CHAPTER 13
TESTS AND OPERATIONS

13.0 INTRODUCTION

[Historical Information] The testing and startup operation of the plant systems prior to full power operation of the unit included tests made prior to the initial reactor fuel loading, precritical tests, zero power tests, and power level escalation, plus tests made as part of the zero power and power ascension program inherent with each core loading cycle and periodic test requirements of the Technical Specifications.

The purpose of the program has been to test and operate the reactor and its various systems (1) to make certain that the equipment has been installed and will operate in accordance with the design requirements, (2) to provide procedures for safe initial fuel loading or fuel reloading and to determine zero power values of core parameters significant to the design and operation, and (3) to bring the unit to its rated capacity in a safe and orderly fashion.

Prior to initial full-power operation of Indian Point Unit 2, the plant underwent a thorough, systematic testing program that successively demonstrated the capability and safety of the plant to proceed to each following stage of testing until full power was achieved and maintained. WEDCO, a wholly owned subsidiary of Westinghouse, had the overall responsibility for engineering, construction management, and initial startup testing. The initial startup tests were subdivided into several stages, each to be completed before the next stage was undertaken. Following the startup and testing program, periodic system and plant performance tests are performed as described in the Technical Specifications.

Detailed procedures stating the test purpose, conditions, precautions, and limitations are prepared for each test. The procedures include a delineation of administrative procedures and test responsibility, equipment clearance procedures, and an overall sequence of startup operations. The procedures specify the sequence of tests and measurements to be conducted and conditions under which each is to be conducted to ensure both safety of operation and the relevancy and consistency of the results obtained. If significant deviations from design predictions should exist, unacceptable behavior be revealed, or apparent anomalies develop, testing is suspended and the situation reviewed by the licensee and technical advisors as appropriate to determine whether a question of safety is involved and what corrective action is to be taken prior to resumption of testing. The ultimate responsibility for these determinations rests with the licensee.

The test objectives incorporate testing of redundant equipment where it is involved. Abnormal plant conditions may be simulated during testing when such conditions do not endanger personnel or equipment, or contaminate clean systems. Where predicted emergency or abnormal conditions are involved in the testing program, the detailed operation is provided in the test procedure.

Acceptance criterion for all components and systems is that the test results are acceptable when the test objectives are met within the design specification limits and within the applicable Technical Specifications.

The test program described in the following sections is based upon the reference plant design and experience gained during startup of other units. The detailed procedures include expected values and acceptance criteria that demonstrate the degree to which the facility does meet design criteria.

13.1 TESTS PRIOR TO INITIAL REACTOR FUEL LOADING [Historical Information]

The first stage of the initial tests was a comprehensive testing program, which ensured that equipment and systems performed in accordance with design criteria prior to fuel loading. As the installation of individual components and systems was completed, they were tested and evaluated according to predetermined and approved written testing techniques, procedures, or checkoff lists. Field and engineering analyses of test results were made to verify that systems and components were performing satisfactorily and to recommend corrective action, if necessary.

The program included tests, adjustments, calibrations, and system operations necessary to ensure that initial fuel loading and subsequent power operation could be safely undertaken. In general, the types of tests were classified as installation, flushing, hydrostatic, hot functional, and preoperational tests. These tests were aimed at verifying that the system or equipment was capable of performing the function for which it was designed.

Where practical, preoperational tests involved actual operation of the system and equipment under design or simulated design conditions. In addition, the reactor protection and safeguards instrumentation systems were performance tested prior to initial core loading.

The reactor coolant system vibration testing program overlapped the plant testing program. Data for this particular program were taken during cold hydro and hot functional testing prior to fuel loading and also during the low-power physics tests that followed initial fuel loading (refer to Section 13.5).

The list below is the sequence of major startup tests and operations performed to place all equipment in the specified system in service for the initial reactor fueling. Table 13.1-1 describes the objectives of the tests. Con Edison, in cooperation with Westinghouse/WEDCO, prepared detailed test procedures prior to the scheduled initial testing of systems and determination of reactor physics parameters. The tests conducted on the engineered safety systems are included under the safety injection system, the containment spray system, and the containment air recirculation cooling and filtration system:

1. Switchgear system.
2. Voice communication systems.
3. Service water system.
4. Fire protection system.
5. Instrument and service air systems.
6. Nitrogen storage system.
7. Reactor coolant system cleaning.
8. Reactor containment air recirculation and filtration system.
9. Feedwater and condensate circulation systems.
10. Auxiliary coolant system.
11. Chemical feed system.
12. Chemical and volume control system.
13. Containment spray system.
14. Safety injection system.
15. Fuel handling system.

16. Containment isolation and isolation valve seal-water systems.
17. Containment penetration and weld channel pressurization system.
18. Reactor containment high-pressure test.
19. Cold hydrostatic tests.
20. Radiation monitoring system.
21. Nuclear instrumentation system.
22. Radioactive waste disposal system.
23. Sampling system.
24. Instrument calibration.
25. Hot functional tests.
    a. Reactor coolant system.
    b. Chemical and volume control system.
    c. Sampling system.
    d. Auxiliary coolant system.
    e. Safety injection system.
    f. Radioactive waste disposal system.
    g. Ventilation system.
26. Primary and secondary systems safety valves tests.
27. Turbine steam seal and blowdown systems.
28. Emergency diesel electric system.

TABLE 13.1-1
(Sheet 1 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading [Historical Information]

System or Test	Test Objective
1. Switchgear system (electrical tests)	To ensure continuity, circuit integrity, and the correct and reliable functioning of electrical apparatus. Electrical tests were performed on transformers, switchgear, turbine generator, motors, cables, control circuits, excitation switchgear, dc system, annunciator systems, lighting distribution switchboard, communication system, and miscellaneous equipment. Special attention was directed to the following tests: a. 480-V switchgear breaker interlock test. b. Station loss of voltage auto-transfer test. c. Critical power transfer test. d. Tests of protective devices. e. Equipment automatic start tests. f. Check exciter for proper voltage buildup.
2. Voice communication systems	To verify proper communication between all intraplant stations, for interconnection to commercial phone service, and to balance and adjust amplifiers and speakers.
3. Service water system	To verify, prior to critical operations, the design head capacity characteristics of the service water pumps; that the system would supply design flow rate through all heat exchangers; and would meet the specified requirements when operated in the safeguards mode.
4. Fire protection system	To verify proper operation of the system by ensuring that the design specifications would be met for the fire service booster pump and fire service pumps, checking that automatic start functions operate as designed, and that level and pressure controls meet specifications.
5. Instrument and service air systems	To verify the operation of all compressors to design specifications, the manual and automatic operation of controls at design setpoints, design air-dryer cycle time and moisture content of discharge air, and proper air pressure to each instrument served by the system.
6. Nitrogen storage system	To verify system integrity, valve operability, regulating and reducing station performance, and the ability to supply nitrogen to interconnecting systems as required.

TABLE 13.1-1
(Sheet 2 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
7. Reactor coolant system cleaning	To flush and clean the reactor coolant and related primary systems to obtain the degree of cleanliness required for the intended service. Provisions to maintain cleanliness, integrity, and protection from contamination sources were made after system cleaning and acceptance. The system, component, or section of a system was considered clean when the flush cloth showed no grindings, filings or insoluble particulate matter larger than 40 μm (lower limit of naked eye visibility). After systems were flushed clean of particulate matter within the limit specified, the cleanliness integrity of the system was maintained filled with water, which met the system cold chemistry requirements. After fill and pressurization and prior to hot operation, cold chemistry requirements were maintained. Oxygen was analyzed and brought into specification prior to exceeding 200°F.
8. Reactor containment air recirculation and filtration system	To verify, prior to critical operation, the fan capacities, and the remote and automatic operation of system louvers and valves in accordance with the design specifications.
9. Feedwater and condensate circulation systems	To verify proper operation of feedwater and circulating water pumps according to specifications, valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing was performed when the steam supply became available.

