14 INITIAL TEST PROGRAM

14.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

Chapter 14 is a historic summary of the initial startup testing programs conducted for the R. E. Ginna Nuclear Power Plant, both at 1300 and 1520 MWt. Therefore, many of the system parameters, such as setpoints, flows, operating modes, and operating procedures have changed and no longer accurately reflect the present plant configuration and variables.

14.1.1 STARTUP AND POWER TESTING AT 1300 MEGAWATTS THERMAL

14.1.1.1 Summary

During the transition from a construction-oriented job to a commercial power producing plant, equipment and systems were tested to prove their capability in accordance with design criteria.

The initial startup test program at the R. E. Ginna Nuclear Power Plant was performed in order to ensure the safe and efficient operation of the plant up to its initial rating of 1300 MWt. The reactor was shown to be stable at all power levels up to 1300 MWt with induced disturbances to the reactor system. Perturbations to the secondary system were 10% load swings, 50% load reductions, and 100% turbine trip. Control rods were used for a dynamic rod drop test, ejected and dropped rod worth measurements, and a xenon oscillation test. Maximum and/or minimum values of critical reactor system parameters during plant transient tests were within allowable limits. In addition, core thermal-hydraulic limits were not exceeded for steady-state or transient situations.

The startup and power testing program results substantiated design predictions. The core thermal and hydraulic performances showed that the core operated within the specified thermal and hydraulic limits. Reactor system stability measurements were within applicable criteria. Control rod reactivity worth measurements and rod insertion scram times were satisfactory.

The results of the preoperational testing program and the operational and transient tests for operation up to 1300 MWt were reported to the NRC in the Technical Supplement Accompanying Application to Increase Power, February 1971 (*Reference 1*). The staff reviewed and reported on these results in the Safety Evaluation issued by letter dated January 20, 1972 (*Reference 2*).

This section (14.1.1) lists the planned initial tests and test objectives from preoperational tests through completion of the 100-hour rated full power acceptance test at 1300 MWt. Section 14.1.2 summarizes the test program conducted in March and April 1972 to increase rated power output from 1300 to 1520 MWt. Section 14.6.1 includes the individual test descriptions for the initial (1300 MWt) startup test program. Section 14.6.2 describes the test program for increasing rated power from 1300 to 1520 MWt.

14.1.1.2 Tests Prior to Reactor Fueling

14.1.1.2.1 Summary

Rochester Gas and Electric Corporation in cooperation with Westinghouse Electric Corporation prepared detailed test procedures prior to scheduled initial testing of systems and determination of reactor physics parameters.

The tests conducted on the engineered safety systems were included under the tests of the containment system, safety injection system, the containment spray system, and the containment recirculation fan coolers (CRFC) and filtration system.

The test objectives incorporated testing of redundant equipment where it was involved.

Abnormal plant conditions were simulated during testing when such conditions did not endanger personnel or equipment or contaminate clean systems. Where predicted emergency or abnormal conditions were involved in the testing program, the detailed operation was provided in the test procedure.

The acceptance criterion for all components and systems was that the test results were acceptable when the test objectives were met within the design specification limits and within the applicable Technical Specifications.

The following is an extensive tabulation of major startup tests and operations performed to place the equipment in the specified system in service. The systems and items tested are listed in approximately chronological order.

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

17. Cold hydrostatic tests.
18. Radiation monitoring system.
19. Nuclear instrumentation system.
20. Radioactive waste disposal system.
21. Sampling system.
22. Hot functional tests.
    - Reactor coolant system.
    - Chemical and volume control system.
    - Sampling system.
    - Auxiliary coolant system.
    - Safety injection system.
    - Waste disposal system.
    - Ventilation system.
23. Primary system safety valves tests.
24. Turbine steam seal and blowdown systems.
25. Emergency diesel electric system.

14.1.1.2.2 Test Objectives

The objectives of the tests prior to reactor fueling were as follows:

1. Switchgear system (electrical tests).

   Ensure continuity, circuit integrity, and the correct and reliable functioning of electrical apparatus. Electrical tests will be performed on transformers, switchgear, turbine generator, motors, cables, control circuits, excitation switchgear, dc system, annunciator system, lighting distribution switchboard, communication system, and miscellaneous equipment. Special attention will be 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 protection devices.
   E. Equipment automatic start tests.
   F. Check exciter for proper voltage buildup.

2. Voice communication systems.

   Verify proper communication between all intraplant stations, for interconnection to commercial phone service and to balance and adjust amplifiers and speakers.

3. Service water (SW) system.

Verify, prior to critical operations, the design head and capacity characteristics of the service water (SW) pumps, the system design flow rate through all heat exchangers, and the specified requirements when operated in the safeguards mode.

4. Fire protection system.

   Verify proper operation of the system by ensuring that the design specifications are met for the fire service booster pump and the 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.

   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.

   Verify system integrity, valve operability, regulating and reducing station performance, and the ability to supply nitrogen to interconnecting systems as required.

7. Reactor coolant system cleaning.

   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 will be made after system cleaning and acceptance. The system, component, or section of a system shall be considered clean when the flush cloth shows no grindings, filings, or insoluble particulate matter larger than 40 microns (lower limit of naked eye visibility). After systems have been flushed clean of particulate matter within the limit specified, the cleanliness integrity of the system will be maintained filled with water which meets the system cold chemistry requirement. After fill and pressurization and prior to hot operation, cold chemistry requirements will be maintained. Oxygen will be analyzed and brought into specification prior to exceeding 200°F.

8. Reactor containment air circulating system.

   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.

   Verify valve and control operability and setpoints, flushing and hydro as applicable, and inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.

10. Auxiliary coolant system.

    Verify component cooling flow to all components and proper operation of instrumentation, controllers, and alarms. Specifically, each of the three loops (i.e., component cooling loop, residual heat removal loop, and spent fuel pool (SFP) cooling loop) will be tested to ensure the following:

    A. All manual and remotely operated valves are operable manually and/or remotely.

    B. All pumps perform according to manufacturers' specifications.

C. All temperature, flow, level, and pressure controllers function to control at the required setpoint when supplied with appropriate signals.

D. All temperature, flow, level, and pressure alarms provide alarms at the required locations when the alarm setpoint is reached and clear when the reset point is reached.

E. Design flow rates established through heat exchangers.

11. Chemical feed system.

Verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.

12. Chemical and volume control system.

Verify, prior to critical operation, that the system functions as specified in the system description and appropriate technical manuals, more specifically:

A. All manual and remotely operated valves are operable manually and/or remotely.

B. All pumps perform to manufacturers' specifications.

C. All temperature, flow, level, and pressure controllers function to control at the required setpoint when supplied with appropriate signals.

D. All temperature, flow, level, and pressure alarms provide alarms at the required locations when the alarm setpoint is reached and clear when the reset point is reached.

E. The reactor makeup control controls blending, dilution, and boration as designed.

F. The design seal-water flow rates are attainable to each reactor coolant pump.

G. The boric acid evaporator package functions as specified in the manufacturers' technical manuals.

13. Containment spray.

Verify performance of the containment spray pumps.

14. Safety injection system.

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 manufacturers' technical manuals, and check the time required to actuate the system after a safety injection signal is received; more specifically:

A. All manual and remotely operated valves are operable manually and/or remotely.

B. All pumps perform their design functions satisfactorily.

C. For each pair of valves to redundant flow paths, disabling one of the valves does not impair remote operation of the other.

D. The proper sequencing of valves and pumps occurs on initiation of a safety injection signal.

E. The fail position on loss of power for each remotely operated valve is as specified.

F. Valves requiring coincidence signals of safety injection and high containment pressure operate when supplied with these signals.

G. All level and pressure units are set at the specified points and provide alarms at the required locations, and clear when the reset point is reached.

H. The time required to actuate the system is within the design specifications.

15. Fuel handling system.

Show that the system design is 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 will be designed to verify:

A. The major structures required for MODE 6 (Refueling), such as the reactor cavity, refueling canal, spent fuel storage, and decontamination facilities, are in accordance with the design specifications.

