in Table 8.5.1-1. In each case the process computer (3P3) calculated power levels compared favorably with those from the manual calculations. All objectives of the test were met.

Nominal Power (%)	Manually Calculated Power Level (%)	Computer Calculated Power Level (%)
25	23.48	24.37
30	30.40	30.44
40	41.00	40.75
50	50.69	50.47
75	74.70	74.88
90	89.69	89.58
98	97.26	97.50
100	99.99	99.91

8.5.2 SECONDARY PLANT PERFORMANCE

3-INT-8000, Appendix 8006

OBJECTIVES

The objectives of this test were to:

1. Obtain baseline plant operating data at 30, 40, 50, 75, 90 and 100 percent power plateaus for use in the Secondary Plant Performance Monitoring Program.
2. Determine the turbine-generator and secondary plant performance as an initial condition to conducting performance testing during Warranty Run (3-INT-9000, Appendix 9002).
3. Acquire specific system and component data to permit proper comparison of initial performance test results to turbine-generator manufacturer's guarantee values.

DISCUSSION

The secondary plant performance test made maximum use of permanently installed plant instrumentation and plant Process Computer for data acquisition. In addition precision test instruments were installed to monitor low pressure (LP) turbine exhaust pressures, main turbine control valve positions and makeup flow to the hotwell(s). Local gauges were used for low pressure extraction steam pressures. The test procedure was prepared using the ANSI/ASME PTC-6 Steam Turbine Performance Test Code for guidance. The plant process computer data acquisition software was designed to allow data to be recorded on both hardcopy and magnetic tape.

The test was performed, based on plant status over the period of 02-16-86 through 04-19-86. During the 30, 40, and 50 percent power plateaus, a single data run was performed. Two data runs were performed at 75, 90 and 100 percent power levels. Each test run consisted of four distinct steps; cycle isolation, steady state verification, data collection and data reduction/correction.

The cycle isolation step required a systematic check of drain valves, traps, turbine bypass valves, feedwater heater and MSR emergency drain valves, steam seal system valves and pump minimum flow valves. During this step test personnel used portable infrared imaging equipment, digital heat probes and an ultrasonic leak detector to determine the condition of each isolation point. Plant Trouble Reports were submitted for malfunctioning equipment. The overall purpose of this step was to ensure optimum plant component/system performance existed prior to performance data collection.

Steady state verification consisted of acquiring two hours of computerized and local performance data. Variations in selected parameters were compared to a predetermined steady state projected value. Once test personnel determined steady state conditions existed, the data run portion of the test began.

The test run required two additional hours of steady state data collection. At 75, 90 and 100 percent test points, steam generator blowdown was isolated and auxiliary steam requirements were supplied by the auxiliary boiler. This minimized calculational uncertainty in steam flow to the main turbine.

The final section of the test involved averaging and correcting specific parameters to reference cycle conditions. These corrected test values were compared to target or predicted values at each power level. The predicted values and corrections were developed from performing a computer heat balance simulation for each test power level. These computer based heat balances were based on vendor design data modified to reflect both plant "As Built" configuration and actual system alignment.

After all appropriate corrections were made, corrected net turbine heat rate and generator load were calculated. In between the two test data runs conducted at the 75, 90 and 100 percent power plateaus, turbine control valve positions were modified and then returned to their initial positions. Once steady state conditions were reestablished, the second data run commenced. This process ensured independence of data runs. The corrected heat rates from duplicate test runs were required to agree within 0.25 percent.

RESULTS

During the test, a total of ten test runs were performed as power was escalated from 30 to 100 percent NSSS rated power. Table 8.5.2-1 summarizes the corrected net turbine heat rate and generator load calculated for each test hold point and compares them to the heat balance predictions at each power level. As indicated, overall turbine-generator performance exceeded predicted across the various load ranges. In addition, below is a summary of other major component testing.

1. Condenser

   During this test, no attempt was made to evaluate main condenser thermal performance. This was because the original design information was made obsolete as a result of tube change, during construction, out of the original 70-30 Cu-N tubes with titanium alloy tubes.

2. Feedwater Heaters

   Overall feedwater heater performance was close to predicted at rated power. The final feedwater temperature of 439°F was at or slightly above predicted. The only significant performance deviations were noted at three specific points within the three feedwater heater strings. These are the drain cooler approaches (DCA) on 1A, 1B and 1C heaters, subcooler approaches (SC) on 4A, 4B and 4C heaters, and terminal temperature differences (TTD) on 6A, 6B and 6C heaters. Suspected causes and remedial actions are as follow:

Heater No. Problem	Suspected Cause	Recommended Action
1A, 1B, 1C High DCA Temperature	Steam/Vapor bypass into drain cooler inlet	Raise level to break the vapor bypass; reestablish at proper operating level
4A, 4B, 4C Low SC approach temperature	Higher than normal operating level	None; the high level's needed to maintain drain pump NPSH
6A, 6B, 6C Low TTD	Drain level low in heaters	Establish and maintain loop seals

Further in-service testing is planned to establish proper DCA values on 1A, 1B and 1C feedwater heaters. Trouble Plant maintenance requests have have been issued to ensure loop seals on 6A, 6B and 6C heaters are filled. Table 8.5.2-2 provides a comparison of test to predicted performance data for all three feedwater heater strings.

3. <u>Moisture Separator/Reheater Performance</u>

Moisture Separator/Reheater (MSR) performance was reasonably close to predicted performance at rated power. Test data for the two MSRs showed remarkable similarity. This indicates an approximately even flow and duty split between the MSRs. Table 8.5.2-3 gives a comparison of tests to predicted performance values. As noted on this table, most test values are lower than predicted. It is suspected that the reason is more likely a result of difficulty in heat balance modeling than any performance deficiency.

The key MSR performance index is the thermal temperature difference (TTD). A lower TTD indicates more efficient heat transfer.

TEST DATE	TEST LOAD % RATED POWER	CORRECTED NET TURBINE HEAT RATE[2] TEST	CORRECTED NET TURBINE HEAT RATE[2] PREDICTED	CORRECTED GENERATOR LOAD TEST	CORRECTED GENERATOR LOAD PREDICTED
2/16/86	30	11646	12403±1240	308.8	290±29
3/14-15/86	40	11283	11462±1146	424.2	418±42
3/17/86	50	10700	10914±1091	548.7	538±54
3/26/86	75	1[illegible]2	10108±202	868.1	861±18
3/29-30/86	75	10012	10108±202	869.5	861±18
4/12/86	75[1]	9982	10042±201	879.8	875±18
4/15/86	90	9776	9867±197	1077.7	1067±21
4/16/86	90	9805	9867±197	1074.0	1067±21
4/18/86	100	9722	9790±98	1202.0	1194±12
4/19/86	100	9741	9790±98	1199.9	1194±12

1. First test run with Moisture Separator Reheaters in service

2. $NTHR = \frac{\text{SG Power } (MW_{TH})}{\text{Gross Electric Power } (MW_E)} (3412.141 \text{ BTU/KWH})$

Millstone Nuclear Power Station Unit No. 3	TURBINE-GENERATOR PERFORMANCE DATA SECONDARY PLANT PERFORMANCE TESTING	Table 8.5.2-1