TABLE 13.1-1
(Sheet 3 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
10. Auxiliary coolant system	To verify component cooling flow to all components and to verify proper operation of instrumentation, controllers, and alarms. Specifically, each of the three loops, that is, the component cooling loop, the residual heat removal loop, and the spent fuel pit cooling loop, were tested to ensure that: a. All manual and remotely operated valves were operable manually and/or remotely. b. All pumps performed according to manufacturer's specifications. c. All temperature, flow, level, and pressure controllers functioned to control at the required setpoint when supplied with appropriate signals. d. All temperature, flow, level, and pressure alarms functioned at the required locations when the alarm setpoint was reached and cleared when the reset point was reached. e. Design flow rates were established through heat exchangers.
11. Chemical feed system	To verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing was performed when the steam supply became available.

TABLE 13.1-1
(Sheet 4 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
12. Chemical and volume control system	To verify, prior to critical operation, that the chemical and volume control system would function as specified in the system description and appropriate technical manuals. More specifically that: a. All manual and remotely operated valves were operable manually and/or remotely. b. All pumps performed to manufacturer's specifications. c. All temperature, flow, level, and pressure controllers functioned to control at the required setpoint when supplied with appropriate signals. d. All temperature, flow, level, and pressure alarms functioned at the required locations when the alarm setpoint was reached and cleared when the reset point was reached. e. The reactor makeup control accomplished blending, dilution, and boration as designed. f. The design seal-water flow rates were attainable at each reactor coolant pump. g. The boric acid evaporator package functioned as specified in the manufacturer's technical manual.
13. Containment spray system	To verify performance of the containment spray pumps.

TABLE 13.1-1
(Sheet 5 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
14. Safety injection system	To verify, prior to critical operation, response to control signals and sequencing of the pumps, valves, and controllers of this system as specified in the system description and the manufacturer's technical manuals, and check the time required to actuate the system after a safety injection signal is received. More specifically that: a. All manual and remotely operated valves were operable manually and/or remotely. b. All pumps performed their design functions satisfactorily. c. For each pair of valves to redundant flow paths, disabling one of the valves would not impair remote operation of the other. d. The proper sequencing of valves and pumps occurred on initiation of a safety injection signal. e. The fail position on loss of power for each remotely operated valve was as specified. f. Valves requiring coincidence signals of safety injection and high containment pressure operated when supplied with these signals. g. All level and pressure units were set at the specified points and provided alarms at the required location(s), and reset at the specified point. h. The time required to actuate the system was within the design specifications.

TABLE 13.1-1
(Sheet 6 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
15. Fuel handling system	To show that the system design would be capable of providing a safe and effective means of transporting and handling fuel from the time it reaches the plant until it leaves the plant. In particular, the tests were designed to verify that: a. The major structures required for refueling such as the reactor cavity, refueling canal, spent fuel storage pool, and decontamination facilities were in accordance with the design specifications. b. The major equipment required for refueling such as the manipulator crane, spent fuel pit bridge, and fuel transfer system would operate in accordance with the design specifications. c. All auxiliary equipment and instrumentation would function properly.
16. Containment isolation and isolation valve seal water systems	To verify the capability for reliable operation and to demonstrate the manual and automatic operation of the system. To demonstrate the operation and proper sequence of isolation valve closure and seal-water addition. To demonstrate function of isolation valve seal-water system independent of other systems. To demonstrate the operation and system response time induced by an isolation signal. Manual valves were manipulated to ensure proper operation of the seal-gas injection portion of the system.
17. Containment penetration and weld channel pressurization system	To verify the air system and nitrogen backup system integrity, operate valves, check flow-meters and pressure gauges as required to ensure that pressure differentials would meet design specifications.
18. Reactor containment high-pressure test	To verify, prior to critical operation, the structural integrity and leaktightness of the containment.
19. Cold hydrostatic tests	To verify the integrity and leaktightness of the reactor coolant system and related primary systems with the performance of a hydrostatic test at the specified test pressure with no visible leakage or distortion.

TABLE 13.1-1
(Sheet 7 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
20. Radiation monitoring system	To verify the calibration, operability, and alarm setpoints of all radiation level monitors, air particulate monitors, gas monitors, and liquid monitors that are included in the operational radiation monitoring system and the area radiation monitoring system.
21. Nuclear instrumentation system	To ensure that the instrumentation system is capable of monitoring the reactor leakage neutron flux from source range through 120-percent of full power and that protective functions are operating properly. In particular the tests were designed to verify that: a. All system equipment, cabling, and inter-connections were properly installed. b. The source range detector and associated instrumentation would respond to neutron level changes and that the source range protection (high flux level reactor trip) as well as alarm features and audible count rate would operate properly. c. The intermediate range instrumentation reactor protection and control features (high-level reactor trip and high-level rod stop signals) would operate properly and that permissive signals for blocking source range trip and source range high-voltage-off would operate properly. d. The power range instrumentation would operate properly and that the protective features such as the overpower trips and permissive and dropped-rod functions would operate with the required redundancy and separation through the associated logic matrices, and nuclear power signals to other systems were available and operating properly. e. All auxiliary equipment such as the comparator and startup rate channel, recorders, and indicators were operating as specified. f. All instruments were properly calibrated and all setpoints and alarms properly set.

TABLE 13.1-1
(Sheet 8 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
22. Radioactive waste disposal system	To verify satisfactory flow characteristics through the equipment, to demonstrate satisfactory performance of pumps and instruments, to check for leaktightness of piping and equipment, and to verify proper operation of alarms, instrumentation, and controls. More specifically that: a. All piping and components were properly installed as per design specifications. b. All manual and automatic valves were operable. c. All instrument controllers were operating to control processes at required values. d. All process alarms were operable at required locations. e. All pumps performed to manufacturer's specifications. f. All pump indications and controls were operable at designated stations. g. The waste gas compressors packages operated as specified in manufacturer's technical manual. h. The gas analyzer operated as specified in manufacturer's technical manual. i. The waste evaporator operated as specified in manufacturer's technical manual. j. The hydrogen and nitrogen supply packages were sufficient for all modes of operation.
23. Sampling system	To verify that a specified quantity of representative fluid could be obtained safely and at design conditions from each sampling point. In particular the tests were designed to verify that: a. All system piping and components were properly installed. b. All remotely and manually operated valving operated in accordance with the design specifications. c. All sample containers and quick-disconnect couplings functioned properly and as specified.

TABLE 13.1-1
(Sheet 9 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
24. Instrument calibration	Instrumentation and control devices were checked to ensure their accuracy. Primary sensing elements, transducers, transmitters, receivers, recorders and indicators were thoroughly inspected and adjusted for accuracy of their setpoint characteristics. Interconnecting piping and wiring were checked for continuity and functional requirements. Each device was tested in accordance with established test procedures. Limit switches used for initiating indicating lights, alarms, and interlock functions were checked under actual or simulated operating conditions. Control devices were exercised to ensure proper operation with the required accuracy and response characteristics. Setpoints for devices were checked and adjusted to their specified values. Each individual circuit of the reactor and turbine protection systems was tested to verify that appropriate signals initiate reactor and turbine trips. As a signal level corresponding to the particular condition was reached, trip or cutback functions would annunciate as provided in the particular channel under test.