B. The major equipment required for MODE 6 (Refueling), such as the manipulator crane, spent fuel pool (SFP) bridge, and fuel transfer system, operate in accordance with the design specifications.

C. All auxiliary equipment and instrumentation function properly.

16. Reactor containment high pressure test.

Verify, prior to critical operation, the structural integrity and leaktightness of the containment.

17. Cold hydrostatic tests.

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 nor distortion.

18. Radiation monitoring system.

Verify the calibration, operability, and alarm setpoints of all radiation level monitors, air particulate monitors, gas monitors, and liquid monitors which are included in the operational radiation monitoring system and the area radiation monitoring system.

19. Nuclear instrumentation system.

Ensure that the instrumentation system is capable of monitoring the reactor leakage neutron flux from source range through 120% of full power and that protective functions are operating properly. In particular, the tests will be designed to verify:

A. All system equipment, cabling, and interconnections have been properly installed.

B. The source range detector and associated instrumentation 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 operate properly.

C. The intermediate range instrumentation, reactor protective and control features, high-level reactor trip, and high-level rod stop signals operate properly and that permissive signals for blocking source range trip and source range "high voltage off" operate properly.

D. The power range instrumentation operates properly and that the protective features such as the overpower trips, permissive, and dropped-rod functions operate with the required redundancy and separation through the associated logic matrices, and nuclear power signals to other systems are available and operating properly.

E. All auxiliary equipment such as the comparator and startup rate channel, recorders, and indicators operate as specified.

F. All instruments are properly calibrated and all setpoints and alarms are properly set.

20. Radioactive waste disposal system.

Verify satisfactory flow characteristics through the equipment, demonstrate satisfactory performance of pumps and instruments, check for leaktightness of piping and equipment, and verify proper operation of alarms, instrumentation, and controls. More specifically verify that:

A. All piping and components are properly installed as per design specifications.

B. All manual and automatic valves are operable.

C. All instrument controllers operate to control process at required values.

D. All process alarms operate at required locations.

E. All pumps perform to manufacturer's specifications.

F. All pumps indication and controls are operable at designated stations.

G. The waste gas compressors packages operate as specified in manufacturer's technical manual.

H. The gas analyzer operates as specified in the manufacturer's technical manual.

I. The waste boiler operates as specified in the manufacturer's technical manual.

J. The hydrogen and nitrogen supply packages are sufficient for all modes of operation.

21. Sampling system.

Verify that a specified quantity of representative fluid can be obtained safely and at design conditions from each sampling point. In particular, the tests will be designed to verify:

A. All system piping and components are properly installed.

B. All remotely and manually operated valving operates in accordance with the design specifications.

C. All sample containers and quick-disconnect couplings function properly and as specified.

22. Hot functional tests.

The reactor coolant system will be tested to check heatup (using pump heat) and cooldown procedures, demonstrate satisfactory performance of components prior to installation of the core, verify proper operation of instrumentation, controllers, and alarms, and provide operating conditions for checkout of auxiliary systems.

The chemical and volume control system will be tested to determine that water can be charged at rated flow against normal reactor coolant system pressure, check letdown flow against design rate for each pressure reduction station, determine the response of the system to changes in pressurizer level, check procedures and components used in boric acid batching and transfer operations, check operation of the reactor makeup control, check operation of the excess letdown and seal-water flowpath, and verify proper operation of instrumentation, controllers, and alarms.

The sampling system will be tested to determine that a specified quantity of representative fluid can be obtained safely and at design conditions from each sampling point.

The auxiliary coolant system will be tested to evaluate its ability to remove heat from reactor coolant, verify component cooling flow to all components, and verify proper operation of instrumentation, controllers, and alarms.

The safety injection system will be tested to check the time required to actuate the system after a safety injection signal is received, check that pumps and motor-operated valves are properly sequenced, and verify proper operation of instrumentation, controllers, and alarms.

The radioactive waste disposal system will be tested to verify satisfactory flow characteristics through the equipment, demonstrate satisfactory performance of pumps and instruments, check for leaktightness of piping and equipment, and verify proper operation of alarms.

The ventilation system will be tested to adjust proper flow characteristics of ducts and equipment, demonstrate satisfactory performance of fans, filters, and coolers, and verify proper operation of instruments and alarms.

23. Primary system safety valves test.

    Test and set pressurizer safety and relief valves to ensure that each valve lifts, relieves excess pressure, and reseats.

24. Turbine steam seal and blowdown systems.

    Verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.

25. Emergency diesel electric system.

    Demonstrate that the system is capable of providing power for operation of vital equipment under power failure conditions. In particular the tests will be designed to verify:

    A. All system components have been 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.

14.1.1.3 Final Plant Preparation

14.1.1.3.1 Core Loading

The as-loaded core configuration was specified as part of the fuel core design studies conducted well in advance of plant startup and as such was not normally subject to change at plant startup. In the relatively unlikely event that mechanical damage was sustained during core loading operations by a fuel assembly of a type for which no spare was available onsite, a previously examined alternate core loading scheme whose characteristics closely approximate those of the initially prescribed pattern was to be invoked.

The core was assembled in the reactor vessel in water containing enough dissolved boric acid (usually at least 2000 ppm) to maintain the core multiplication constant at 0.90 or lower and was not subsequently distributed or changed until the end of the core cycle. 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 hours during core loading operations.

Core loading instrumentation consisted of two permanently installed plant source range (pulse-type) nuclear channels and two temporary in-core 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 RG&E. 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.
Minimum count rates of two counts per second, attributable to core neutrons, were required on at least two of the four available nuclear channels at all times during core loading operations. Two artificial neutron sources, each rated at approximately 200 Ci of Polonium-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 which 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 which 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.

An initial nucleus of eight fuel assemblies, the first of which bore an installed neutron source, had been determined to be the minimum source-fuel nucleus which would permit subsequent meaningful inverse count rate monitoring. This initial nucleus was 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 was evaluated by both Westinghouse technical specialists and licensed RG&E operations personnel and concurrent approval to proceed was granted before the next prescribed step could be started.

Criteria for safe loading required that loading operations stop immediately if the following conditions occurred:

A. The neutron count rates on all responding nuclear channels double during any single loading step.

B. The neutron count rate on any individual nuclear channel increases by a factor of five during any single loading step.

A containment evacuation alarm was coupled to the plant source range channels with a setpoint at five times the current count rate to provide automatic indication of high count rate during fuel addition.

In the event that an unacceptable increase in count rate were to be 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 would have been invoked by Westinghouse technical specialists with the approval of licensed operational personnel of RG&E.

Core loading procedures specified alignment of fluid systems to prevent inadvertent dilution of the reactor coolant, restricted the movement of fuel to minimize the possibility of mechanical damage, prescribed the conditions under which loading could proceed, identified chains of responsibility and authority, and provided for continuous and complete fuel and core component accountability.

14.1.1.3.2 Postloading Tests

Upon completion of core loading and installation of the reactor upper internals and the pressure 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, the reactor trip circuits, and the in-core thermocouples were tested at the time of installation. 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.

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. By use of dummy signals, the reactor control and protection systems were made to produce trip signals for the various plant abnormalities that required tripping.

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

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

14.1.1.4 Initial Testing in the Operating Reactor

14.1.1.4.1 General

The objectives of the tests performed after the initial core loading until completion of the 100-hour rated full power acceptance run are summarized in Table 14.1-1.

Tests which were performed from the initial core loading to rated power are summarized in Section 14.6.1.

14.1.1.4.2 Initial Criticality

Initial criticality was established by withdrawing the shutdown and control groups of rod cluster control units from the core, leaving the last-withdrawn control group 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 group withdrawal and of boron concentration reduction were monitored by observing changes in neutron count rate as indicated by the regular plant source range nuclear instrumentation as functions of rod cluster control group position and, subsequently, of primary water addition to the reactor coolant system during dilution.

Primary safety reliance was based on inverse count rate ratio monitoring as an indication of the nearness and rate of approach to criticality of the core during rod cluster control group 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 at which the approach to criticality could proceed, indicated values of core conditions under which criticality was expected, and identified chains of responsibility and authority during reactor operations.