HEATER NUMBER	TTD (°F) TEST	TTD (°F) PREDICTED	DCA (°F) TEST	DCA (°F) PREDICTED	TR (°F) TEST	TR (°F) PREDICTED
1A	3.0	3.1	55.4	9.6	74.8	75.3
1B	2.7	3.1	68.2	9.6	77.6	75.3
1C	3.0	3.1	43.8	9.6	78.4	75.3
2A	4.4	7.3	5.3	10.2	39.4	41.9
2B	6.8	7.3	4.9	10.2	41.2	41.9
2C	6.9	7.3	4.9	10.2	37.0	41.9
3A	2.5	3.7	5.7	10.6	40.4	38.1
3B	3.5	3.7	7.1	10.6	41.0	38.1
3C	3.2	3.7	4.5	10.6	39.9	38.1
4A	6.3	5.5	37.4	56.3	64.2	66.2
4B	8.6	5.5	37.8	56.3	62.5	66.2
4C	8.1	5.5	39.2	56.3	65.3	66.2
5A	0.4	3.0	10.1	12.8	64.3	62.8
5B	4.8	3.0	6.1	12.8	65.0	62.8
5C	5.5	3.0	7.2	12.8	63.6	62.8
6A	-0.6	3.1	N/A	N/A	69.8	61.5
6B	-1.0	3.1	N/A	N/A	67.9	61.5
6C	-0.3	3.1	N/A	N/A	68.7	61.5

*Data from Test Hold Point 100.2.1 at 100% Rated Power

		REHEATER A	REHEATER B
STEAM FLOW	test:	607.18	606.52
(Klb_m/hr)	predicted:	660.02	660.02
INLET TEMP	test:	365.9	366.3
(^oF)	predicted:	369.5	369.5
OUTLET TEMP	test:	504.6	504.2
(^oF)	predicted:	513.18	513.18
TEMP RISE	test:	138.7	137.9
(^oF)	predicted:	143.7	143.7
SUPERHEAT	test:	139.5	138.9
(^oF)	predicted:	145.5	145.5
DRAIN TEMP	test:	525.8	523.6
(^oF)	predicted:	537.7	537.7
TTD	test:	21.2	19.4
(^oF)	predicted:	23.9	23.9

*Data from Test Point 100.2.1 at 100% Rated Power

8.5.3 RADIATION MONITORING SYSTEM
3-INT-8000, Appendix 8007

OBJECTIVE

The objectives of this test were to:

1. Measure and document the gamma and neutron radiation levels in selected areas of Millstone Unit 3 during power ascension testing.
2. Determine locations where permanent shielding or engineered barriers (i.e., high radiation area doors, labyrinth entrances, etc.), are deficient or not in conformance with the Millstone Unit 3 FSAR.
3. Compare permanently installed area radiation monitor readings to portable radiation instrumentation results. Compare selected permanently installed process monitor readings with grab sample results.
4. Identify high radiation areas and verify access is controlled as required.
5. Determine neutron spectrum factors for various areas inside the containment building.
6. Log the permanently installed area radiation monitor alarms at the 100 percent reactor power test plateau, the reason for the alarms, and their final disposition.

DISCUSSION

A total of 378 Radiation Base Points (RBPs) were selected to be surveyed at each power level (zero, 30, 50, 75, 90 and 100 percent) during the power ascension testing program. Survey points were chosen at each installed radiation monitor location, along all shield walls, at gate or labyrinth entrances to cubicles projected to become High Radiation Areas, and along boundaries where the prescribed FSAR dose rate changes. The RBP survey locations were labeled with sequentially numbered 11" X 14" signs to aid survey personnel and ensure sampling reproducibility.

A training program was developed and administered to all survey personnel. This training program outlined survey methodology, documentation requirements, ALARA considerations, expected survey instrument responses to the containment subatmospheric environment and nitrogen-16 gamma fields, containment subatmospheric entrance/egress procedures, and biological shield survey experiences at other nuclear power plants.

A mock survey was conducted in containment prior to initial criticality and the drawing of a containment vacuum. This survey was performed in BioPak-60 units to simulate realistic survey conditions. Special attention was given to the movable incore detector regions of containment and the overexposure hazards associated with this "Extra High Radiation Area." This mock survey was used to develop a survey man-hour estimate which would be used to develop a man-rem estimate for the surveys done at power. In addition, the water jugs used for the neutron spectrum factor determination were placed in containment prior to initial criticality and the establishment of containment vacuum. This was done in a further attempt to maintain personnel exposure ALARA and to lower the number of personnel entries required into the containment subatmospheric environment.

In addition to the general surveys conducted at the 378 RBPs, an extensive radiation monitor/TLD/survey meter comparison survey was conducted on two containment radiation monitors. One survey was conducted on 3RMS-RE32 at 90 percent reactor power, and the other on 3RMS-RE35 at 100 percent reactor power. The survey consisted of comparing extrapolated gamma TLD results and various survey meter readings with the plant radiation monitoring system computer readout information.

In addition, a survey meter/installed radiation area monitor comparison survey was also conducted on 11 area monitors located in the Auxiliary, Waste Disposal, and Fuel Buildings. The survey consisted of simultaneously exposing the installed area monitor and selected survey instruments to a Cs-137 source and comparing the various readouts.

Experiments were performed at the University of Lowell in order to determine station survey instrument, TLD, and pocket ionization chamber response to the highly energetic nitrogen-16 gamma radiation. Experiments were also performed at the station to study survey instrument response to subatmospheric conditions. Since both of these conditions exist in containment, it was desirable to determine which instruments responded in the most accurate and reliable manner. Neutron survey meters were sent to the University of Michigan for analysis and calibration using a heavy-water moderated Cf-252 source.

The following installed process monitor readings were compared to grab sample results. This was done to determine the accuracy of the installed process monitors.

CHS69	RCS Gross Activity/Specific Nuclide Monitor
HVQ49	ESF Building Ventilation Monitor
HVR10B	Ventilation Stack Monitor
LWS70	Radioactive Liquid Waste Monitor
ARC21	Steam Jet Air Ejector Monitor
CMS22	Containment Atmosphere Monitor
DAS50	Turbine Building Sump Monitor
HVC16	Control Building Ventilation Inlet Monitor

These process monitors do not represent all process monitors but represent monitors that sample important plant processes and/or are required by Plant Technical Specifications.

RESULTS

1. Shield Surveys

   A. Zero Percent Power

   The inside containment portions were conducted on 12-13-85. The outside containment portions were conducted on 12-23-85 and 12-30-85. All surveys were conducted prior to initial criticality and were intended to verify no sources of radiation were present that would affect subsequent surveys. There were no abnormal findings.

   B. 30 Percent Power

   This portion was conducted on 02-15-86. This survey indicated steam generator loop general area radiation levels of up to 2.6 R/hr (gamma). Contact readings on the RCS loop crossover lines (coolant line connecting reactor coolant pump to steam generator cold leg) read between 7.9 to 9.0 R/hr (gamma). No appreciable neutron dose rates in these areas were observed. In the loop areas on the 24'6" elevation of the containment, readings were 700 to 800 mR/hr (gamma). These rates were consistent between loop areas on this elevation. Surveys of the -11'3" elevation of the containment produced readings of 1800 mRem/hr (neutron).

   A neutron radiation area was discovered outside the containment equipment hatch on top of the Hydrogen Recombiner Building. This area was posted and levels never exceeded the 15 mRem/hr neutron limits of the FSAR. Also, an additional radiation area was discovered on the 43'6" elevation of the auxiliary building. This was determined to have been caused by radiation streaming through a penetration in the volume control tank shield wall. Other than these two items, the results of the 30 percent survey were as expected.