TABLE 13.1-1
(Sheet 10 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
25. Hot functional tests	The reactor coolant system was tested to check heatup (using pump heat) and cooldown procedures; to demonstrate satisfactory performance of components prior to installation of the core; to verify proper operation of instrumentation, controllers, and alarms; and to provide operating conditions for checkout of auxiliary systems. The chemical and volume control system was tested to determine that water could be charged at rated flow against normal reactor coolant system pressure, to check letdown flow against design rate for each pressure reduction station, to determine the response of the system to changes in pressurizer level, to check procedures and components used in boric acid batching and transfer operations, to check operation of the reactor makeup control, to check operation of the excess letdown and seal-water flowpath, and to verify proper operation of instrumentation, controllers, and alarms. The sampling system was tested to determine that a specified quantity of representative fluid could be obtained safely and at design conditions from each sampling point. The auxiliary coolant system was tested to evaluate its ability to remove heat from reactor coolant, to verify component cooling flow to all components, and to verify proper operation of instrumentation, controllers, and alarms. The safety injection system was tested to check the time required to actuate the system after a safety injection signal is received, to check that pumps and motor-operated valves were properly sequenced, and to verify proper operation of instrumentation, controllers, and alarms. The radioactive waste disposal system was tested to verify satisfactory flow characteristics through the equipment, to demonstrate satisfactory performance of pumps and instruments, to check for leaktightness of piping and equipment, and to verify proper operation of alarms. The ventilation system was tested to adjust proper flow characteristics of ducts and equipment; to demonstrate satisfactory performance of fans, filters, and coolers; and to verify proper operation of instruments and alarms.

TABLE 13.1-1
(Sheet 11 of 11)
Objectives of Tests Prior to Initial Reactor Fuel Loading

System or Test	Test Objective
26. Primary and secondary systems safety valves tests	To test pressurizer and boiler safety and relief valves to ensure that each valve was operable.
27. Turbine steam seal and blowdown systems	To verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing was performed when a steam supply became available.
28. Emergency diesel electric system	To demonstrate that the system was capable of providing power for operation of vital equipment under power failure conditions. In particular the tests were designed to verify that: a. All system components were properly installed. b. The emergency diesels function according to the design specification under emergency conditions. c. The emergency units are capable of supplying the required power to vital equipment under emergency conditions. d. All redundant features of the system function according to the design specifications.

13.2 FINAL PLANT PREPARATION [Historical Information]

13.2.1 Core Loading

[Historical Information] Fuel loading did not begin until the prerequisite system tests and operations as defined in the detailed core loading procedures were satisfactorily completed and the facility operating license was obtained. Upon completion of fuel loading, the reactor upper internals and pressure vessel head were installed and additional mechanical and electrical tests were performed. The purpose of these activities was to prepare the system for nuclear operation and to establish that all design requirements necessary for operation had been achieved.

The overall responsibility and direction for initial core loading was exercised by the general superintendent. During the initial core-loading operation, the WEDCO refueling manager was in charge of the Westinghouse activities. The process of initial core loading was, in general, directed from the operating floor of the containment structure. Standard procedures for the control of personnel and the maintenance of containment security were established prior to fuel loading. The core configuration was specified as part of the core design studies conducted well in advance of station startup and as such was not subject to change at startup. The core was assembled in the reactor vessel, submerged in water containing sufficient quantities of boric acid to maintain the fully loaded core substantially subcritical. Core-loading procedures specify alignment of fluid systems to prevent inadvertent dilution of the boron in the reactor coolant, restrict the movement of fuel to preclude the possibility of mechanical damage, prescribe the conditions under which loading may proceed, identify chains of responsibility and authority, and provide for continuous and complete fuel and core component accountability.

The core-loading procedure documents included a detailed tabular check sheet that prescribed and verified the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Multiple checks were made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components. The results of each loading step were evaluated by the Con Edison licensed senior reactor operator and the WEDCO refueling manager before the next prescribed step was started.

Core moderator chemistry conditions (particularly boron concentration) were prescribed in the core-loading procedure document and were verified by chemical analysis of moderator samples every 8 hr during core-loading operations.

The reactor coolant system was isolated and applicable tagging and administrative procedures used to prevent unauthorized change in the boron concentration. The boric acid tank was filled with concentrated boric acid solution and the residual heat removal system placed in service and available to provide moderator mixing and temperature control, if required. A detailed preloading checkoff list was followed to ensure that all systems, equipment and conditions affecting the loading operation were met. Periodically, the checkoff list was reviewed to ensure that systems and equipment continued to meet requirements of the core-loading operation.

The core-loading sequence followed a step-by-step procedure to ensure at each loading step that:

1. Fuel assemblies of the correct enrichments were installed in the proper locations.

2. Rod cluster control assemblies were inserted into the proper fuel assemblies prior to loading the assemblies into the core.

3. Neutron sources and neutron detectors were properly located in the core during fueling. Continuous radiation monitoring was provided at the core-loading stations during fuel-handling and core-loading operations.

Core-loading instrumentation consisted of two permanently installed plant source range (pulse-type) nuclear channels and two temporary incore source range channels plus a third temporary channel to be used as a spare. The permanent channels were monitored in the control room by licensed plant operators; the temporary channels were installed in the containment and were monitored by technical specialists of Westinghouse and by licensed senior reactor operators of Con Edison. At least one plant channel and one temporary channel were equipped with audible count range indicators. Both plant channels and both regular temporary channels displayed neutron count rate on count rate meters and strip chart recorders. Two artificial neutron sources, each rated at approximately 200 Ci of Po-210 alpha activity, were introduced into the core at appropriate specified points in the core-loading program to ensure a neutron population large enough for adequate monitoring of the core.

Fuel assemblies together with inserted control components (rod cluster control units or burnable poison inserts) were added to the core one at a time according to a previously established and approved sequence that had been developed to provide reliable core monitoring with minimum possibility of core mechanical damage. The core-loading procedure documents included a detailed tabular check sheet that prescribed and verified the successive movements of each fuel assembly and its specified inserts from the initial position in the storage racks to the final positions in the core.

An initial nucleus of eight fuel assemblies, the first of which included an activated neutron source, was determined to be the minimum source-fuel nucleus that would permit subsequent meaningful inverse count rate monitoring. This initial nucleus is known by calculation and previous experience to be markedly subcritical (k_{eff} = 0.90) under the required conditions of loading.

Subsequent fuel additions were made one assembly at a time with detailed inverse count rate ratio monitoring after each addition. The results of each loading step were evaluated by both Westinghouse technical specialists and licensed Con Edison operations personnel; concurrent approval to proceed had to be granted before the next prescribed step was started.

Criteria for safe loading required that loading operations stop immediately if:

1. The neutron count rates on all responding nuclear channels doubled during any single loading step.

2. The neutron count rate on any individual nuclear channel increased by a factor of 5 during any single loading step.

A containment evacuation alarm was coupled to the plant source range channels to provide automatic indication of high count rate during fuel addition.

In the event that an unacceptable increase in count rate was observed on any or all responding nuclear channels, special procedures involving fuel withdrawal from the core, detector relocation and charging of additional boric acid into the moderator could have been invoked by Westinghouse technical specialists with the approval of licensed operational personnel of Con Edison.

13.2.2 Precritical Tests [Historical Information]

Upon completion of core loading and installation of the reactor upper internals and the reactor vessel head, certain mechanical and electrical tests were performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, primary and secondary trip circuits, and the incore thermocouples were tested. Final operational tests were repeated on these electrical items.

Mechanical and electrical tests were performed on the rod cluster control unit drive mechanisms. Tests included a complete operational checkout of the mechanisms. Checks were made to ensure that the rod position indicator coil stacks were connected to their proper position indicators. Similar checks were made on the rod cluster control unit drive coils.

After filling and venting was completed, the final hydro tests were conducted.

Tests were performed on the reactor trip circuits to test manual trip operation. Actual rod cluster control unit drop times were measured for each rod cluster control at operating temperature, pressure, and flow. By use of dummy signals, the various plant abnormalities that require tripping were simulated and accurate trip delay times were measured for the control and protection system circuitry.