14.1.1.4.3 Zero Power Testing

Upon establishment of criticality, a prescribed program of reactor physics measurements was undertaken to verify that the basic statics and kinetics characteristics of the core were as expected and the values of kinetics 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 coefficient 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 and the adequacy of the control and protection systems were verified under zero power operating conditions.

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.

14.1.1.4.4 **Power Level Escalation**

When the operating characteristics of the reactor at zero power had been verified, a program of power level escalation in successive stages was undertaken. Both reactor and plant operational characteristics were closely examined at each stage and the relevance of the safeguards analysis was verified before escalation to the next programmed level was effected.

Reactor physics measurements were made to determine the magnitudes of the power coefficient of reactivity, rod cluster control group differential worth, and relative power distribution in the core as functions of power level and rod cluster control group position.

Concurrent determinations of primary and secondary heat balances were made to ensure that the several indications of plant power level were consistent and to provide the bases for calibration of the power range nuclear channels. The ability of the reactor control and protection systems to respond effectively to signals from plant primary and secondary instrumentation under a variety of conditions encountered in normal operations was verified.

At prescribed power levels the response characteristics of the reactor coolant and steam systems to dynamic stimuli were evaluated. The responses of system components were measured for 10% loss of load and recovery, full loss of load, turbine trip, loss of flow, and trip of a single rod cluster control unit.

Adequacy of radiation shielding was verified by gamma and neutron radiation surveys in the vicinity of the containment and throughout the plant site.

The sequence of tests, measurements, and intervening operations was prescribed in the power escalation procedures, together with specific details relating to the conduct of the several tests and measurements.

14.1.1.4.5 **Post Startup Surveillance and Testing Requirements**

The equipment verification program was designed to provide assurance that essential systems, which included equipment components and instrument channels, were always capable of functioning in accordance with their original design criteria. These requirements can be separated into two categories:

A. The system must be capable of performing its function, i.e., pumps deliver at design flow and head, and instrument channels respond to initiating signals within design calibration and time responses.

B. Reliability is maintained at levels comparable to those established in the design criteria and during early plant life.

The testing requirements, as described in the Technical Specifications, establish this reliability and, in addition, provide the means by which this reliability is continually reconfirmed. Verification of operation of complete systems is checked at MODE 6 (Refueling) intervals. Individual checks of components and instrumentation are made at more frequent intervals as outlined in the Technical Specifications.

The techniques used for the testing of instrument channels include a preoperational calibration which confirms values obtained during factory test programs. These reconfirmed calibration values become the reference for recalibration maintenance at MODE 6 (Refueling) intervals during plant life. Periodic testing, as defined in the Technical Specifications, includes the insertion of a predetermined signal that will trip the channel bistable. Indication of the operation is confirmed and recorded.

Testing of components is initiated through manual actuation. If response times are important, they are measured and recorded. The capability to deliver output is checked by instrumentation and compared against design data. Allowable discrepancies are established in the vendor manuals, performance specifications, and Technical Specifications.

Rochester Gas and Electric Corporation believes that such testing provides a realistic basis for determining maintenance requirements and as such ensures continued system capabilities, including reliability, equal to those established in the original criteria.

14.1.2 POWER TEST PROGRAM TO 1520 MEGAWATTS THERMAL

An amendment to the operating license was issued on March 1, 1972, which authorized an increase in the plant output from 1300 to 1520 MWt. A diverse and thorough testing program was used in the power escalation performed from March 8 to April 14, 1972.

The program consisted of a number of tests and measurements at power levels of 1300, 1380, 1455, and 1520 MWt. At each of these power levels, in-core flux maps, delta T measurements, containment radiation surveys, and primary coolant activity measurements were performed. Additional flux maps were obtained at 1455 MWt to calibrate the axial offset monitoring. The flux maps, delta T measurements, and the containment radiation surveys all showed very good agreement with predictions.

The response of system components to increases in core power output was studied. The reactor was operated for a short period at 1520 MWt and performed satisfactorily. Core physics parameters agreed well with design data and there was considerable margin to core safety limits. Core instrumentation continued to accurately reflect the behavior of the core.

A detailed discussion of the uprating test program is included in Section 14.6.2. Rochester Gas and Electric Corporation reported the results of the test program to the AEC in a letter dated August 14, 1972 (*Reference 3*).

REFERENCES FOR SECTION 14.1

1. Letter from Lamb, LeBoeuf, Leiby & MacRae to Peter A. Morris, Division of Reactor Licensing, AEC, Subject: Technical Supplement Accompanying Application to Increase Power, February 1971, dated February 8, 1971.

2. Letter from D. V. Skovholt, AEC, to Edward V. Nelson, RG&E, Subject: Safety Evaluation by the Division of Reactor Licensing, R. E. Ginna Power Increase, Enclosure 3, dated January 20, 1972.

3. Letter from R. R. Koprowski, RG&E, to E. V. Block, NRC, Subject: R. E. Ginna Nuclear Power Plant Unit No. 1 Power Escalation to 1520 MWt, March 1972, dated August 14, 1972.

Table 14.1-1
INITIAL TESTING SUMMARY - INITIAL CRITICALITY THROUGH 100 - HOUR ACCEPTANCE TEST

Test	Conditions	Objectives	Acceptance Criteria
Rod cluster control unit drop tests	MODE 5 (Cold Shutdown) MODE 3 (Hot shutdown)	Measure the scram time of rod cluster control units under full flow and no flow conditions	Drop time less than value assumed in safety analysis
Thermocouple/ resistance temperature detector intercalibration	Various temperatures during system heatup at zero power	Determine in-place isothermal correction constants for all core exit thermocouples and reactor coolant resistance temperature detectors	Resistance temperature detectors verify that resistance temperature detector system meets setpoint requirements of Technical Specifications
Nuclear design check tests	All two dimensional rod cluster control groups configurations at hot zero power	Verify that nuclear design predictions for endpoint boron concentrations, isothermal temperature coefficients and power distributions are valid	FSAR limiting values for $\delta\rho/\delta T$, $F_{\Delta H}$
Rod cluster control group calibration	All rod cluster control groups at hot zero power	Verify that nuclear design predictions for control group differential worths with and without partial length rod cluster control units are valid	FSAR limiting values for $\delta\rho/\delta T$, $\Delta\rho/h$
Power coefficient measurement	0% to 100% of full power	Verify that nuclear design predictions for differential power coefficients are valid	FSAR limiting values for $\delta\rho/\delta q$
Automatic control system checkout	~20%	Verify control system response characteristics for the steam generator level control system, rod cluster control automatic control system, and turbine control system	No safety criteria applicable
Power range instrumentation calibration	During static and/or transient conditions at 30%, 70%, 90%, and 100%	Verify that all power range instrumentation consisting of power range nuclear channels, in-core flux mapping system, core exit thermo-couple system and reactor coolant resistance temperature detectors is 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	±10% steps at ~30%, ~70%, and ~100%	Verify reactor control system performance	No safety criteria applicable
Plant trip	Full load rejection from ~30% and ~100%	Verify reactor control performance	No safety criteria applicable
Pressurizer effectiveness	Hot shutdown	Verify that pressurizer pressure can be reduced at the required rate by pressurizer spray actuation	No safety criteria applicable
Circulation (nuclear heat)	~7% of rated power both reactor coolant system pumps off	Verify that natural circulation is established	Enough natural circulation to remove long-term residual heat
Circulation (partial cooldown)	Shutdown - both reactor coolant system pumps off - one steam generator isolated	Verify ability to cooldown with natural circulation	Partial cooldown completed
Minimum shutdown verification	Hot zero power	Verify the nuclear design prediction of the minimum shutdown boron concentration with one "stuck" rod cluster control unit	Measured minimum shutdown boron concentration must be less than the minimum operating boron concentration
Pseudo ejection	Hot zero power	Verify nuclear design predictions of effects on core reactivity and power distribution of ejection of one rod cluster control unit from a fully inserted control group	FSAR limiting values for $F_{\Delta H}$, reactivity insertion
Pseudo ejection	~70% of rated power	Verify nuclear design predictions of effects on core reactivity and power distribution of ejection of one rod cluster control unit from typical operating configuration	FSAR limiting values for $F_{\Delta H}$, reactivity insertion
Power redistribution follow	~70% of rated power	Verify that ex-core nuclear instrumentation adequately monitors changes in core power distribution under transient xenon conditions	FSAR symmetric offset F_Z correlation