C. <u>50 Percent Power</u>

This portion was conducted on 03-17-86. This survey indicated steam generator loop general area readings of up to 5.0 R/hr (gamma). Contact readings on the RCS loop crossover lines read between 14.0 to 18.0 R/hr (gamma). No appreciable neutron dose rates in these areas were observed. In the loop areas on the 24'6" elevation of the containment, readings were approximately 2 R/hr. Again, the readings between loops were very consistent. Surveys of the -11'3" elevation of the containment produced readings of 500 mRem/hr (neutron). This same survey location at 30 percent reactor power indicated 1800 mRem/hr neutron. It appears that the neutron reading taken at 50 percent power was not in the exact location as the survey point taken at 30 percent power. It should be noted that at 100 percent power the surveyor, while approaching this survey location, detected neutron levels exceeding 1000 mRem/hr.

All survey readings were within the levels discussed in the Millstone Unit 3 FSAR.

D. <u>75 Percent Power</u>

This portion was conducted on 03-26-86. At the time of the survey, the containment personnel air lock inner door was inoperable making the containment inaccessible. Only the points outside the containment were surveyed. All survey points were within specification except for point number 109 which is located adjacent to 3CHS-RE69 (failed fuel monitor) on the 4'6" elevation of the auxiliary building. Upon evaluation, the larger than expected dose rate was the result of the letdown piping on 3CHS-RE69 and not due to a deficiency in adjacent shield walls.

E. <u>90 Percent Power</u>

This portion was conducted on 04-15-86. Due to ALARA concerns, the containment survey points were eliminated from this power level. The 90 percent radiation values were considered redundant to the values scheduled to be taken at 100 percent power. No new problems were encountered during the out of containment portion of the survey.

F. <u>100 Percent Power</u>

This portion was conducted on 04-18-86. For ALARA considerations and because previous readings between loops had been similar, only one loop area on the 24'6" elevation of containment was surveyed. General area readings of between 7 to 10 R/hr (gamma) were measured. Two loops on the 3'8" elevation were surveyed from 10' outside the loop area using a teletector and readings of 30 R/hr (gamma) were observed. From this 10' approach distance to the loops at elevation 3'8", no appreciable neutron dose rates were observed. Neutron radiation levels on -11'3" elevation were measured in excess of 1000 mRem/hr. No further neutron rad level quantification was attempted at the -11'3" evaluation in order to minimize exposure to the survey personnel. Outside containment, five survey points were determined to be in excess of the FSAR established limits. In each case, these discrepancies were the result of adjacent component piping and not deficiencies in shielding. At the 100 percent plateau, two monitors were alarming because the actual normal exceeded the expected normal and setpoints for these monitors were revised.

2. Installed Area Radiation Monitor Evaluation

The permanently installed area radiation monitoring system was evaluated at the 90 and 100 percent power plateaus. This evaluation was conducted during the period from 04-17-86 to 04-30-86. This evaluation was done to verify the response of the area radiation monitors at other than very low levels of radiation. This evaluation, plus comparison of containment area monitors at 100 percent power, indicated a good correlation between radiation monitor readings and survey meter readings.

3. Installed Process Radiation Monitor Evaluation

The comparison of process radiation monitor readings to survey results indicates that process monitors show accurate radiation trends, but are not all accurate in determining the absolute value of radiation in the process. Monitors that require accuracy do provide accurate readings.

4. Neutron Spectrum Factor Determination

TLDs used to determine the neutron spectrum factors in containment have been removed and data reduction is in progress. The results of this analysis will be utilized to enhance the Unit 3 neutron dosimetry program through the determination of accurate quality factors.

5. Conclusion

This test verified that radiation levels in the plant are as stated in the FSAR with the exception of a radiation area caused by the letdown piping to the failed fuel monitor, 3CHS-RE69. A Plant Modification Request has been submitted to provide permanent shielding of the letdown piping, and additional shielding is being installed in various identified areas to keep exposure ALARA.

Comparison of area radiation monitor readings to survey results shows that the area monitors provide a good indication of radiation levels. Some process radiation monitor results were not as accurate. They do, however, provide a good indication of trends in the monitored process. Significant is the fact that the liquid waste discharge monitor, 3LWS-RE70, the failed fuel monitor, 3CHS-RE69, and the containment atmosphere monitor, 3CMS-RE22 do provide accurate radiation levels. Investigation is continuing on other process monitors to provide more accurate source term calibration correlatable to field results.

Approximately 1.3 man-rem and 260 man-hours were expended in performing the Reactor Power Shield Survey. An ALARA review of the job estimated that 3.795 man-rem would be expended for the entire survey. Because observed dose rates were lower than expected, and survey points were deleted at various power plateaus, less exposure was received than originally predicted.

8.5.4 VENTILATION SYSTEM OPERABILITY
3-INT-8000, Appendix 8008

OBJECTIVE

The objectives of the test were to:

1. Verify that the containment air ventilation systems (containment air recirculation system and CRDM cooling systems) are capable of maintaining the containment air temperature less than the EEQ equipment design limit of 90°F.
2. Verify that the Main Steam Valve Building (MSVB) ventilation system can maintain the MSVB within the EEQ equipment design range of 50°F to 104°F.

The acceptance criteria for the test was to verify that the containment air ventilation systems maintain containment air temperature within the Technical Specification range of 80°F to 120°F.

DISCUSSION

Temperature data for the containment was monitored using 41 permanent RTDs located throughout the containment structure. In addition, the reactor plant chilled water (CDS) temperature to the containment air coolers were monitored as well as containment pressure, outside ambient air temperature, and reactor power level. Data was taken at 24 hour intervals during power ascension testing.

Temperature data for the MSVB was monitored using 5 permanent RTDs located at various levels in the structure. In addition, outside ambient air temperature and reactor power level were also monitored. Data was taken at 24 hour intervals throughout power ascension testing.

RESULTS

At the 100 percent power level, all upper elevation areas in the containment exceeded the EEQ equipment design temperature of 90°F by an average of 15°F. However, the Technical Specification upper temperature limit of 120°F (TS 3.6.1.5) was satisfactorily met at all power levels.

In the MSVB, the area between the main steam isolation valves exceeded the upper EEQ equipment design temperature of 104°F by an average of 3°F. All other building areas were maintained within the required limits.

Temperature excursions similar to the above were noted during precore hot functional testing. At that time, plant deficiency UNS 6300 was written to cover the containment excursion and UNS 6452 was written to cover the MSVB. These prior deficiencies were considered enveloping for the power ascension temperature deviations and no new deficiencies were generated. These deficiencies, while not affecting equipment operability, are being reviewed by Engineering to assess the impact on EEQ qualified life of various equipment in the noted areas. In addition, per the requirements of the Facility Operation License, Section 2.C.3, Millstone 3 must, prior to startup following the first refueling, recalculate the qualified service lives of all applicable components located in the containment. These calculations are to be based on actual temperature readings over the first fuel cycle.