A complete electrical and mechanical check was made on the incore nuclear flux mapping system at the operating temperature and pressure.

The incore thermocouple tests checked circuit continuity and compared the thermocouple readings for their relative errors (offsets) in the isothermal condition.

13.3 INITIAL TESTS IN THE OPERATING REACTOR [Historical Information]

After satisfactory completion of fuel loading and final precriticality tests, nuclear operation of the reactor was initiated. This final stage of startup and testing included initial criticality, low-power testing, and power level escalation. The purposes of these tests were to establish the operational characteristics of the unit and core, to verify design prediction, to demonstrate that license requirements were being met, and to ensure that the next prescribed step in the test sequence could be safely undertaken. Reactor control setpoint verification was also performed during this stage of startup testing.

Tests that were performed from the initial core loading to rated power are summarized in Table 13.3-1.

13.3.1 Initial Criticality [Historical Information]

Initial criticality was established by withdrawing the shutdown and control banks of rod cluster control units from the core, leaving the last withdrawn control bank inserted far enough to provide effective control when criticality was achieved, and then slowly and continuously diluting the heavily borated reactor coolant until the chain reaction was self-sustaining.

Successive stages of rod cluster control bank withdrawal and of boron concentration reduction were monitored by observing change in neutron count rate as indicated by the regular plant source range nuclear instrumentation as functions of rod cluster control bank position and, subsequently, of primary water addition to the reactor coolant system during dilution.

The inverse count rate ratio was monitored as an indication of the nearness and rate of approach to criticality of the core during rod cluster control bank withdrawal and during reactor coolant boron dilution. The rate of approach toward criticality was reduced as the reactor approached extrapolated criticality to ensure that effective control was maintained at all times.

Relevant procedures specified alignment of fluid systems to allow controlled start and stop and adjustment of the rate of approach to criticality, indicated values of core conditions under which criticality would be expected, and identified chains of responsibility and authority during reactor operations.

13.3.2 Zero-Power Testing [Historical Information]

Upon establishment of criticality a prescribed program of reactor physics measurements was undertaken to verify that the basic static and kinetic characteristics of the core were as expected and that the values of kinetic coefficients assumed in the safeguards analysis were indeed conservative.

Measurements made at zero power and primarily at or near operating temperature and pressure included verification of calculated values of rod cluster control group and unit worths, isothermal temperature coefficients under various core conditions, differential boron concentration worth, and critical boron concentrations as a function of rod cluster control group configuration. Preliminary checks on relative power distribution were made in normal and abnormal rod cluster control unit configurations.

Concurrent tests were conducted on the plant instrumentation including the source and intermediate range nuclear channels. Rod cluster control unit operation and the behavior of the associated control and indicating circuits were demonstrated.

Detailed procedures specified the sequence of tests and measurements to be conducted, and the conditions under which each was to be performed to ensure the relevancy and consistency of the results obtained. These tests covered a series of prescribed control rod configurations with intervening measurements of differential control rod worths and boron worth during boron dilution or boron injection. As the successive configurations were established, the measurement techniques used were:

1. Dynamic temperature coefficient measurements – Differential moderator coefficient measurement made by continuously increasing or decreasing the moderator average temperature and observing the resultant change in core reactivity.

2. Dynamic control rod worth measurements - Control rod differential worth measurements made by monotonically withdrawing or inserting selected control rods or groups of rods and part-length [***Note*** *- Subsequent to initial plant operation (during the Cycle 2/3 refueling outage), the part-length rod cluster control assemblies were removed from the reactor.*] rods and observing the resultant change in core reactivity.

3. Dynamic boron worth measurements - Differential boron worth measurements made by monotonically increasing or decreasing main coolant boron concentration and observing the resultant change in core reactivity.

13.3.3 Power Level Escalation [Historical Information]

In order to ensure that operation of the core would be as expected in all respects, and that achievement of rated power was under carefully controlled conditions, a power escalation test program was established to carry the plant from completion of zero-power physics testing through full-power operation. The power escalation test program provided for stepwise achievement of full power, with careful review of significant core parameters at each step, to ensure that fuel and control rod mechanical performance, flux distribution, temperature distribution hot channel factors and reactivity control worths were acceptable before additional escalation was undertaken.

The power escalation test program provided for measurements to be made at convenient power levels in the vicinity of minimum self-sustaining power, discrete levels approaching 100-percent, and at rated power. In each case, progression to higher levels was contingent upon acceptable core performance.

Additional reactor physics measurements were made and the ability of the reactor control and protection system to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations was verified. At prescribed power levels, the dynamic response characteristics of the reactor coolant and the steam systems were evaluated.

The sequence of tests, measurements, and intervening operations is prescribed in the power escalation procedures together with specific details relating to the conduct of the several tests and measurements. The measurement and test operations during power escalation are similar to those during normal operation.

The preparation for power escalation is described below. In order to monitor performance, the following analytical results were on hand before power escalation was undertaken:

1. Expected values for local power ratios in each of the incore flux-detector thimbles.

2. Expected values for relative power in each fuel assembly and in individual fuel rods of interest in various control group configurations.

3. Expected values of power peaking factors.

4. Combined power and programmed temperature reactivity defect as a function of primary power level at expected boron concentrations.

5. Equilibrium xenon reactivity defect as a function of primary power level.

6. Identification and integral reactivity worth of the most significant single rod cluster control assemblies in the control group, when fully withdrawn, with various operating control rod configurations, for both full- and part-length rods.

7. Identification and integral reactivity worth of the most significant single rod cluster control assemblies among all groups, for both full- and part-length rods.

Other conditions that were to be met before commencement of the power escalation test program were as follows:

1. The following plant conditions were established:

    a. The zero-power reactor physics test program had been successfully completed as prescribed. Experimental values of zero power reactivity parameters had been deduced and were available for guidance in the elevated power program.

    b. Discrepancies between analytically predicted and experimentally measured values of physics parameters had been identified and appropriate revisions had been made in the values of expected primary coolant boron concentrations and rod cluster control group positions listed in the power escalation test sequence.

    c. The reactor coolant system and all required components of the secondary coolant system were fully assembled, mechanically tested, and ready for service as required.

d. All control, protection, and safety systems were fully installed; all required preoperational tests satisfactorily completed; and all components ready for service as required.

e. The reactor coolant was at required temperature, pressure, and lithium and boron concentration.

f. Demineralized water was available in adequate quantity for extensive boron dilution.

g. Concentrated boric acid solution was available in sufficient quantity to permit increases in main coolant boron concentration as required.

h. All special equipment and instrumentation required for the power escalation test program was installed and calibrated and available for service as specified.

i. Thermocouple correction constants derived from the hot, isothermal calibrations.

j. Reactor coolant flow coastdown measured and found acceptable.

2. A pretest checkoff list indicating the required status of all systems and auxiliary equipment affecting the power escalation test program was available. The pretest checkoff list included, but was not limited to, provisions for verification and certification of all items specified in item 1, above.

3. Experimental procedures, suitable for executing the power escalation test sequence, were available for distribution to all personnel concerned with the power escalation test program.

4. The procedure, schedule, and personnel assignments and responsibilities were thoroughly discussed with and understood by the operational and experimental personnel.

The following tests were conducted during the power escalation test program:

1. <u>Electrical trip testing</u> - Electrical tripping relays that are initiated by plant on-power malfunctions were retested and the consequent trip sequence rechecked under operating conditions for correct operation and sequence.

2. <u>Turbine trip testing</u> - The turbine protection system was checked to confirm that the appropriate initiation would either trip the turbine through the main trip solenoid or would mechanically trip the turbine. As the various setpoints or status conditions were reached, the trip or runback functions were verified.