Static rod cluster control drop	~70% of rated power	Verify that a single rod cluster control unit inserted fully or part way below the control bank can be detected by ex-core nuclear instrumentation and core exit thermocouple under typical operating conditions and to provide bases for adjustment to protection system setpoints	Inserted rod detectable with instrumentation
Rod cluster control insertion	~70% of rated power	Determine the effect of a single fully inserted rod cluster control unit on core reactivity and core power distribution under typical operating conditions as bases for setting turbine runback limits	See next step
Dynamic rod cluster control drop	~70% of rated power	Verify automatic detection of dropped rod, and subsequent automatic rod stop and turbine cutback	Required power reduction and rod withdrawal block accomplished
Load reduction	~50% reduction from ~70% ~50% reduction from ~100%	Verify reactor control system	No safety criteria applicable
Part-length group operational maneuvering	~90%	Verify that part-length rod cluster control maneuvering scheme is effective in containing and suppressing spatial xenon transients	FSAR limiting values for F_Z, $F_{\Delta H}$
Load cycle	~40% to ~85%	Verify that all plant systems are capable of sustaining load follow operations without encountering unacceptable operational limits through a typical weekly cycle	FSAR limiting values for F_Z, $F_{\Delta H}$, shutdown margin
Turbine generator startup	Pre-and post-synchronization	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
Turbine generator	~30% of rated power	Verify normal trouble-free performance of the turbine generator at low power	Performance within manufacturers' limitations
Control valve	~70% of rated power	Verify capability of exercising control valves at significant load and evaluate function of valves and controls	Normal trouble-free operation
Acceptance run	100 hr at full rated power	Verify reliable steady-state full power capability	100-hr reliable equilibrium plant operation at full power

14.2 INITIAL ORGANIZATION AND STAFFING

14.2.1 INITIAL STARTUP AND OPERATING STRUCTURE

The organization chart in Figure 14.2-1 shows the normal complement of full-time employees for Ginna Station at the time of initial tests and operations. In addition to the 67 employees, the plant management had available technical advice and services from the consultants in nuclear engineering, environmental sciences, medicine, and industrial hygiene. During initial startup and testing of the station, additional personnel were available to assist in the initial testing of the plant. Westinghouse also provided assistance for precritical tests, core loading, achieving criticality, post-critical tests, and plant performance tests.

Ultimate responsibility for startup and testing rested with RG&E, the holder of the facility license.

The plant organization in general was similar to that used successfully in the past for other RG&E thermal plants. The major difference was the selection and training of the personnel, the Operations Engineer, and the addition of radiation protection personnel. The responsibility of the plant organization was to operate the plant efficiently and competently in a manner consistent with safety to the public and plant personnel. Plant supervision was under the direction of the Superintendent of Electric and Steam Generation who received technical support from the RG&E Engineering Department.

14.2.2 ONSITE TRAINING PRIOR TO STARTUP

The onsite training school program began in the spring of 1968. It was expected that initial criticality would occur approximately 11 months after the start of the training school, leaving ample time for the proper training of personnel. The instructors for the school were members of the supervisory staff of the RG&E Operating Department and technical personnel from the engineering group. The latter group discussed the systems and their components from the designer"s viewpoint. Engineering and scientific personnel from Westinghouse and Gilbert Associates, Inc., assisted in presentations covering design aspects of equipment supplied by their respective organizations. Technical consultants and other vendors" representatives were invited to discuss topics of special interest.

The instructional responsibilities were divided among the plant supervisory staff as follows:

Plant Engineer	Atomic structure, nuclear theory, reactor physics and the containment.
Maintenance Engineer	Reactor kinetics, primary coolant system and components, fuel handling and emergency procedures.
Reactor Engineer	Basics of electricity, plant electrical systems, reactor control, core layout, safety injection system, and administrative procedures.
Results and Test Engineer	Radiation detection, nuclear instrumentation, and auxiliary coolant system.
Operations Engineer	Chemical and volume control system and operating instructions.

Assistant Plant Superintendent	Radioactive waste treatment, Technical Specifications, and site contingency procedures.
Plant Superintendent	Secondary plant systems.
Health Physics Department	Radiation protection, health physics, plant ventilation, water treatment, and sampling.
Westinghouse training coordinator	Instrumentation and control.

The supervisory staff was heavily assisted in class preparation and presentation by the six shift foreman candidates.

All reactor operation trainees were experienced power plant operators. During the training program, time was allowed such that plant systems and operations were studied, observed during construction, and operated during checkout.

14.3 TEST PROCEDURES

14.3.1 PRE FUEL LOADING TESTS

Test procedures were written and approved by both Westinghouse and RG&E prior to plant testing. Westinghouse provided technical direction for the tests; however, all tests and test procedures were under the control of the Plant Superintendent to ensure that proper emphasis was placed on safety by all during the tests.

14.3.2 POST FUEL LOADING TESTS

Rochester Gas and Electric Corporation, in association with Westinghouse, developed the pre-operational and post-operational tests which were to determine the adequacy of design installation and operability of those systems affecting safety. These tests were conducted after their approval by both Westinghouse and the RG&E operating and engineering personnel. All tests were performed under the direction of Westinghouse, using RG&E personnel as operators and data collectors.

The functional system testing was a joint effort between Westinghouse and RG&E. Rochester Gas and Electric Corporation personnel actively participated during this testing phase by performing the various system test requirements and recording all pertinent data.

14.4 CONDUCT OF TEST PROGRAM

14.4.1 CONDUCT OF INITIAL TEST PROGRAM

The Plant Superintendent was responsible for ensuring that each test had been reviewed by all responsible parties, that initial plant conditions and prerequisites to the test had been met, and that proper personnel were available and understood the test procedures and precautions.

As part of the precautions, all licensed senior reactor operators and manufacturers' 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 was to be promptly continued only if minor modification to the test procedure was required and it was approved by the Plant Superintendent, or the Superintendent's representative, and the Westinghouse representative. If substantial revision was required, however, the Plant Superintendent had to review the change with the same approach as a new test procedure before the revision could be approved and the test could be resumed.

During the post fuel loading test period the lines of authority were as follows: all Westinghouse instructions were to be issued to the specified senior reactor operator on duty who would in turn relay all messages to the reactor operator or operating personnel necessary to perform the task.

If at any time during testing the reactor operators or other responsible cognizant personnel were to feel that an unsafe condition existed or could occur or the test was not being done in accordance with procedure, they were to take steps to interrupt the test and put the plant in a safe condition.

The questionable condition was then to be reviewed by Westinghouse and the RG&E Plant Operations and Review Committee and the RG&E Nuclear Safety Audit and Review Board as appropriate. If considered unsafe, the procedure was to be rewritten in a safe manner and then the test was to be reperformed.

Rochester Gas and Electric Corporation Engineering and consultants and Westinghouse Design had to agree on the general program including the extrapolations and implications of previous results, and had to resolve anomalies and approve resolution of disagreement among the senior operations personnel above.

Administrative responsibility relating to Westinghouse Atomic Power Department (WAPD) personnel onsite and to normal communications between WAPD personnel onsite and the customer rested with Westinghouse Nuclear Power Service.

Technical responsibility at each individual phase of actual startup rested with the senior onsite representative of the functional group most directly concerned with the results of that phase. In the event of apparent overlap of areas of technical responsibility, resolution had to be obtained within the functional group.

Onsite representatives of supporting functional groups provided technical advice, recommendations and assistance in planning and executing the respective phases of plant startup.