RCS CHEMISTRY ATTRIBUTE	SPECIFICATION LIMIT
pH	4.2-10.2
Conductivity	N/A-Expected range: 1.0-40.0 uMhos/cm
Dissolved Oxygen	≤100 ppb
Chlorine	≤150 ppb
Fluoride	≤150 ppb
Dissolved Hydrogen	25-50 cc/kg water
Lithium	0.2-2.2 ppm as Li
Boron	0-4000 ppm
Silica	≤1000 ppb
Aluminum	≤50 ppb
Calcium & Magnesium	≤50 ppb
Magnesium	≤25 ppb
Specific Activity (D.E. I-131)	≤1.0 uCi/gm
Gross Activity	As required by procedure

POWER LEVEL (%)	30	50
POWER LEVEL (MWT)	1023	1706
SAMPLE DATE	02-16-86	03-18-86
SAMPLE TIME	1648	0850

ANALYSIS RESULTS			UNITS
pH/temperature	5.95/26.1	6.26/24.0	pH/°C
conductivity/temperature	25.8/25.5	21.5/24.0	uMhos/cm/°C
Dissolved oxygen	<5.0	<5.0	ppb
Chloride	<10	<1	ppb
Fluoride	<20	<1	ppb
Dissolved Hydrogen	40	36	cc/kg
Lithium	1.6	1.73	ppm
Boron	1297	1201	ppm
Silica	423	450	ppb
Aluminum	14.4	21.0	ppb
Calcium + Magnesium	1.5	<1	ppb
Magnesium	<1	<1	ppb
D.E. I-131	1.94E-04	2.66E-04	uCi/gm
Gross Activity	3.14E-02	5.63E-03	uCi/gm

POWER LEVEL (%)	75	100
POWER LEVEL (MWT)	2558	3411
SAMPLE DATE	03-27-86	04-19-86
SAMPLE TIME	0840	0900

ANALYSIS RESULTS			UNITS
pH/temperature	6.53/26.0	6.48/26.9	pH/°C
conductivity/temperature	23.1/26.0	22.2/25.0	uMhos/cm/°C
Dissolved oxygen	<5	<5	ppb
Chloride	<10	<10	ppb
Fluoride	<20	<20	ppb
Dissolved Hydrogen	35.5	43.5	cc/kg
Lithium	2.02	1.98	ppm
Boron	1133	1076	ppm
Silica	388	335	ppb
Aluminum	25.0	8.0	ppb
Calcium + Magnesium	1.52	10.9	ppb
Magnesium	0.47	2.8	ppb
D.E. I-131	6.79E-04	8.38E-04	uCi/gm
Gross Activity	1.25E-01	1.579E-01	uCi/gm

8.5.6 NEUTRON SHIELD TANK COOLING TEST

3-INT-8000, Appendix 8010

OBJECTIVE

The objective of this test was to verify that the Neutron Shield Tank Cooling System performs within design limitations at 100 percent power. The shield tank consists of an annular tank surrounding the reactor vessel. Its purpose is to serve as neutron shielding to adjacent areas of the containment structure. Cooling water in the tank circulates under natural convection from the tank to the neutron shield tank cooler where it is cooled with water from the reactor plant chilled water system. In addition to the shielding function, the tank serves as the support structure for the reactor vessel.

DISCUSSION

The test was performed on 02-16-86, 03-17-86, 03-26-86, 04-15-86, and 04-18-86 at plant power levels of 30, 50, 75, 90, and 100 percent, respectively. The temperature of the neutron shield tank was monitored and recorded at each power plateau during the power ascension. The shield tank outlet temperature (inlet to the neutron shield tank cooler) and the neutron shield tank return water temperature (outlet from the neutron shield tank cooler) was recorded at each power level and compared against the acceptance criteria.

RESULTS

All data obtained met the acceptance criteria which required that the tank temperature be maintained less than 135°F at all power levels. The highest neutron shield tank temperature recorded during the test was 123°F.

8.5.7 CONTAINMENT PENETRATION TEMPERATURE MONITORING
3-INT-8000, Appendix 8011

OBJECTIVE

The purpose of this test was to verify that the hot containment piping penetrations were within design temperature during power ascension and at full reactor power. The penetration coolers consist of liquid cooled annular structures surrounding selected hot containment piping penetrations. They form an integral portion of the piping penetrations and run the entire depth of the containment structure. The coolers are supplied cooling water from the reactor plant component cooling water system. Liquid cooled penetrations are used on the main steam, feedwater, RCS letdown, steam generator blowdown and steam supply lines to the turbine driven auxiliary feedwater system.

DISCUSSION

The test was performed on 02-17-86, 03-17-86, 03-27-86, 04-15-86 and 04-18-86 at plant power levels of 30, 50, 75, 90, and 100 percent, respectively. With the reactor plant component cooling flow at a minimum to the penetration coolers, the containment concrete temperature adjacent to the penetration was measured. Data was obtained at four points (90° apart) on each penetration.

RESULTS

All data met the acceptance criteria which required all temperatures to be less than 150°F. Actual temperatures were between 58°F and 140°F.

8.5.8 TURBINE PLANT COMPONENT COOLING WATER SYSTEM BALANCING
3-INT-8000, Appendix 8019

OBJECTIVE

The objective of this test was to verify adequate flow balancing of the turbine plant component cooling water system (CCS) at 100 percent power.

DISCUSSION

The test was conducted as plant conditions permitted over the period from 02-08-86 to 05-06-86. The CCS flow rates to system heat exchangers were initially adjusted as part of the preoperational test program. These flows were then modified in response to increased turbine plant heat loads, at 30 and 100 percent power. Final flow modifications were completed at 100 percent power and the final throttle valve positions were recorded in the test appendix for future reference. Flows were verified to be adequate by monitoring temperatures and flows at various system locations using permanently installed and temporary instrumentation.

RESULTS

The objective of this test was satisfied. Adequate cooling water flow was verified to all CCS heat loads.

8.5.9 PIPING FLUID TRANSIENT VIBRATION MONITORING
3-INT-8000, Appendix 8029

OBJECTIVE

The objective of this test was to verify, by visual inspection and instrumented measurement, the vibrational response of plant piping systems during selected fluid transient events that are credible within plant operating modes.

DISCUSSION

The test was conducted over the period of 04-21-86 to 04-24-86.

The transients selected for this test were:
1. Main turbine trip
2. Closure of the feedwater isolation valves

During each transient event, qualified test personnel observed the response of piping and associated supports. In addition, temporary test instrumentation was installed at selected pipe supports.

RESULTS

All test acceptance criteria for the main turbine trip and feedwater isolation valve closure transients were met. No permanent deformation or damage was observed.

8.5.10 THERMAL EXPANSION AND RESTRAINT MONITORING
3-INT-8000, Appendix 8034

OBJECTIVE

The objective of this test was to verify, by visual inspection and instrumented measurement, that the feedwater and main steam piping systems are free to thermally expand as designed.

DISCUSSION

This test was conducted over the period of 02-03-86 through 04-21-86. The inspections were performed at plateaus of zero, 30, 50, 75, and 100 percent power levels. Test data which was collected by visual inspection, system walkdowns and instrumented measurement, was compared to design ranges. Discrepancies (piping interferences or snubber indication out of design range) were evaluated and resolved by Engineering.

RESULTS

All potential contact of piping with structures, components and conduit was evaluated by Engineering. This evaluation noted no potential interference which could restrict piping or components from expanding. Furthermore, all data points outside of the predetermined acceptance criteria were evaluated and found to be acceptable by Engineering.

8.5.11 LOOSE PARTS MONITORING

3-INT-8000, Appendix 8035

OBJECTIVE

The objectives of this test were to:

1. Obtain baseline system signal data during the power ascension test phase.
2. Obtain baseline system signal data with the plant at full power.
3. Determine the approximate frequency of spurious alarms.

DISCUSSION

The test was performed on 02-16-86, 03-17-86, 03-26-86, 04-16-86, and 04-18-86 with the plant power at levels of 30, 50 75, 90, and 100 percent, respectively.