3. <u>Elevated power reactivity coefficient evaluation</u> - During the approach to full power and during initial operation at power, a sequence of reactor physics measurements was carried out to determine experimentally the power and temperature coefficients and power defects at various power levels, differential (full- and part-length) control rod worth and boron worths during boron dilutions, and xenon worth during initial operation. Measurements techniques were:

a. Dynamic differential power coefficient - Differential power coefficient measurements were made at elevated power over a limited range in power level by initiating a small power level change. The change in core reactivity associated with the compensating control rod motion is related to the net change in power level.

b. Elevated power transient response evaluation - As the power level was increased during the initial power escalation, a series of transient response measurements was made to determine plant response to load changes. The test technique in each case consisted of establishing the transient change in plant conditions and closely monitoring the system response during and after the transient period. The responses of system components were measured for 10-percent loss of load and recovery, loss of load with steam dump, turbine trip, loss of reactor coolant flow, and trip of single rod cluster control units. Reactor coolant coastdown was also measured.

c. Elevated power determination of power distribution - At successive power levels and in prescribed control rod configurations (full- and part-length), measurements of flux and power distributions within the core were made and nuclear hot channel factors evaluated. Use was made of the miniature incore flux detector system and of the incore thermocouples to determine the nuclear power and thermal and hydraulic conditions within the core. Ex-core nuclear instrumentation was calibrated to indicate actual incore axial power distribution.

d. Determination of primary coolant flow rate - Primary coolant flow rate was evaluated by measuring primary coolant pump power and elbow tap pressure differential.

e. Verification of remote control stations - After the plant was certified to operate at elevated power levels, the capability for manually taking the plant to hot shutdown from stations remote from the control room was verified. This test demonstrated that controls and information available in the local control stations were functioning properly and were sufficient to permit the operators to trip the plant, control heat removal, and borate in an orderly manner to reach and maintain the reactor in a hot shutdown status should the control room ever become uninhabitable.

TABLE 13.3-1 (Sheet 1 of 5)
Initial Testing Summary [Historical Information]

Test	Conditions	Objectives	Acceptance Criteria
RCC_1 unit drop tests	1. Cold shutdown 2. Hot shutdown	To measure the scram time of RCC units under full flow and no flow conditions	Droptime less than value assumed in safety analysis
Thermocouple/RTD intercalibration	Various temperatures during system heatup at zero power	To determine in—place isothermal correction constants for all core exit thermocouples and reactor coolant RTDs	RTDs verify that RTD system meets setpoint requirements of Technical Specifications
Nuclear design check tests	All two dimensional RCC control group configurations at hot, zero power	To verify that nuclear design predictions for endpoint boron concentrations, isothermal temperature coefficient, and power distributions are valid	FFD and SAR_2 limiting values for $\delta\rho/\delta T$, $F_{\Delta}H$
RCC control group calibration	All RCC control groups at hot, zero power	To verify that nuclear design predictions for control group differential worths with and without part—length RCC units are valid	FFD and SAR limiting values for $\delta\rho/\delta h$, $\Delta\rho/h$
Power coefficient measurement	0-percent to 100-percent of full power	To verify that nuclear design predictions for differential power coefficient are valid	FFD and SAR limiting values for $\delta\rho/\delta q$
Automatic control system checkout	Approximately 20-percent	To verify the control system response characteristics for the: a. Steam generator level control system b. RCC automatic control system c. Turbine control system	No safety criteria applicable

TABLE 13.3-1 (Sheet 2 of 5)
Initial Testing Summary

Test	Conditions	Objectives	Acceptance Criteria
Power range instrumentation calibration	During static and/or transient conditions at: 30-percent 70-percent 90-percent 100-percent	To verify all power range instrumentation consisting of: power range nuclear channels, in—core flux mapping system, core exit thermocouple system, and reactor coolant RTDs are responsive to changes in reactor power level and power distribution, and to intercalibrate the several systems	Verify that setpoints cited in Technical Specifications are met
Load swing test	± 10-percent steps at: ~40-percent to 50-percent ~100-percent	To verify reactor control system performance	No safety criteria applicable
Plant trip	Full load rejection from: ~50-percent ~100-percent	To verify reactor control performance	Proper operation of steam dump and feedwater overrides.
Pressurizer effectiveness test	Hot, shutdown	To verify that pressurizer pressure can be reduced at the required rate by pressurizer spray actuation	No safety criteria applicable
Minimum shutdown verification	Hot, zero power	To verify the nuclear design prediction of the minimum shutdown boron concentration with one “stuck” RCC unit	Verify stuck rod shutdown criteria
Psuedo ejection test	Hot, zero power	To verify nuclear design predictions of effects on core reactivity and power distribution of ejection of one RCC unit from a fully inserted control group	FFD and SAR limiting values for $F_{\Delta H}$, reactivity insertion

TABLE 13.3-1 (Sheet 3 of 5)
Initial Testing Summary

Test	Conditions	Objectives	Acceptance Criteria
Pseudo ejection test	~30-percent of rated power	To verify nuclear design predictions of effects on core reactivity and power distribution of ejection of one RCC unit from typical operating configuration.	FFD and SAR limiting values for $F_{\Delta H}$, reactivity insertion
Loss of flow test	Hot shutdown	Measure reactor coolant flow coastdown following trip of reactor coolant pumps	Flow coastdown no faster than FFD and SAR curves
Power redistribution follow	~70-percent of rated power	To verify that ex—core nuclear instrumentation adequately monitors changes in core power distribution under transient xenon conditions	FFD and SAR symmetric offset F_Q correlation
Static RCC drop test	~50-percent of rated power	To verify that a single RCC unit inserted fully or part way below the control bank can be detected by ex—core nuclear instrumentation and core exit thermocouples under typical operating conditions and to provide bases for adjustment of protection system setpoints	Inserted rod detectable with instrumentation
RCC insertion test	~50-percent of rated power	To determine the effect of a single fully inserted RCC unit on core reactivity and core power distribution under typical operating conditions as bases for setting turbine runback limits	See next test

TABLE 13.3-1 (Sheet 4 of 5)
Initial Testing Summary

Test	Conditions	Objectives	Acceptance Criteria
Dynamic RCC drop test	~70-percent of rated power	To verify automatic detection of dropped rod, and subsequent automatic rod stop and turbine cutback	Required power reduction and rod withdrawal block accomplished
Load reduction test	~50-percent reduction from ~70-percent ~50-percent reduction from 100-percent	To verify reactor control system	No safety criteria applicable
Part—length group operational maneuvering	~90-percent	To verify that the part—length RCC maneuvering scheme is effective in containing and suppressing spatial xenon transients	FFD and SAR limiting values for F_Q , $F_{\Delta H}$
Load cycle test	~40-percent ~85-percent	To verify that all plant systems are capable of sustaining load follow operations without encountering unacceptable operational limits through a typical weekly cycle	FFD and SAR limiting values for F_Q , $F_{\Delta H}$, shutdown margin
Turbine—generator startup tests	Pre— and Post—synchronization	To verify that the turbine—generator unit and associated controls and trips are in good working order and ready for service	Successful completion of all mechanical and electrical and control functional checks

TABLE 13.3-1 (Sheet 5 of 5)
Initial Testing Summary

Test	Conditions	Objectives	Acceptance Criteria
Turbine—generator	Power level sufficient for turbine auxiliaries to be operating	To verify normal trouble free performance of the turbine—generator at low power	Performance within manufacturers limitations
Control valve tests	~70-percent of rated power	To verify capability of exercising control valves at significant load and evaluate function of valves and controls	Normal trouble free operation
Acceptance test run	100 hours at rated full power	To verify reliable steady state full power capability	100 hours reliable equilibrium plant operation at full power

Notes:
1. Rod cluster control.