If apparent deviations of test results from design predictions or acceptance criteria were revealed or if other apparent anomolies developed, relevant test data were reviewed and, if necessary, the tests themselves were repeated or supporting tests were made to verify the results. If the apparent discrepancy or anomoly was found to be real, the situation was reviewed by the Plant Operations and Review Committee to determine whether a question of plant safety was involved. If such was found to be the case, evaluation of the effect of the discrepancy or anomoly on plant safety was accomplished at the appropriate level of review. If after evaluation it was determined that an unreviewed safety question existed, a detailed evaluation of the consequences of possible accidents under actual (as opposed to predicted) conditions was made. Similar testing under more stringent conditions was not resumed until any questions relating to reactor safety had been resolved satisfactorily.

14.4.2 REVIEW, EVALUATION, AND APPROVAL OF TEST RESULTS

Analysis of test data was made by Westinghouse and RG&E operational and engineering personnel to ascertain the system capability to operate safely and fulfill all design specifications.

Equipment and/or systems were considered tested as satisfactory when accepted by the RG&E Operating Department, RG&E Engineering Department, and Westinghouse Atomic Power Department.

14.4.3 TEST RECORDS

Test records were turned over to RG&E after the testing was satisfactorily completed and will be retained in Central Records by RG&E for the life of the plant.

14.5 TEST PROGRAM SCHEDULE

14.5.1 INITIAL CRITICALITY TO ACCEPTANCE

The reactor was made critical for the first time on November 9, 1969, at 0530. Following initial criticality, the initial low power physics testing was carried out. The initial reactor physics tests were completed satisfactorily, but the plant had to be cooled down on November 19, 1969, because of a problem with the pressurizer level instrumentation.

The reactor was brought critical again on November 28, 1969, and initial synchronization of the generator occurred on December 2, 1969.

The reactor power was brought up to 30% of thermal rating on December 3, 1969, with the necessary testing at 30% power completed in approximately 1 week.

The planned low power physics testing that was done through December 1969 showed that the agreement between experimental and predicted values was excellent.

The unit was shut down on December 24, 1969, for further maintenance and for reactor startup training of the operators.

On January 16 and 17, 1970, tests were conducted to show that with the power range detectors, core outlet thermocouples, and in-core moveable detectors the operators could detect and verify that a control rod cluster was misaligned with respect to its bank before core design limits are exceeded during MODES 1 and 2. All licensed operators either performed or witnessed these tests. In all cases it was very apparent to the operator that the control rod cluster was misaligned to its bank.

Power was escalated to 50% thermal power on January 21, 1970, with subsequent testing at 50% thermal power being completed in about 1 week. Because of condensate pump failure, operation at 50% thermal power was extended until February 5, 1970. Sixty percent thermal power was reached on February 5, 1970, and on February 12, 1970, 75% thermal power was achieved. From February 12 until March 11, 1970, due to condensate pump and heater drain pump problems, the unit was operated between 50% and 75% thermal power. The unit was removed from service for a period of 39.50 hr beginning on March 2, 1970, for the purpose of repacking the pressurizer spray valves.

The unit was returned to 75% thermal power on March 3, 1970. Testing at 75% thermal power was completed in approximately 4 days at that time. The heater drain pump repairs were completed and the unit was raised to 100% thermal power on March 11, 1970. The 100-hr 100% power acceptance test run began at 2200 on March 11 and ended at 0200 on March 16, 1970.

After the acceptance test run, 10% and 50% load reduction tests were completed without incident.

The unit was shut down on March 30, 1970, for scheduled maintenance to prepare the unit for commercial operation. The unit was declared to be in commercial operation on July 1, 1970.

14.5.2 1520 MEGAWATTS THERMAL POWER TEST PROGRAM

The power escalation test program from 1300 to 1520 MWt was performed from March 8 to April 14, 1972. See Sections 14.1.2 and 14.6.2.

14.6 INDIVIDUAL TEST DESCRIPTIONS

14.6.1 INITIAL STARTUP AND POWER TEST PROGRAM

The following is a summary of the initial startup testing program for the R. E. Ginna Nuclear Power Plant.

14.6.1.1 Safety Injection Systems Preoperational Tests

14.6.1.1.1 Safety Injection Test

The purposes of this test were as follows:

A. Verify the operation of the steam line isolation sequence.

B. Verify that the proper sequencing and operation of valves, circuit breakers, and diesel generators occurs on initiation of safety injection and containment spray signals.

C. Verify the operation of indicating and status lights of the above-mentioned equipment.

D. Verify that the proper safeguards equipment is operated with each of the two logic trains.

E. Initiate a safeguards signal by simulating an abnormal condition to transmitters related to safeguard systems.

F. Verify the operation of the motor-driven auxiliary feed pumps by simulating a low-low steam generator signal.

G. Verify the operation of the steam-driven auxiliary feed pump by simulating a low-low steam generator level signal in both steam generators.

The operation of the steam line isolation sequence and the operation of the motor-driven feedwater pumps and the steam-driven feedwater pumps by their respective safety signals from both the A and B logic trains were performed satisfactorily.

The proper operation of the diesel generators was witnessed for both logic trains.

Proper sequencing and operation of safeguard valves and the proper operation of their respective indicating and status lights upon receiving a safety injection and/or containment spray signal were satisfactorily performed with the results tabulated as in Figure 14.6-1 which typifies data sheets used.

The following circuit breakers were tested coincidentally with the safety injection valves and initiated by the same safety injection signal as that which actuated the safeguard valves.

BREAKER	COMPONENT	BUSES
52/BT 17-18	(Tie breaker between buses 17 and 18)	---
52/BT 16-15	(Tie breaker between buses 15 and 16)	---
52/BT 14-13	(Tie breaker between buses 14 and 13)	---
52/FP	Fire pump	17

BREAKER	COMPONENT	BUSES
52/IHIA	Intake heaters	18
52/IHIB	Intake heaters	17
52/IHIC	Intake heaters	18
52/IHID	Intake heaters	17
52/MCC 1G1	Screen house motor control center	18
52/MCC 1G2	Screen house motor control center	17
52/BT 16-14	(Tie breaker between buses 14 and 16)	---
52/CCP 1A	Component cooling pump 1A	14
52/CCP 1B	Component cooling pump 1B	16
52/CHP 1A	Charging pump 1A	14
52/CHP 1B	Charging pump 1B	16
52/CHP 1C	Charging pump 1C	16
52/PHBG	Pressurizer heater backup group	16
52/PHCG	Pressurizer heater control group	14

As with the valves, the breakers listed in this step were tested one logic train at a time to ensure that each train functioned properly.

In order to ensure that each breaker worked properly, it was necessary to initiate the safety injection signal a number of times. The procedure and results are combined in the following test summaries.

Safety injection was initiated in tests 1 through 6 by turning the block-unblock switch on the main circuit breaker to the unblock position.

Test 1

This was a test of the A train logic so the power was removed from the B train logic by turning off circuit 9 of dc control panel 1B.

Tie breaker 52/BT 17-18 was closed so that bus 17 was fed from bus 18. Tie breaker 52/BT 16-15 was closed so that bus 15 was fed from bus 16. Tie breaker 52/BT 14-13 was closed so that bus 13 was fed from bus 14. These three breakers then tripped on safeguard initiation as required.

Breakers 52/FP1A, 52/FP1B, reactor trip breaker A, reactor trip breaker B, and the fire pump breaker were closed but then tripped on a safety injection signal.

The intake heater breakers and 52/MCC1G1 were closed. Breakers 52/IH1A, 52/IH1C, and 52/MCC1G1 tripped because of safety injection, and 52/IH1B and 52/IH1D tripped because

of undervoltage on bus 17. This was determined by closely watching the breaker indicators on the breakers themselves. By close coordination with the control board, it was determined that 52/IH1A, 52/IH1C, and 52/MCC1G1 tripped immediately upon safety injection initiation. Thus, it was assumed that these three trips were caused by safety injection signal.

Breakers 52/IH1B and 52/IH1D are fed from bus 17; thus, in this case these two heaters tripped on undervoltage a second or so after the previous three breakers tripped. Therefore, test 1 was performed satisfactorily.

Test 2

This was also a test of logic train A, but tested the tripping of other breakers. Power was not restored to logic train B. Breaker 52/BT 17-18 was closed so that bus 18 was fed from bus 17. Breaker 52/BT 16-14 was closed so that bus 16 was fed from bus 14. Both of these breakers tripped on safeguard initiation as required.