Baseline signal data was obtained by using a spectrum analyzer which was connected to the auxiliary output jack on the Loose Parts Monitoring system (LPM) cabinet. Hardcopy spectrum analysis data was obtained for all eight monitoring channels during the testing plateaus. The frequency of spurious alarms caused by the noise of normal plant operation was also monitored.

The LPM was supplied by Rockwell and consists of a monitoring cabinet with audio output system and integral cassette recorder. There are eight accelerometers located on the primary system: two located on the reactor vessel head, two located on the lower reactor vessel and one on each steam generator in the channel head area. The system has been modified by the addition of a 1500 hertz bandpass filter to enhance the capabilities to detect loose parts of a large mass (30 pounds).

RESULTS

All baseline LPM signal data was obtained with no problems encountered. The frequency of spurious alarms was approximately three per day. In accordance with Engineering direction provided following the phase five testing (see Section 5.11), the gains of the 1500 hertz filter were adjusted for the upper and lower reactor vessel LPM channels to reduce the number of spurious alarms. No adjustments were required on the remaining channels. The alert levels for power ascension and initial commercial operation were determined to be between 0.1 to 0.38 ft-lbs for a 30 pound object impacting 3 feet from the transducer and between 0.01 and 0.08 ft-lbs for a 0.25 pound object impacting 3 feet from a transducer. Additional testing indicated that the alert levels may need to be increased further to obtain a false alarm rate of approximately one per day. It is anticipated that any further adjustments will result in alert levels no greater than 0.5 ft-lb kinetic energy, 3 feet away from a transducer.

9.0 WARRANTY RUN TEST SUMMARY
3-INT-9000

This test proved the reliability of the NSSS system. The plant was maintained at rated power for 100 hours. Appropriate data was recorded to allow plant performance to be analyzed. The warranty run was conducted from 04-25-86 to 04-29-86.

9.1 CALORIMETRIC

3-INT-9000, Appendix 9001

OBJECTIVE

The objective of this test was to determine plant thermal power by means of the plant process computer calorimetric calculation, plant process computer data collection with manual calculation, and manual data collection with manual calculation. These calculated values were used as input to the readjustment of the power range (PR) instrumentation.

DISCUSSION

The test was conducted at 100 percent power. Once stable plant conditions were established, data was collected on selected plant parameters. In each case, data was taken for 15 minutes at 5 minute intervals. This data was then reduced and the plant power level calculated.

RESULTS

The results of this test are as follows:

Plant Process Computer Calorimetric Calculation	100.5%
Plant Process Computer Data Collection with manual data reduction	100.1%
Manual Data Collection with Manual Calculation	100.2%

In each case the calculated power levels compared favorably with the power range instrumentation. All objectives of this test were met. There was no formal acceptance criteria for this test.

9.2 SECONDARY PLANT PERFORMANCE
3-INT-9000, Appendix 9002

OBJECTIVES

The objectives of the test were to:

1. Obtain performance data needed to properly compare actual performance to General Electric Company (GE) warranty values for the turbine generator.
2. Acquire baseline operating data at rated power for routine monitoring and reporting requirements.
3. Estimate the loss of efficiency associated with operating the turbine in the full arc steam admission mode.

DISCUSSION

This test was performed over the period 04-19-86 to 04-29-86 with the unit operation at a 100 percent power level. The test procedure was prepared using the ANSI/ASME PTC-6 Steam Turbine Performance Test Code for guidance. Prior to performing the test, an uncertainty analysis on all heat rate inputs was performed. Heat rate uncertainty was determined to be approximately 0.7 percent. Overall test uncertainty was calculated at less than 1.0 percent.

Test prerequisites required calibration checks of selected plant instrumentation within 30 days of testing. During testing, steam generator blowdown was isolated and auxiliary steam was supplied by the auxiliary boiler. The test procedure required inventory losses of less than 0.25 percent of valve wide open (VWO) main turbine throttle flow. In addition, cycle component alignment was verified and a systematic isolation check was completed within two hours of testing.

Each test point required four hours of data acquisition. The first two hours were taken to verify steady-state operation. The plant process computer provided most data acquisition needs with very limited local data taking required. Duplicate test runs were conducted with turbine control valve positions upset between tests.

Corrected test heat rates from duplicate tests were compared according to ASME PTC-6 which requires agreement of parallel runs within 0.25 percent.

RESULTS

Turbine generator net turbine heat rate[1] (NTHR) exceeds the warranty value by approximately 0.1 percent (12 Btu/kWh) at the warranty point. Refer to Figure 9.2-1, Specified Heat Rate Warranty Curve, for comparison. Overall test uncertainty is approximately 0.75 percent. Per the ANSI/ASME PTC-6 Steam Turbine Performance Test Code, verification of NTHR also verifies that warranteed electrical load has been achieved. The mass flow warranty value was verified from valve wide open test results.

Corrected test values obtained during the Initial Performance Test at 100 percent rated power (3411 MW_{TH}) with steam generator blowdown isolated and auxiliary steam load supplied by the auxiliary boiler were:

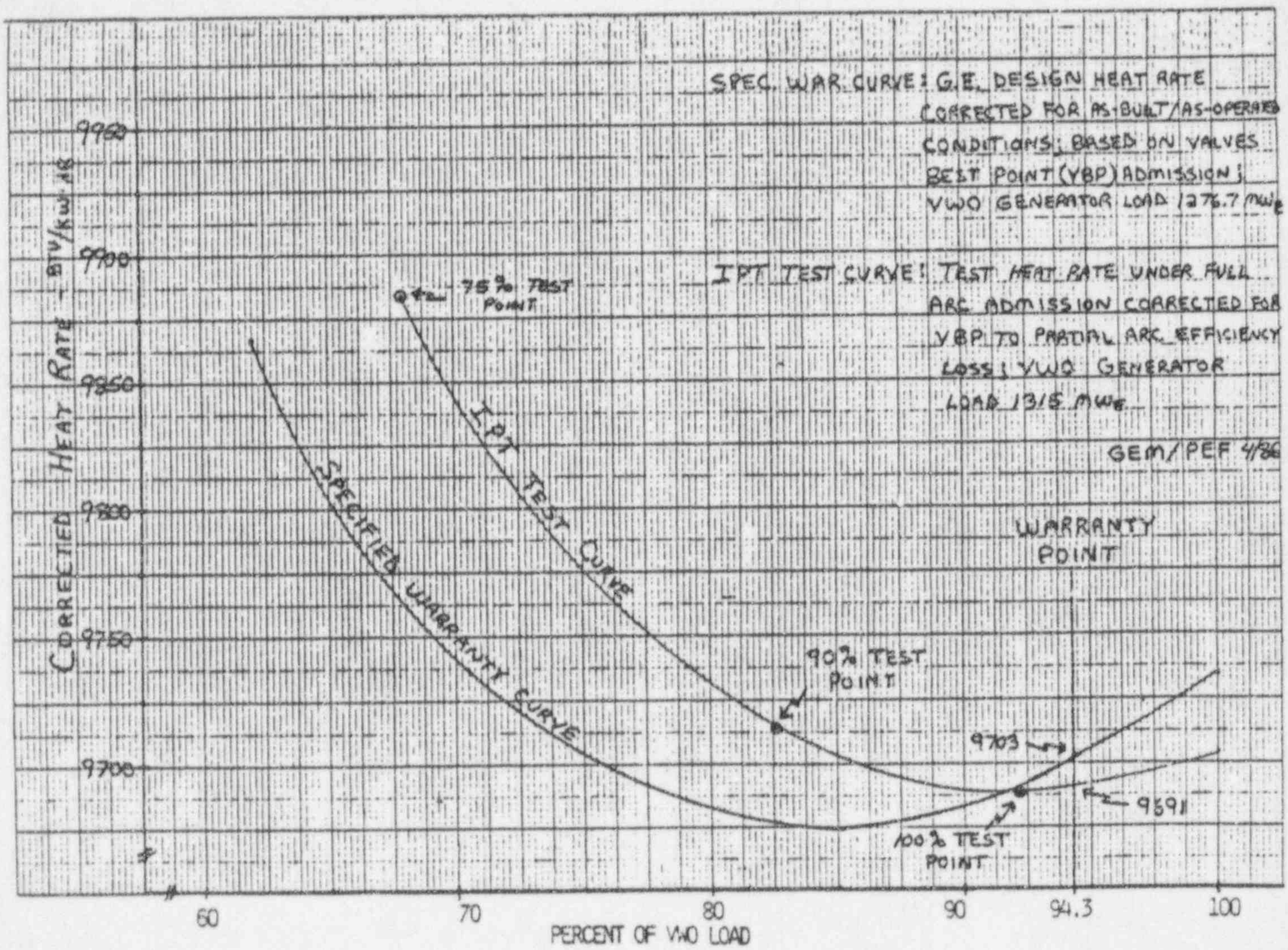
Gross Generator Load	1203.9 MW_E
Station Service Load	47.7 MW_E
Net Turbine Heat Rate	9707 Btu/kWh
Valve Wide Open Volumetric Flow[2]	1982 Ft^3/S