2. Final facility description and safety analysis report.

13.4 OPERATING RESTRICTIONS

13.4.1 Safety Precautions

[Historical Information] Measurements and test operations during zero-power and power escalation phases are always performed under several active trip functions. Any verification program is concluded by several trip functions if the program attempts to violate any of the criteria of the protective circuitry. Furthermore, to ensure that transients are concluded early in the life of the transient, several of the setpoints of the trip functions are reduced, as referenced in Chapter 7.

Measurements are made at various points in the power escalation program as power level is increased. Considerations are made of the instrument accuracy and extrapolations are made for these parameters before proceeding in the program, including both instrument inaccuracies and uncertainties. A continuing verification is then made that the reactor parameters are no more limiting than those assumed in the accident analysis, which are the most limiting values.

Each power step is relatively small, so that a high degree of certainty is associated with the prediction of plant parameters. The accuracy of the prediction obtained for each power level is a major factor in determining further power escalation.

The reactor protection system ensures that the public safety is further protected, as stated above.

13.4.2 Initial Operation Responsibilities

Ultimate responsibility for the facility rested with the holder of the operating license. During the transition from a construction oriented project to a commercial power-producing plant, equipment and systems were tested to prove their capability in accordance with design criteria. Test procedures for the initial startup program were written and approved by both Westinghouse and Con Edison prior to plant testing. Post-core-load test procedures were prepared by Westinghouse and reviewed prior to performance by Con Edison through the Nuclear Facilities Safety Committee. Pertinent safety comments from the committee were factored into the procedures prior to performance. All tests and test procedures were under the control of the general superintendent of the plant to ensure that proper emphasis was placed on safety by all during these acceptance tests (i.e., each test was reviewed by all responsible parties, initial plant conditions and pre-requisites to the test had been met, and proper personnel were available and understood the test procedures and precautions). Westinghouse provided technical direction for these tests.

As part of the precautions, all licensed senior reactor operators and manufacturer's representatives whose equipment was being tested were instructed to stop a test or a portion of a test if the test was not being performed safely or in accordance with the written test procedures. The test would be promptly continued only if minor modifications to the test procedure were required and the test was approved by the general superintendent or his representative and the Westinghouse representative. If substantial revisions were required, however, the general superintendent would review the change with the same approach as that taken with a new test procedure before the test could be continued.

The Joint Test Group (consisting of responsible WEDCO and Con Edison personnel) reviewed and concurred in the release of test procedures for implementation. Technical responsibility for each individual phase of actual startup resided with the functional group most directly concerned with the results of the test. WEDCO and Westinghouse had onsite representatives of supporting functional groups to provide technical advice, recommendations, and assistance in planning and executing the respective stages of unit startup.

All system operations in the testing program were performed by station operators in accordance with the approved written procedures. These procedures included such items as delineation of administrative procedures and test responsibilities, equipment clearance procedures, test purpose, conditions, precautions, limitations, and sequence of operations. Procedural changes were made only in accordance with an approved standard operating procedure that required review and approval of the changes by experienced supervisory personnel.

Test procedures stating the test purpose, conditions, precautions, limitations, and criteria for acceptance were prepared for each test by WEDCO and/or Westinghouse technical advisors. All such procedures were reviewed and concurrence given by the Joint Test Group in accordance with approved standard operating procedures prior to implementation.

All test results received a preliminary review and evaluation by Con Edison site personnel. Cognizant WEDCO/Westinghouse startup engineers and technical advisors determined the adequacy of test data for verification of design objectives. Detailed analyses of test results and issuance of final test reports were performed by WEDCO site startup and/or Westinghouse engineering and design personnel with input from Con Edison where appropriate. Con Edison reviewed all final test results to determine that design objectives and criteria had been met and gave final approval as to the acceptability of plant components, systems, and operating characteristics of the facility.

13.5 REACTOR COOLANT SYSTEM VIBRATION TESTING PROGRAM [Historical Information]

Two test programs were performed on the Indian Point Unit 2 reactor coolant system to measure the dynamic behavior of the reactor coolant system. The two programs were (1) reactor coolant system impedance test and (2) reactor internals and reactor coolant system loop vibration test under steady-state and transient conditions.

13.5.1 Reactor Coolant System Impedance Test

The purpose of the impedance test was to determine the natural frequencies, mode shapes, and damping of the main components of the reactor coolant system. These tests were performed with the reactor coolant system filled with water and were performed prior to the installation of the core and control rods. The reactor coolant and charging pumps were not in operation during this test.

Electromagnetic shakers were attached at several points on one of the reactor coolant system loops so that normal modes of the structure could be excited. Accelerometers were used to measure the response of the structure. The mode shape and damping at the natural frequencies were then deduced from acceleration measurements made at several points on the structure while vibrating at a natural frequency. The shaker was attached at selected locations on the steam generator, reactor coolant pumps, and loop piping; the test plans called for the following locations:

1. Steam generator 21, approximately 65-ft elevation, circumferentially (i.e., tangential to the wall of the vapor container).

2. Steam generator 21, between the 100-ft elevation and the 120-ft elevation.
3. Main coolant pump 21, approximately 62-ft elevation, circumferentially.
4. Main coolant pump 21, approximately 83-ft elevation, circumferentially.
5. Main coolant pump 21, approximately 83-ft elevation, radially.
6. Intermediate leg, loop 21, approximately 54-ft elevation, radially.

Thirteen monitoring accelerometers were attached to the structure at the locations specified in Table 13.5-1 under external transducers. In addition, hand-held accelerometers were moved from point to point to establish the exact mode shape. All shakers and accelerometer cables were routed to a readout station from which the excitation was controlled and response measured.

An initial impedance plot was obtained by exciting the structure at a constant, low force level from a frequency not less than 1 Hz to a frequency not greater than 300 Hz. This was followed by additional sweeps at higher force levels to facilitate detection of natural frequencies that have relatively low response. A determination of the mode shape at each natural frequency of interest was made by measuring the amplitude and phase of the acceleration response at a large number of points relative to the drive point.

Data from which damping could be deduced were obtained by suddenly opening the electrical input of the shaker while driving at a natural frequency and recording the resulting decrement.

13.5.2 Steady-State and Transient Internals and Loop Vibration Measurements

The objectives of the instrumentation program for the second program of testing were:

1. To obtain data that provided increased confidence in the adequacy of the internals structures by establishing the design margins at key locations on the structure. The strain gauge and maximum displacement indicators were used primarily for this purpose.

2. To obtain data that could be used to develop improved analytical tools for the prediction of internals vibrations. Comparison with the 1/7 scale model data and establishing model validity were part of this objective.

Instrumentation was provided for the reactor coolant system major components, that is, reactor vessel, reactor internals, reactor coolant piping, reactor coolant pumps, and steam generators.

13.5.2.1 Reactor Vessel and Loop Piping

Six accelerometers were located on the vessel, three on the vessel head studs and three on the bottom of the vessel. The six vessel transducers were arranged so that the rigid body motion of the vessel could be measured. The loop piping was instrumented with pressure transducers installed in temperature wells on an inlet and outlet leg. In addition, the data from the external transducers were correlated with the internals data to establish remote estimation of internals motions.

13.5.2.2 Steam Generators

One of the steam generators was instrumented in the same manner as described in Section 13.5.1 and its gross motion measured. The dynamic analysis performed on the steam generator tube bundle is described in Chapter 4.

13.5.2.3 Reactor Coolant Pumps

One of the reactor coolant pumps was equipped with three accelerometers mounted at the top of the motor support stand (see Table 13.5-1). They were mounted in a horizontal plane to pick up circumferential and radial vibrations of the pump. Prior to vibration testing (during preoperational tests), the reactor coolant pumps were checked to ensure that they were within limits. The balance and alignment were adjusted if they were not within limits initially (see Chapter 4 for further description).