Breakers 52/FP, 52/IH1B, 52/IH1D, 52/CCP1B, and 52/MCC1G2 were closed prior to initiating safety injection and remained closed after initiation of that signal. This is because these breakers are shunt-tripped from sequence logic train B, not logic train A. Breaker 52/CCP1A was closed and did trip on the safety injection signal. This was caused because of the safety injection signal combined with the tripping of bus 14. Bus 14 tripped because the safety injection signal combined with the undervoltage on bus 18. This undervoltage on bus 18 was caused by safety injection because bus 18 was being supplied with power from bus 17 via 52/ BT 17-18; 52/IH1A and 52/LH1C were closed. Through close coordination between the control board and the main at the circuit breakers, it was determined that these two breakers tripped immediately upon initiating safety injection and not a second later. This indicates that these breakers tripped because of the safety injection signal and not because of the undervoltage condition on bus 18.

Breakers 52/CHP1A and 52/PHCG were closed and did trip when safety injection was initiated. It was later realized that these trips could have been caused by either the safety injection signal or the undervoltage signal that occurred on bus 14. These were tripped by pressing the buttons on SI-11X and SI-12X. Both breakers tripped. Therefore, test 2 was performed satisfactorily.

Test 3

This was a continuation of the testing of logic train A. Power, therefore, was not restored to train B logic. Breaker 52/BT 16-14 was closed so that bus 14 was fed from bus 16. On initiation of safety injection, this breaker tripped. Breakers 52/CCP1A and 52/CCP1B were both closed. On initiating safety injection, 52/CCP1A tripped and 52/CCP1B did not. Test 3 was performed satisfactorily.

Test 4

This was a test of the B logic train so power was removed from the A train logic by turning off circuit 12 of dc control board panel 1A.

Tie breaker 52/BT 17-18 was closed so that bus 18 was fed from bus 17. Tie breaker 52/BT 16-15 was closed so that bus 15 was fed from bus 16. Tie breaker 52/BT 14-13 was closed so that bus 13 was fed from bus 14. These three breakers tripped on safeguard initiation.

Breakers 52/FP1A, 52/FP1B, reactor trip breaker A, reactor trip breaker B, and the fire pump breaker were closed and did trip on a safety injection signal.

The intake heater breakers and 52/MCC1G2 were closed. Breakers 52/IH1B, 52/IH1D, and 52/MCC1G2 tripped because of safety injection and 52/IH1A and 52/IH1C tripped because of undervoltage on bus 18. This was determined by closely watching the breaker indicators on the breakers themselves. By close coordination with the control board, it was determined the 52/IH1B, 52/IH1D, and 52/MCC1G2 tripped immediately upon safety injection initiation. It was assumed that these three trips were caused by the safety injection signal.

Breakers 52/IH1A and 52/IH1C are fed from bus 18; thus, in this case these two heaters tripped on undervoltage a second or so after the previous three breakers tripped. Test 4 was performed satisfactorily.

Test 5

This was also a test of logic train B, but tested the tripping of other breakers. Power was not restored to logic train A. Tie breaker 52/BT 17-18 was closed so that bus 17 was fed from bus 18. Tie breaker 52/BT 16-14 was closed so that bus 14 was fed from bus 16. Both of these breakers tripped on safeguard initiation.

Breakers 52/IH1A, 52/IH1C, 52/MCC1G1, and 52/CCP1A were closed prior to initiating safety injection and remained closed after the initiation of that signal. This is because these breakers are tripped from logic train A, not logic train B.

Breaker 52/CCP1B was closed and did trip on the safety injection signal. This was caused because of the safety injection signal combined with the tripping of bus 16. Bus 16 tripped because the safety injection signal combined with the undervoltage on bus 17. The undervoltage on bus 17 was caused by safety injection because bus 17 was being supplied with power from bus 18 via 52/BT 17-18. Breakers 52/IH1B and 52/IH1D were closed. Through close coordination between the control board and the operator at the circuit breakers, it was found that these two breakers tripped immediately upon initiating safety injection and not a second later. This indicates that these breakers tripped because of the safety injection signal and not because of the undervoltage condition that occurred on bus 17. Breakers 52/CHP1B, 52/ CHP1C, and 52/PHBG were closed and did trip when safety injection was initiated. It was later realized that these trips could have been caused by either the safety injection signal or the undervoltage signal that occurred on bus 16. Breakers 52/CHP1B and 52/CHP1C were closed and then tripped by pressing the button on relay SI-21X. This worked properly.
Breaker 52/PHBG was tripped by placing a jumper across contacts 19 and 23 of relay SI-22X. Test 5 was therefore performed satisfactorily.

Test 6

This was a continuation of the testing of logic train B. Power, therefore, was not restored to train A logic. Tie breaker 52/BT 16-14 was closed so that bus 16 was fed from bus 14. On

initiation of safety injection, this breaker tripped. Breakers 52/CCP1A and 52/CCP1B were both closed. On initiating safety injection, 52/CCP1B tripped and 52/CCP1A did not. Test 6 was performed satisfactorily.

Test 7

This test ensures that reactor trip breakers function on a manual safety injection signal in the A logic train. Test 1, on the other hand, tests the A logic train on automatic safety injection signals.

Reactor trip breakers A and B were closed and the B logic train was shut off. The manual safety injection button was pushed and both breakers tripped. Logic train B was restored to operation.

Test 8

This test ensures that reactor trip breakers function on a manual safety injection signal in the B logic train. Test 4 on the other hand tests the B logic train on automatic safety injection signals.

Reactor trip breakers A and B were closed and the A logic train was shut off. The manual safety injection button was pushed and both breakers tripped. Logic train A was restored to operation.

14.6.1.1.2 Accumulator Blowdown Test

This test had basically three goals:

A. Determine the magnitude of pipe displacement and stress resulting from reaction to the fluid blowdown.

B. Determine the amount of water forced back through the reactor coolant pump into the low portion of piping between the steam generator and pump suction.

C. Measure the blowdown transient for comparison with the analysis reported in Chapter 14 of the original FSAR.

Four test runs were made, each accumulator being subjected to both a run at 300 psig initial pressurization and 740 psig initial pressurization.

Strain and displacement readings were taken on each accumulator discharge line; measurements of the pressure-time transient were made and the volume of water collected in the loop seal region was measured for each run.

Pipe Reaction Results

The pipe displacements and stresses were measured under the supervision of Gilbert Associates and Brewer Engineering Laboratories. Pipe reactions were not excessive and are given quantitively in the Brewer report, "Accumulator Piping Vibration Test Results."

Investigation of Water Blowback Through the Reactor Coolant Pump

During the design of Ginna Station, the dynamics of the water jet entering the reactor coolant pipe were analyzed due to a concern that the accumulator flow might divide and flow back through the pump, losing water intended for the reactor vessel.

There are two features of the reactor coolant pump which resist such backward flow. One is the diffuser assembly which forms a dam to flow within an inch or two of the top of the reactor coolant pipe at pump discharge. The other is the pumping action of the reactor coolant pump itself which, even while coasting down, strongly rejects water attempting to flow in reverse through the pump.

The analysis showed that the discharge of water into the reactor coolant pipe from the accumulator would cause the water level to rise above the diffuser assembly into the impeller.

The opinion of hydrodynamics consultant Dr. V. L. Streeter of the University of Michigan was that the configuration of the pump is such as to dissipate jet effects, requiring the spinning impeller only to prevent reverse flow against a foot or two of water head.

An analysis of the length of the pump deceleration transient under loss-of-coolant conditions and the pumping effect of the reactor coolant pump with a voided suction showed that the pump will provide the necessary pumping effect to prevent reverse flow for the period required to ensure effective delivery of accumulator water to the core.

This test showed that water did rise above the diffuser assembly.

Blowdown Transient Behavior

The blowdown flow transients showed well-behaved, predictable transients for three of the four runs. One anomalous run occurred in the case of the high pressure blowdown of the loop B accumulator, resulting in further investigation and analysis.

In spite of the bad run, the basic goals of the program were met.