Note: During normal plant operation, gross generator load will be lower and NTHR higher by approximately 0.5 percent since steam generator blowdown will be in operation with auxiliary steam load supplied from the main steam system.

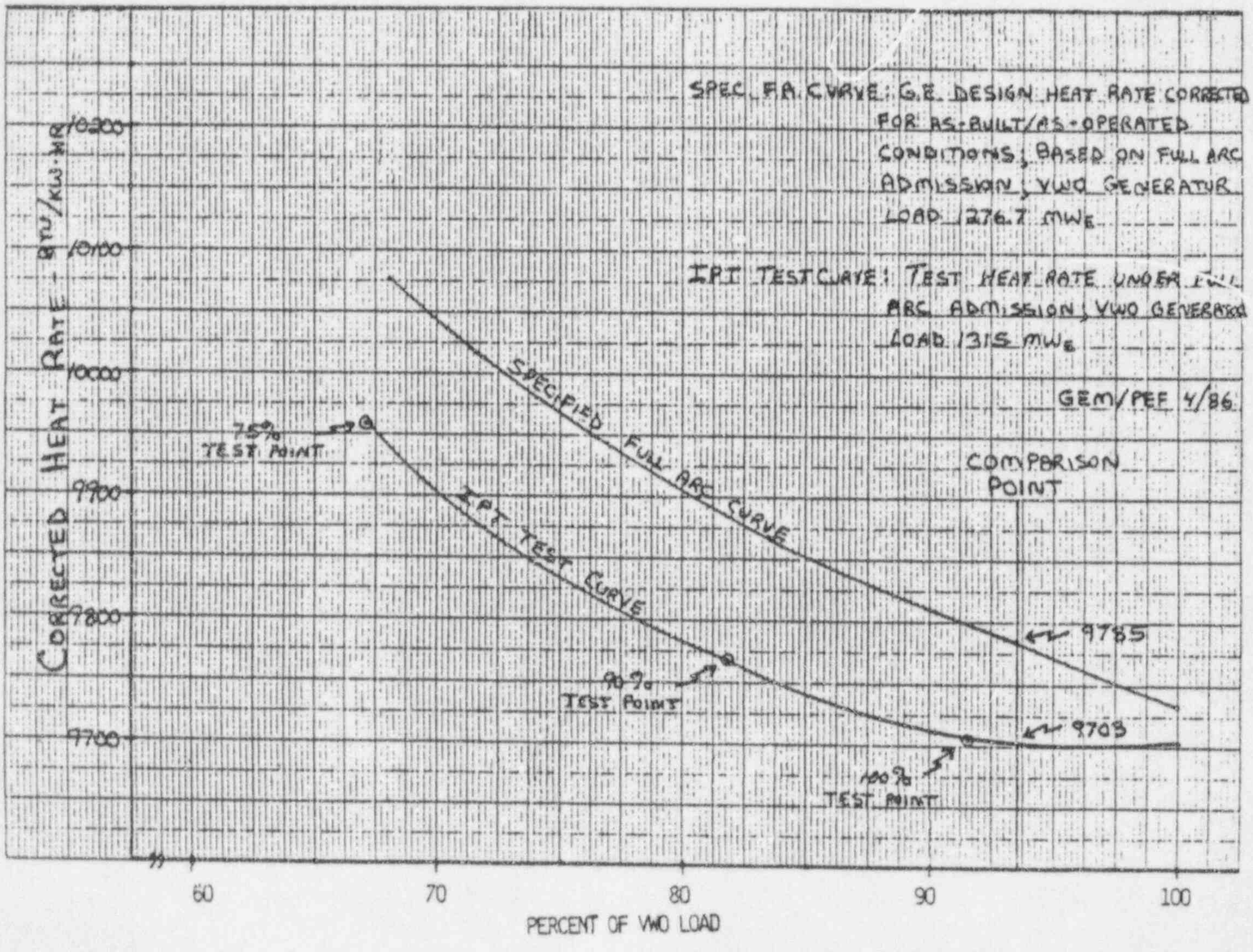
[1]Net Turbine Heat Rate = $\frac{\text{Steam Generator Power}}{\text{Gross Generator Load}}$

[2]Indicates turbine is passing approximately 4.0 percent excess flow

The test NTHR and gross generator load exceeded predicted full arc admission target values by 0.75 to 1.0 percent. The full arc target values for NTHR and gross generator load at 100 percent rated power are 9785 Btu/kWh and 1194.3 MW_E, respectively. Refer to Figure 9.1-2, Full Arc Specified Heat Rate Curve.



Millstone Nuclear Power Station Unit No. 3	SPECIFIED HEAT RATE WARRANTY CURVE SECONDARY PLANT PERFORMANCE TESTING	Figure 9.2-1



Millstone Nuclear Power Station Unit No. 3	FULL ARC SPECIFIED HEAT RATE CURVE SECONDARY PLANT PERFORMANCE TESTING	Figure 9.2-2

APPENDIX A

FINAL SAFETY ANALYSIS REPORT TESTING DEVIATIONS

Introduction

FSAR Chapter 14 details testing and operational commitments from Initial Inspection and Component Testing through Warranty Run. During the Startup Program, certain aspects of test procedures and performance deviated from FSAR Chapter 14 as stated. These deviations were documented and approved by the use of Quality Assurance forms and procedures relating to FSAR Changes. As such, the changes were reviewed by the site Plant Operations Review Committee for unreviewed safety question significance.