13.5.2.4 Reactor Internals

The reactor internals were monitored with strain gauges, accelerometers, pressure transducers, and maximum-displacement indicators. There were 46 strain gauges, 14 accelerometers, 5 pressure transducers, and 14 maximum-displacement indicators.

The instrumentation was used as follows:

1. Guide Tubes - The instrumentation used on the guide tubes was the same as that used on the Zorita and Ginna reactors. Three guide tubes were instrumented with strain gauges. The central guide tube was selected because it would have no set cross flow velocity during four-pump operation; a guide tube near the outlet nozzle at approximately 150 degrees was selected because it was expected to have the highest cross flow velocity with the initial complement of guide tubes. A guide tube near the opposite outlet nozzle at approximately 330 degrees was selected because it was expected to have the highest cross flow velocity for plutonium recycle operation. In the 1/7 scale model tests for Indian Point Unit 2, the guide tube located at about 150 degrees was similarly instrumented. These data were used to compare the scale model with the actual plant.

   The response of the guide tubes over the expected range of vibration frequencies was measured with strain gauges and accelerometers to provide strain versus amplitude data and to ensure that the proper location for the strain gauges had been chosen prior to installation in the reactor vessel.

2. Upper Core Barrel - Strain was measured at two locations on the core barrel: (1) just below the core barrel flange and (2) at the upper to lower core barrel weldment, which is a reduced cross-section elevation (see Table 13.5-1). In addition, an axial strain gauge was placed on the outside surface of the barrel, radially inward from the centerline of an inlet nozzle. This gauge was used to obtain an indication of the stress due to the ram effect of the inlet flow against the core barrel and to compare with previous data taken at this location on the 1/7 scale Indian Point Unit 2 model, the 1/13 ENEL/SENA model, and the Obrigheim plant.

   Accelerometers were located on the upper core barrel to determine the vibration of the upper core barrel in its shell modes. This information contributed significantly to understanding the upper barrel strain gauge readings.

Accelerometers were also placed on two thermal shield support blocks to obtain information on the vibration of the core barrel in its ring modes and beam modes. Data were available from the 1/7 scale model at similar locations.

3. <u>Thermal Shield</u> - The measurement of the maximum stress in the thermal shield with a reasonable number of strain gauges was impossible because of the number and nonuniform spacing of supports and the flexibility of the core barrel. The most highly strained bolt that fastens the top of the shield to the core barrel was instrumented with four strain gauges. One of the four gauges was redundant so that loss of one gauge would not result in the loss of all information from this location. To measure the desired strains, the gauges were in a vertical plane passing through the core centerline when the final torque on the bolt was reached (see Table 13.5–1).

Three flexures were instrumented. The locations of the gauges were 0 degrees, 90 degrees, and 240 degrees. These gauges provided the data needed to determine the forces in each of the instrumented flexures.

Three accelerometers were located at the mid-elevation of the shield and one near the bottom to provide data to assist in the interpretation of the strain gauge results and to compare with 1/7 scale model data. Supporting data were obtained from model and full-scale impedance tests.

Pressure measurements were made at the inside and outside wall of the thermal shield. Four pressure transducers to measure the fluctuating static pressure were located near the top (82.5 degrees) and bottom (28 degrees) of the thermal shield.

Fourteen maximum displacement indicators were installed into the thermal shield snubber holes, which were not occupied by pressure transducers (eleven at the upper end and three at the lower end).

The maximum decrease in the proximity of the thermal shield to the core barrel and the vibratory motion of the thermal shield relative to the core barrel were obtained from these indicators by interpretation of styli scratches.

4. <u>Upper Core Plate</u> - Four accelerometers on the upper core plate were used to define the horizontal motion of the upper core plate. This information was used to determine the degree to which base motion excites the guide tubes and support columns (refer to Table 13.5-1).

5. <u>Top Support Plate</u> - A pressure transducer was mounted on the top support plate to be sensitive to vertical pressure fluctuations in the upper plenum. In addition to providing pressures in the upper plenum it was useful in relating the other pressure transducer signals to each other. A pressure transducer was placed in a similar location in the Obrigheim reactor.

13.5.2.5 Instrumentation Description

Transducers measuring strain, acceleration, and pressure as well as maximum displacement indicators were used.

1. Strain gauges - The strain gauges were integral lead gauges similar to those used for the Zorita and Ginna experiments. The minimum sensitivity was greater than 3 μin./in. from 0 to 1000 Hz.

2. Accelerometers - Piezoelectric accelerometers having a sensitivity of approximately 200 pc/g were used with resolution greater than 0.005 g from 5 Hz (± 0.002-in.) to 1000 Hz.

3. Pressure transducers - Piezoelectric pressure transducers were used, which had a resolution of 0.2 psi. The diaphragms of the pressure transducers were flush with the surface where pressure was measured.

4. Maximum-displacement indicators - The maximum-displacement indicators were similar to those used in the Zorita and Ginna experiments. The internal spring-loaded plunger within the displacement pin was designed to follow the relative cyclic motion between the thermal shield and core barrel, thus causing the two stationary spring-loaded styli to leave small markings on the plunger. These marks provided a direct indication of the magnitude of the vibratory motion. The displacement indicators consisted of a cylindrical pin held by means of a clamping fit within a housing block mounted on the thermal shield. The pin was assembled and adjusted within the block so that it was tight against the outer diameter of the core barrel. Sufficient clamping force was exerted on the pin to ensure that the pin would move within the housing block only by a relative motion of the thermal shield toward the core barrel. This created a gap between the end of the pin and the core barrel that was measured during the post hot functional inspection. These measured gaps provided an indication of the total relative motion between the thermal shield and core barrel resulting from thermal differential expansion, hydraulic forces, and vibration.

13.5.2.6 Test Conditions

For these tests the following conditions were required:

1. During cold hydrostatic testing, data were taken at one primary coolant temperature (less than 150°F). This temperature was established by the temperature that existed when time for the testing occurred in the schedule. The temperature was kept within ±20°F during the testing.

2. During the hot functional tests, data were taken at a low temperature (less than 150°F) and at the maximum test temperature. Again, the main coolant temperature was kept within ±20°F while data was being taken. During heatup, a selected number of instruments were monitored continuously.

3. At the completion of hot functional testing, all instruments were removed except six strain gauges on two guide tubes, three strain gauges on the core barrel, one pressure transducer on the top support plate, and the thirteen

accelerometers on the outside structure. These instruments were monitored during precritical testing after the core was loaded. The measurements were made on these instruments for steady-state and transient conditions. Data were taken during control rod exercising, with and without moving the rods in the instrumented guide tube at the same temperature conditions as specified in items 1 and 2, above. For the above tests, data were recorded during startup transients, shutdown transients, and steady flow with several combinations of reactor coolant pumps running including each pump operating individually and all four pumps operating simultaneously. At the first refueling, the internal transducers were removed.

This reactor coolant system testing program, when coupled with experience from offsite testing, model testing, and data from other testing programs on operating plants provided assurance that inservice vibration monitoring instrumentation is not required. (See Chapter 4 for a discussion on the metal impact monitoring system installed since the original test program.)