A. The runs showed that the assumption of an adiabatic gas expansion used in the original FSAR loss-of-coolant analysis was valid.

B. The runs showed that, for both accumulators, the piping resistance is about two-thirds of the value used in the original accident analysis providing a flow margin about 15% over the flow initially calculated for the original FSAR accident analysis.

C. The data were good enough to show a correlation between the high and low pressure runs on the loop A accumulator such that the discharge pipe resistance factors calculated from each run agreed within about 20%, the major part of which is probably due to errors in reading the recorder charts.

D. Between the two low pressure runs (loop A versus loop B) the variations in calculated resistance factor was about as predicted by the piping resistance calculations used as input to the accident analysis in the original FSAR.

Table 14.6-1 shows numerical results in support of the above conclusion.

The long blowdown of the loop B high pressure run was cause for concern since it appeared that a substantial resistance, about three times normal, had suddenly been introduced in the line.

A review of piping resistance calculations and layout drawings was made without uncovering any reason for the long blowdown during that specific run.

Items investigated were as follows:

1. Possibility: The initial gas pressure was low.

   This was ruled out since two pressure indicators showed 720 psi before start of the test and because the quantity of N_2 was metered into the tank and observed to be the same as a subsequent loop A high-pressure run.

2. Possibility: The chart speed was inadvertently increased.

   Ruled out due to corresponding times of transient between chart and stop watch.

3. Possibility: An obstruction in the line.

   Ruled out due to disassembly of valves, observations by boroscope, and by swabbing of the line segments which could not be observed.

4. Possibility: Stuck check valve.

   Ruled out due to obvious free movement and seat tightness of valves when observed on disassembly. If the valves had been subjected to nearly 700 psi differential, some evidence in valve damage might have been noted.

5. Possibility: Jammed motor-operated valve.

   Inspection showed no sign of mechanical damage or of loss of freedom of movement. No foreign matter was in the valve.

With the rest of the loop B accumulator flow path shown clean and free flowing and with one low-pressure run which corresponded well with the line resistance and performance of the loop A unit, attention was turned to the possibility that the motor-operated valve started, but did not complete its stroke. This appears to be the most likely reason for the anomalous run at high pressure on loop B.

AA. The loop A and loop B runs are practically identical up to about 4 sec into the transient when loop B data show establishment of a constant resistance (about 1275 L/D). From that point until the termination of blowdown, the pressure transient is predictable for an unchanging resistance.

BB. The mark automatically put on the chart with the contacting of the valve open-limit switch is missing from only this run.

CC. There was no direct observation that the valve did open, although it was operationally tested over its full stroke after the test without any difference in pressure across the disk.

DD. Two operators have testified that the monitor lights did not change to indicate picking up the full-open-limit switch on this run. The weight of evidence which we collected led to the following judgments:

1. The accumulator lines were clean and should function with a fully open isolation valve in a manner that is entirely consistent with the commitments made in the original FSAR.
2. The motor-operated isolation valve started but did not complete its stroke.

Since this motor-operated valve is normally open and is not required to function during an accident, it should not be considered as in impediment to the safety of the plant. Test recordings are on file at Ginna Station.

14.6.1.1.3 Safety Injection Flow Test

The purposes of this test were as follows:

A. Verify the safety injection pumps shutoff head.
B. Verify the safety injection pumps pressure and flow characteristics.
C. Verify the residual heat removal pumps pressure and flow characteristics to the reactor coolant system.
D. Demonstrate the residual heat removal pumps recirculation to all three safety injection pumps.
E. Demonstrate the residual heat removal pump A recirculation to the safety injection pump C.

The shutoff heads of the safety injection pumps and the residual heat removal pumps are 1520 psi and 141 psi, respectively. The design pressure and flow are 1080 psi at a flow of 300 gpm for the safety injection pumps and 121 psi at a flow of 1560 gpm for the residual heat removal pumps. The test demonstrated that the design flow characteristics of all five pumps were realistic and proven or exceeded in practice. Trace recordings of pressure and flow were made during the test runs and are on file.

The ability of the two residual heat removal pumps to deliver to the three safety injection pumps was demonstrated and the various flows and pressures of interest were recorded. The ability of the residual heat removal pump A to deliver to the safety injection pump C was satisfactorily demonstrated with the flows and pressures of interest recorded.

The capability of the safety injection system was reevaluated after tests at Point Beach Unit 1 and H. B. Robinson indicated that pipe resistance was greater than predicted. The system was modified, including the opening of all four injection line isolation valves, the installation of a second check valve in each injection branch line, and the installation of a manual globe valve in the branch line to the cold leg of loop A. It was concluded from the reevaluation that the as-built modified safety injection system met the design objectives for safe operation at 1300 MWt and at the higher power level of 1520 MWt. On March 31, 1971, a safety injection flow capability test was performed with all fuel elements removed from the core, the reactor vessel head removed, and the refueling cavity flooded to near the normal MODE 6 (Refueling) level.

The purpose of the test was threefold: (1) demonstrate the runout capability of the safety injection pumps, (2) verify the ratio between flows in the individual branch lines, and (3) demonstrate the effect of the pump miniflow on the system delivery capability.

The evaluation of the test data led to the following conclusions regarding the as-built performance of the safety injection system after the modifications:

AA. The injection capability of the system as determined by using pipe resistance derived from the test is approximately the same as used in the core shutdown and cooling analyses.

BB. The allowable runout flow of the pumps is greater than was assumed in the core analyses, i.e., higher core injection flow is available at low reactor pressures such as in the case of large pipe break accidents.

CC. The flow balance between branch lines is excellent; the addition of the manual globe valve in the loop A, cold-leg branch improved the system performance.

DD. The miniflow for each pump was determined to be very near the design value of 30 gpm at the shutoff head of the pump.

14.6.1.1.4 Containment Spray System Test

The shutoff head of the containment spray pumps was tested and found to be higher than the shutoff head on the design curve. The original FSAR, Table 6.4-1, lists the design head and flow of the spray pumps as being 435 ft and 1615 gpm. Since there is no way to test design flow and head without flooding the containment building, pump performance was evaluated by comparing the pump flow and head at recirculation flow (45 gpm) to the design head curve. This tested satisfactorily. The valve operation and sequencing of this system was tested satisfactorily in the safety injection functional test. The remainder of the piping from the last valve to the nozzles in the spray ring headers was tested by charging the piping with compressed air and suspending a helium-filled balloon with tell tails in front of each nozzle. Each nozzle opening was proven free and clear.

14.6.1.1.5 Residual Heat Removal System Test

The purpose of the residual heat removal test was to verify that the system components were capable of meeting their design requirements and that the system interlocks and interlocks to other systems operated as intended. The capability of the residual heat removal pumps to meet design requirements was successfully demonstrated in the safety injection flow test, Section 14.6.1.1.3. A functional test of the interlocks involved in the residual heat removal system was performed as outlined and all interlocks operated as required.

Testing of the residual heat exchangers to ensure that heat exchanging capabilities met specifications was performed during a cool-down period. Test results demonstrated that the specifications for these exchangers were conservative. To ensure that the recirculation phase of safety injection could be performed, a recirculation functional test was written and successfully completed demonstrating valve operation and flow from sump B through the residual heat removal pumps.

14.6.1.1.6 Safeguards Systems Operational Test

The intention of this test procedure was to ensure that all safeguards systems were operationally checked out before criticality. This checkout involved a test of individual

channel triping followed by logic trains A and B tripping where applicable. Safeguards systems valves and motors were not actuated for this test since actuation of these components had been performed in other tests, but rather the actuating devices of the components such as relays, controllers, etc., were monitored for operation. Verification of proper operation of alarms and indicating lights was a part of this test procedure. The following is a list of the safeguards systems that underwent the operational checkout in this test:

A. Steam line isolation.

B. Safety injection and initiation of the following safeguard action subsequent to initiation of safety injection:
   - Feedwater system isolation.
   - Reactor trip.
   - Emergency diesel starting.
   - Auxiliary feedwater pump starting.
   - Fan cooling starting.
   - Service water pump starting and system isolation.