Preoperational/Acceptance Test Deviations

1. Boron Thermal Regeneration System (BTRS) Testing - This FSAR Change allowed BTRS testing to be performed after completion of HFT as plant conditions permit due to lack of system availability as well as the fact that BTRS is not covered by Technical Specifications nor is it a safety-related system.
2. Spent Fuel Pool Cooling (SFC) System Testing - This FSAR Change allowed SFC testing to be performed after completion of HFT as plant conditions permit due to lack of system availability as well as the fact that SFC is not covered by Technical Specifications nor were were the untested portions of the system safety-related.
3. Control Rod Drive Mechanism (CRDM) Testing - This FSAR Change deleted CRDM testing at hot standby conditions because equivalent/more limiting testing was performed during cold shutdown conditions.
4. Rod Drop Testing - This FSAR Change deleted hot, no-flow rod drop time testing because equivalent/more limiting testing was performed during cold full-flow conditions.
5. Rod Drop Testing - This FSAR Change administratively took exception to the RG 1.68 requirement to perform hot no-flow rod drop testing deleted in (4) above.

6. Main Feedwater Testing - This FSAR Change allowed certain transient Feedwater system testing to be performed post-HFT during Power Ascension when plant conditions were better able to support testing.

Startup Test Deviations

1. Natural Circulation Testing - This FSAR Change eliminated some specific natural circulation testing requirements which were incorrectly identified for performance during Post-Core Hot Functional Test.
2. Shutdown From Outside The Control Room Test - This FSAR Change allowed credit to be taken for the required Cold Shutdown demonstration as part of the Shutdown from Outside the Control Room Test because of equivalent testing performed previously.
3. Loss of Power Test - This FSAR Change deleted a prerequisite for the Station Blackout test which required all plant loads to be supplied from the turbine-generator because it allowed greater test flexibility and the fact that equivalent turbine-generator testing would be performed during the 100 percent Power Trip.
4. Pseudo Ejected Rod Test - This FSAR Change deleted the Pseudo Ejected Rod Test at 30 percent power because of the excessive flux tilt it would have caused, credit taken for like testing at other similar design plants, and previous similar testing performed at zero percent power.
5. Pseudo Ejected Rod Test - This FSAR Change administratively took exception to the RG 1.68 requirement to perform a Pseudo Ejected Rod Test at greater than 10 percent power which was deleted in (4) above.
6. 50 Percent Reactor Trip - This FSAR Change deleted the requirement to perform a 50 percent Power Reactor Trip and substituted a 10 percent Load Swing for the following reasons:
   a. There was no regulatory requirement to perform a 50 percent trip.
   b. The NSSS supplier deleted the requirement to perform a rod drop/negative rate trip test at 50 percent power.

c. The NRC requested performance of a 10 percent Load Swing at 50 percent power.

d. The plant challenge involved was significantly less.

APPENDIX B
STARTUP TEST PROCEDURE LISTING

PROCEDURE NUMBER	TITLE	STARTUP REPORT SECTION
3-INT-4000	Initial Fuel Load	4.0
Appendix 4003[1]	Core Load Instruments and Neutron Source Requirements	4.0
Appendix 4004	Inverse Count Rate Ration Monitoring	4.0
Appendix 4005	Initial Core Loading	4.0
Appendix 4006	Core Map	4.0
3-INT-5000	Postcore Hot Functional Test	5.0
Appendix 5001	Shutdown Margin	5.1
Appendix 5002	TC/RTD Testing (Incore TCs-RCS RTDs)	5.2
Appendix 5004	Rod Control Slave Cycler/CRDM Timing	5.3
Appendix 5006	RCS Leak Detection	5.4
Appendix 5007	Pressurizer Heaters and Spray	5.5
Appendix 5008	Rod Drop Testing	5.6
Appendix 5009	RCS Flow Measurement	5.7
Appendix 5010	RTD Bypass Loop Verification	5.8
Appendix 5011	Movable Incore Detectors	5.9
Appendix 5015	Digital Rod Position Indication	5.10
Appendix 5016	Loose Parts Monitoring	5.11
Appendix 5017	RCS Flow Coastdown	5.12
Appendix 5018	Rod Control	5.13
Appendix 5031	Chemical and Volume Control System	5.14
Appendix 5033	RCS Loop Stop Valve/Pump Interlocks	5.15
3-INT-6000	Initial Criticality	6.0
Appendix 6001	Inverse Count Rate	6.0
3-INT-7000	Low Power Physics Testing	7.0
Appendix 7001	HZP Testing Range Determination	7.1
Appendix 7002	Reactivity Computer Checkout	7.2
Appendix 7003	Boron Endpoint	7.3
Appendix 7004	Isothermal Temperature Coefficient	7.4
Appendix 7005	RCCA or Bank Worth Measurement	7.5
Appendix 7006	Natural Circulation (Low Power)	7.8

(1) Some appendices were deleted prior to performance and remaining appendices were not renumbered. Therefore, some numbers were not listed.

PROCEDURE NUMBER	TITLE	STARTUP REPORT SECTION
3-INT-8000	Power Ascension Testing	8.0
Appendix 8001	Calorimetric	8.5.1
Appendix 8002	Operational Alignment of Nuclear Instrumentation	8.2.1
Appendix 8003	Calibration of Steam and Feedwater Flow	8.2.3
Appendix 8004	Operational Alignment of Process Temperature Instrumentation	8.2.2
Appendix 8005	Reactor and Turbine Control	8.3.1
Appendix 8006	Secondary Plant Performance	8.5.2
Appendix 8007	Radiation Survey and Process Radiation	8.5.3
Appendix 8008	Ventilation System Operability	8.5.4
Appendix 8009	Chemistry and Radio Chemistry	8.5.5
Appendix 8010	Neutron Shield Tank Cooling	8.5.6
Appendix 8011	Containment Penetration Temperature Monitoring	8.5.7
Appendix 8013	Steam Dump Control	8.3.2
Appendix 8015	RCS Flow Measurement	8.1.1
Appendix 8016	Turbine Overspeed	8.4.1
Appendix 8017	Automatic Reactor Control	8.3.3
Appendix 8018	Automatic Steam Generator Level Control	8.3.4
Appendix 8019	Turbine Plant Component Cooling System Balancing	8.5.8
Appendix 8020	Power Coefficient	8.1.2
Appendix 8022	10 Percent Load Swing	8.4.2
Appendix 8023	Reactor Trip and Shutdown From Outside the Control Building	8.4.3
Appendix 8026	Large Load Reduction	8.4.4
Appendix 8028	Axial Flux Difference Instrumentation Calibration	8.2.4
Appendix 8029	Pipe Fluid Transient Vibration Testing	8.5.9
Appendix 8030	Loss of Power (20 Percent)	8.4.5
Appendix 8031	Reactor Coolant System Boron Measurement	8.1.3
Appendix 8032	Generator Trip (100 Percent)	8.4.6

PROCEDURE NUMBER	TITLE	STARTUP REPORT SECTION
Appendix 8034	Thermal Expansion and Restraint	8.5.10
Appendix 8035	Loose Parts Monitoring	8.5.11
Appendix 8037	Main Steam Line Isolation Valve Closure	8.3.5
3-INT-9000	Warranty Run	9.0
Appendix 9001	Calorimetric	9.1
Appendix 9002	Secondary Plant Performance	9.2

APPENDIX C

PREOPERATIONAL TESTS COMPLETED DURING THE STARTUP TEST PROGRAM

The following preoperational tests were completed during the startup test program. The individual tests were completed consistent with Technical Specification system operability requirements.