Table 13.5-1 (Sheet 1 of 4)
[Historical Information]
Transducer Locations for Vibration Experiments

Structure	Outer Wall	Inner Wall	Elevation	Angle, Degrees	Dir. Of Sensitivity	Accelerometer	Pressure Transducer	Strain Gage
Core Barrel	X	X	Upper Core Barrel	0	A			2
	X	X	Below Flange	90	A			2
	X	X	Weldment	270	A			2
	X		Behind Inlet Nozzle	67-1/2	A			1
	X	X	Weldment Upper	0	A			2
	X	X	Lower Core Barrel	0	C			2
	X	X		90	A			2
	X	X		90	C			2
		X	Nozzle Elevation	0	R	1		
	X			45	R	1		
		X		90	R	1		
		X		270	R	1		
	X			22-1/2	R	1		
	X		On Thermal Shield Support Blocks	112-1/2	R	1		

A = AXIAL
C = CIRCUMFERENTIAL
R = RADIAL

Table 13.5-1 (Sheet 2 of 4)

Transducer Locations for Vibration Experiments

Structure	Outer Wall	Inner Wall	Elevation	Angle, Degrees	Dir. Of Sensitivity	Accelerometer	Pressure Transducer	Strain Gage
Thermal Shield	X	X	Snubber Pin Holes Top	82.5	R		2	
	X	X	Bottom	28	R		2	
	X		Mid Elevation	0	R	1		
				90	R	1		
			Flexures	0	R			6
				90	R			6
				240	R			6
			Top Support Bolt	67-1/2	R			4
	X		Mid Elevation	270	R	1		
	X		Near Bottom	90	R	1		
Upper Core Plate			Top Surface	0	R	1		
				0	C	1		
				180	R	1		
				180	C	1		
Top Support Plate			Bottom Surface		A		1	

Table 13.5-1 (Sheet 3 of 4)

Transducer Locations for Vibration Experiments

Structure	Outer Wall	Inner Wall	Elevation	Angle, Degrees	Dir. Of Sensitivity	Accelerometer	Pressure Transducer	Strain Gage
Guide Tube			Near Top Support Plate					3
								2
			Pos. D-14 (Plut. Recycle)					
			H-8 (Center)					4
			K-2 (Max. Vel.)					
Vessel	X		Vessel Head Studs	0	C	1[a]		
	X			0	R	1[a]		
	X			180	C	1[a]		
	X		Bottom Of Vessel	-	R	1[a]		
	X				C	1[a]		
	X				A	1[a]		
	X		Inlet Leg (21, 22 & 24)				3	
	X		Outlet Leg (21)				1	
Steam Generator No. 21			~65 Feet (Support Pad Elev.)		C	1[a]		
			~120 Feet		C	1[a]		
			(Near Top)		R	1[a]		

Table 13.5-1 (Sheet 4 of 4)

Transducer Locations for Vibration Experiments

Structure	Outer Wall	Inner Wall	Elevation	Angle, Degrees	Dir. Of Sensitivity	Accelerometer	Pressure Transducer	Strain Gage
Main Coolant Pump No. 21			~62 Feet (Support Pad Elev.)		C	1[a]		
			~83 Feet (Top Motor Flange)		C R	1[a] 1[a]		
Intermediate Leg (Loop 21)			~54 Feet (Center of Pipe)		R	1[a]		
Containment Floor			~6 Feet		R C	1 1		

[a] These instruments in addition to portable accelerometers were used during the impedance test to determine mode shapes

13.6 TESTS FOLLOWING REACTOR REFUELING

During the initial return to power following a refueling shutdown or following a cold shutdown where fuel assemblies have been handled (inspection for example), a series of tests are carried out on the new core. The objectives of these tests are:

1. To demonstrate that the core performance during reactor operation will not exceed safety analysis and Technical Specification limits.

2. To verify the nuclear design calculations.

3. To provide the bases for the calibration of reactor instrumentation.

13.6.1 Reload Startup Physics Test Program

A typical reload startup physics test program may include, but is not limited to, the following:

1. Precriticality tests

   Calibration check of the incore thermocouples and reactor coolant resistance temperature detectors.

2. Hot zero power and beginning of core life condition tests

   a. Determination of the isothermal temperature coefficients and all rods out condition and boron end points for the following conditions:
   (1) All rods out of core.

   b. Determination of the differential and integral rod worths for the following banks of control rods:

   (1) Control bank D.
   (2) Control bank C with control bank D inserted.
   (3) Control bank B with control banks C and D inserted.
   (4) Control bank A with control banks B, C and D inserted.

   -OR-

   Determination of the integral rod worth for each individual Control and Shutdown Bank.

   c. Movable incore detector flux map performed at a power level less than or equal to 30-percent.

3. Power ascension tests

   a. Movable incore detector flux maps performed at various power levels.
   b. Overpower ΔT and overtemperature ΔT setpoint determination.
   c. Ex-core/incore instrumentation calibration.
   d. Heat balance/thermal power measurements.
   e. Reactor coolant flow measurements.

Core loading verification is carried out by monitoring the movement of each assembly during actual core loading. The location of each assembly as it is loaded into the core is verified using a detailed procedure prepared from the reload loading pattern. A final loading verification can be carried out visually upon the completion of core loading to verify the as-loaded core against the design loading pattern.

Cold, zero-power physics testing is not included for reload core heatup, initial criticality, and power ascension. Since reactor operations in the initial cold condition are nonexistent, and initial warmup can be accomplished without nuclear heat (pump heat only), no meaningful information could be gained from such cold, zero-power testing.

Hot Testing

Hot, zero-power physics testing is used to verify that the reactor core can be safely operated and that it meets its design objectives. Hot, zero-power physics testing is accomplished with the reactor coolant system temperature and pressure at the no-load conditions.

Initial Criticality

The core conditions are established at their no-load values with all rod cluster controls inserted. A "1/M" plot is maintained during all periods of rod withdrawal and boron dilution.

Determination of Zero Power Flux Level

The ideal flux level for conduct of zero-power physics testing is one in which the flux level is sufficiently high enough to give a high signal-to-noise ratio and at the same time sufficiently low enough to avoid the reactivity feedback associated with nuclear heating.

All-Rods-Out Boron Concentration

Although this test applies to the all-rods-out condition, it may be employed to determine endpoints of other control configurations.

Moderator Temperature Coefficient

The moderator temperature coefficient is determined from the measured all rods out isothermal temperature coefficient to assure that Technical Specification requirements are satisfied.

Differential Rod Worth

Differential rod worth is measured by incrementally moving the rods from one endpoint to another and measuring the reactivity addition per increment of movement. The endpoints used are generally the fully inserted and fully withdrawn core configuration for each control bank. Normally, bank overlap is not used at this time. In order to keep the flux level within the selected span for physics testing, boron is traded for rod position so that the overall reactivity status core and the flux level remain
relatively constant.

Integral Rod Worth

The integral rod worth curves are developed by integrating the differential rod worth curve as a function of rod height. An alternate measurement technique Dynamic Rod Worth (DRWM), can also be used to measure the integral rod worth provided the technique, evaluation criteria, and remedial actions identified in Attachment 4 of Reference 1 are followed. The NRC documented their acceptance of this technique in a Safety Evaluation Report[2].

Power Ascension
The power ascension program involves slow increase in power level up to 100-percent power accompanied by testing to verify that the core is operating within the required limits.

In particular, movable detector flux traces are run at various power levels to ensure that the fuel was properly loaded and that the power distribution is within design limits. Reactor coolant system flow is determined to ensure that the total reactor coolant system flow exceeds the required minimum rate.

For the low-power physics test to measure control rod worth and shutdown margin the reactor may be critical with all but one control rod inserted [Historical Information].

13.6.2 Test Results

Test results are compared against nuclear design results; in all cases acceptance criteria are in accordance with Technical Specification limits. If the cycle reload is such that it falls within the conditions specified below for preparation and submittal of a startup physics test report to the NRC, such a report summarizing the results of the startup tests is so prepared and submitted.

Startup Report

A summary report of the plant startup and power escalation testing shall be submitted following (1) amendments to the license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the appropriate tests identified in the UFSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

REFERENCES FOR 13.6

1. Letter from Nicholas J. Liparulo, Westinghouse to NRC, Document Control Desk, October 10, 1995.

2. Letter from Robert C. Jones, NRC to Nicholas J. Liparulo, Westinghouse, dated January 5, 1996.