Test Number	Title	Date Completed
3307AP001	Low Pressure Safety Injection	12-07-85
3308-P002	High Pressure Safety Injection	12-06-85
3309-P001	Quench Spray	12-30-85
3311CP	Post Accident Sampling	01-29-86
3312CP	Containment Atmospheric Monitoring	01-12-85
3313DP	Containment Filtration	03-05-86
3313FP (Rev 1)	Containment Vacuum	12-31-85
3314BP	Fuel and Waste Disposal Building HVAC	03-03-86
3314DP	ESF Building HVAC	12-06-85
3314FP	Control Building HVAC	12-19-85
3314IP	Supplemental Leak Collection and Release	12-31-85
3315BA (Rev 1)	Main Steam Valve Building HVAC (Retest)	01-29-86
3317-A	Moisture Separator Reheater	02-03-86
3319CP001	Condensate Polishing	03-24-86
3320-P	Feedwater Heater Drains and Vents	01-11-86
3322-P	Auxiliary Feedwater	12-16-85
3324DA	Stator Cooling	01-30-86
3325DA	Condenser Tube Cleaning	04-05-86
3330AP	Reactor Plant Component Cooling Water	01-03-86

Test Number	Title	Date Completed
3330CP	Reactor Plant Chilled Water	11-26-85
3331DA	Hot Water Heating/Preheating	11-25-85
3335BP	Radioactive Liquid Waste	02-23-86
3335CP	Boron Recovery	01-27-86
3337-P	Radioactive Gaseous Waste	01-27-86
3341BP (Rev 1)	Fire Protection-Halon (Retest)	11-25-85
3341CP	Fire Protection-CO_2	12-30-85
3345CP006	Battery Duty Cycle Testing	01-10-86
3404-P	Digital Radiation Monitoring	11-18-85
3410[illegible]	Reactor Vessel Level	12-31-85
3720BP (Rev 1)	Station Emergency Lighting (Retest)	12-20-85
3999-P	Pipe/Pipe Support Steady-State Vibration	02-04-86
3-INT-2001 Appendix P5 (Rev 1)	Secondary Plant Performance	12-21-85
3-INT-2001 Appendix R10	Incore (Power Distribution)	01-12-86
3-INT-2001 Appendix R11	Estimated Critical Position	04-21-86
3-INT-2001 Appendix R12	Shutdown Margin	01-06-86
3-INT-2007	ISI Valve Stroke Time Testing	01-09-86

The following preoperational tests were completed after the startup test program was completed.

Test Number	Title	Date Completed
3721-A001	Electrical Distribution - Security	05-22-86
3721-A002	Integrated System Test - Security	05-30-86
3-INT-2001	Computer Programs Test	05-23-86
3-INT-2008	Efficiency Testing of Air Filtration Units	07-18-86

The following preoperational tests are yet to be completed. Provided is a summary of test status and plan for test completion.

Test Number	Title
3304DP	Boron Thermal Regeneration

The preoperational test has not yet been begun due to equipment problems. The system is currently isolated and not required for plant operation. Testing will be completed in accordance with plant requirements but no later than startup following the first refueling outage. As this test is referenced in Chapter 14 of the Millstone 3 FSAR, a proposed revision to the FSAR has been submitted to permit performance of the test as dictated by plant requirements.

3305-P Spent Fuel Pool Cooling and Purification

The safety-related portion of the system was satisfactorily tested as a prerequisite to receiving nuclear fuel. The remaining (non-safety) portions will be tested once the spent fuel pool is filled to support refueling and subsequent fuel storage activities. It is therefore anticipated the remaining testing will be completed prior to the first refueling outage. As this test is referenced in Chapter 14 of the Millstone 3 FSAR, a proposed revision to the FSAR has been submitted to permit completion of the test as dictated by plant requirements.

Test Number	Title
3311EA	EEQ Area Temperature Monitoring System

Physical testing is complete but the test procedure is being kept open while a revision to various EEQ area temperature alarm setpoints are made. The procedure will then be utilized to cover the system retest with the revised setpoints.

3319CP002 Condensate Liquid Waste

The test is partially complete. Currently the system is not required to support plant operations. Plans are to complete the test in a manner consistent with plant operations requirements.

3328-A Chlorine

During the startup of Millstone 3, the medium used for biological growth control in the service water system was switched from chlorine gas injection to sodium hypochlorite injection. The sodium hypochlorite system is presently in service and performing its intended function. The testing of the system will be completed consistent with plant requirements.

APPENDIX D

SUMMARY OF PROBLEMS ENCOUNTERED DURING THE LOSS OF POWER TEST (3-INT-8000, APPENDIX 8030)

	PROBLEM	COMMENTS/RESOLUTION
1.	CCP*P1B did not go from OFF to ON during Loss of Power (LOP).	Test logic was incorrect in that P1B was in pull-to-lock at the time of LOP. P1C was aligned to train B and was observed to function properly. A test change was issued to correct this problem with the test procedure.
2.	CHS*P3B did not go from OFF to ON during LOP.	Test logic was incorrect. P3A was running initially, tripped on LOP and subsequently automatically restarted. A test change was issued to correct this problem with the test procedure.
3.	FWA*AOV26 did not go from OPEN to CLOSE during LOP.	Plant deficiency UNS 7572 was issued to document this problem. Plant maintenance personnel investigated and found a limit switch problem. Limit switch was adjusted and retested satisfactorily.
4.	HVK*CHL1B did not go from OFF to ON during LOP.	Plant deficiency UNS 7573 was issued to document this problem. Contrary to the problem description, review of the Sequence of Events (SOE) digital printout indicate:

PROBLEM	COMMENTS/RESOLUTION
HVK*CHL1B did not go from OFF to ON during LOP. (4. continued)	1. HVK*CHL1A which was running at the time of LOP, tripped on LOP. 2. Approximately 80 seconds after restoration of power, HVK*CHL1B automatically started. This is as per design. 3. Approximately 148 seconds after starting HVK*CHL1B tripped. The postulated cause is low Freon level. 4. Approximately 15 minutes after tripping on LOP, HVK*CHL1A, responding to operator action, started. An automatic timer feature prevents the restart of a chiller for 15 minutes after a chiller is stopped. Therefore, with the exception of the B chiller tripping, both chillers operated per design. Regarding the B chiller trip, based on past operating history of these chillers, it is postulated the B chiller tripped because of low Freon level. Plant Maintenance personnel

	PROBLEM	COMMENTS/RESOLUTION
		recharged the Freon in the B chiller. The unit has performed satisfactorily since then.
5.	HVR*FN6B did not go from OFF to ON during LOP.	Per a change to the system operating procedure (OP 3314A), the variable inlet vanes (VIV) on the fan must be placed in MANUAL at a 20% open position for the fan to start automatically. During LOP, VIV were in AUTO. This was an improper system alignment. Plant Operations personnel were advised of this and action was taken to ensure proper system alignment in the future.
6.	IAS-C2B did not go from OFF to ON during LOP.	Plant deficiency UNS 7574 was issued to document this problem. Plant Electrical Maintenance personnel investigated and determined the problem was caused by a fault in an overload heater circuit which caused an inoperable control circuit. After repair, retest under a simulated LOP condition was satisfactorily.

	PROBLEM	COMMENTS/RESOLUTION
7.	SWP*MOV130B did not go from CLOSED to OPEN during LOP.	Error in test procedure. HVR*ACU1B was in pull-to-lock so no open signal was sent to valve. A test change was issued to correct this problem with the test procedure.
8.	SWP*P1A was not running before or after LOP. This is contrary to the test procedure.	Error in test procedure. The procedure assumed the alternate pump on each SWP train would be running. A test change was issued to correct this administrative problem.
9.	SWP*P1C was running before and after LOP. This is contrary to the test procedure.	See discussion under number 8.