C. Containment spray.

D. Containment isolation.

E. Containment ventilation isolation.

14.6.1.1.7 Emergency Diesel Generator Test

This test was performed to verify that the two diesels and auxiliary equipment will perform their designed functions when required to do so.

The first part of the test was concerned with the capacity of the air storage tanks and their ability to crank the diesels for 45 sec. Although they were incapable at first, one additional air storage tank was added to each diesel starting air supply thus doubling the air-storage capacity. A second test was performed which proved that the air starting system was capable of cranking the diesels for 45 sec. It was also necessary to test the starting signals for both diesels. Where signals can come from redundant sources these sources were checked individually. All start signals performed according to design.

The undervoltage relay circuits are designed to clear a bus of all large loads except the motor control centers if there is an undervoltage condition on that bus. Thus, the diesel will not be connected to a fully loaded bus, which would probably trip the diesel on overcurrent because of the high starting current required. Each redundant undervoltage relay circuit was tested for the 480-V safeguard buses and performed to clear the bus involved and also to start the appropriate diesel.

The diesel was tested to ensure that it was capable of starting and that the control circuitry could place it on line in 10 sec. Load tests were run and showed that the diesels were capable of their rated capacity and could satisfactorily carry their safeguard load during steady-state and safeguard sequencing load pickup.

During the test, the sequencing relays for safeguard equipment starting were set and would start equipment within 3 sec of the design times. Safeguard valves not covered by the safety injection test were tested to ensure that they would close or open as required on safeguard initiation. In addition, control and alarm and tripping circuitry was tested to ensure that these functions were properly performed.

The interlocks on the following breakers were also tested to ensure proper operation: 52/EG1A1, 52/EG1A2, 52/EG1B1, 52/EG1B2, 52/14, 52/16, 52/17, 52/18, 52/BT16-14, and 52/BT17-18. Because of this testing, one design change and several wiring modifications were made. Once the changes were made, the tests on these breakers were completed satisfactorily.

14.6.1.1.8 Direct Current Test

Each battery system was tested in two basic ways. First, the battery charger voltage outputs for varying loads and varying charger input voltages were tested. It was found that although the 480-V ac input voltage was varied by 10%, the dc output voltage of the chargers did not vary by more than 1% under varying load conditions.

The second basic test was designed to show that the battery itself could sustain a discharge rate of 131 amps for 8 hours while not lowering the output voltage below 105 V. Although neither battery passed in the first test, each passed after a number of cells in each battery were replaced.

14.6.1.2 Preoperational Instrumentation and Control Tests

14.6.1.2.1 Reactor Coolant System Pressure Comparison Test

The purpose of this test was to verify the calibration of the primary coolant system pressure instrumentation at various actual system pressures. The test was performed while heating up the system to no-load temperature and pressure conditions. At various pressure levels the pressure instrumentation of the reactor coolant system was checked against the reading of a deadweight tester nulled across a differential pressure cell to the actual system pressure. This test was completed successfully on June 28, 1969.

14.6.1.2.2 Resistance Temperature Detector Cross Calibration Test

This test procedure was used to determine isothermal corrections for reactor coolant resistance temperature detectors (RTD) and in-core thermocouples. The reactor coolant temperature was maintained at a constant shutdown temperature of 545°F. Resistance measurements of the 10 RTDs of the reactor coolant loop A were taken three different times with a precision ohmmeter and averaged. The temperature of each RTD was then calculated. The same procedure was followed for determining the temperature of loop B. Averaging the temperatures of the RTDs in loop A resulted in a temperature of 545.5°F and that of loop B also resulted in a temperature of 545.5°F.

In-core thermocouple maps were obtained by computer printout while the RTD measurements were being taken with good agreement between the printouts and RTD measurements.

14.6.1.2.3 Steam Generator Manual Control and Level Instrumentation Test

In essence, this test was a functional test of the steam generators, condensate system, feedwater system, auxiliary feedwater system, and the instrumentation of these systems.

Analog simulators were used to inject signals into steam generator level channels. These signals were varied to allow verification of bistable setpoints and calibration of the level indicators. The functions that were verified and their respective setpoints are as follows:

A. Low-low water level single channel alert - 15%.

B. Low-low water level reactor trip - 15%.

C. Steam generator level setpoint deviation - ±5%.

D. Steam generator high-level loop A/B channel alert alarm - 68%.

E. Steam generator high-level alarm - 68%.

F. Feedwater valves close - 68%.

G. Feedwater bypass valves open - 10%.

Steam flow and feedwater flow indicators were calibrated by simulating signals to the indicators. The steam flow-feedwater flow mismatch circuits were adjusted to give low feedwater flow single-channel alert alarm and reactor trip at a 0.7×10^6 lb/hr of steam flow in excess of feedwater flow deviation. Steam generator high feedwater flow alarms were set for a deviation of 0.7×10^6 lb/hr of feedwater flow in excess of steam flow.

Pressure signals were simulated to the steam generator pressure channels to calibrate the pressure indicators and set the pressure-related bistables. The low steam pressure loop A/B alarms and channel status lights were set for 600 psig. Steam line low-low pressure loop A/B channel alert was set for 500 psig. The turbine first stage pressure indications and alarm checkout was performed by simulating a pressure signal. The channel status trip setpoint was set at 45.5 psig for the two turbine first stage pressure channels.

The test required the stroking of all valves in the condensate, feedwater, and auxiliary feedwater systems with final position of the valves in the normal operating position. The condenstate and feedwater pumps were started and flow measurements versus feedwater bypass valve position were taken.

The automatic start of the preferred auxiliary feedwater pumps was verified by tripping the main feedwater pumps.

14.6.1.2.4 Rod Position Indication System Test

Verification of the satisfactory performance of the rod position indicating system for each control rod and each control rod bank under MODE 3 (Hot Shutdown) conditions was demonstrated in this test. Voltage readings were taken and recorded at the output of each linear voltage differential transducer at various intervals of rod travel for each rod. Associated alarms were verified and the bank overlap of each bank was set at 195 steps for rod withdrawal and 35 steps for rod insertion.

14.6.1.2.5 Rod Stepping Test

This test was designed to verify that the rod control systems satisfactorily perform the required stepping operations for each individual rod under both hot and cold conditions. Each rod was fully withdrawn and fully inserted while recordings were made of current flows to the various rod drive mechanism coils. These recordings are on file at Ginna Station.

14.6.1.2.6 Rod Cluster Control Assembly Drop Time and Partial Length Rods Operational Tests

The purpose of this test was to determine the drop time of each full-length rod cluster control assembly under a number of reactor coolant system operating conditions. The data sheets following are samples of the data sheets used for this test noting operating condition of the system and the rod drop times. Originally, the Ginna Technical Specifications gave a maximum rod drop time of 2.7 sec based on earlier PWR design and experience. These specifications were modified to take into consideration the longer control rods of the Ginna Plant. The specifications were changed to read that the maximum elapsed time to the dash pot shall not exceed 1.8 sec and shall not exceed 5 sec to bottom out.

A second purpose of this test was to functionally check the partial length control rod drive system to determine proper indication of rod position and the operational characteristics of the system when the 440-V power is interrupted. The results of the test are typified in the data sheets of Figures 14.6-2 and 14.6-3.

14.6.1.2.7 In-Core Thermocouples Test

The purpose of this test was to provide a functional checkout and demonstration of the in-core thermocouple and readout system at MODE 3 (Hot Shutdown) conditions. The reactor coolant system was maintained at a constant temperature of 549°F for the duration of this test. Analog readings were taken and recorded for each of the in-core thermocouples. A computer readout was also obtained for each of the in-core thermocouples. The reactor coolant system RTD readings were taken at the time and compared to the analog and computer in-core readings. The results of this test were satisfactory.

14.6.1.2.8 Movable In-Core Detector System Test

This test provided a functional demonstration of the in-core flux mapping system.

Each of the four detectors was operated simultaneously and then separately in all possible modes of scan. Limit switches were set, associated alarms were verified, scan rates were set, and position readouts and indicating lights were verified. The test results were satisfactory with only minor discrepancies which have since been corrected.

14.6.1.2.9 Reactor Makeup Blender and Boric Acid Transfer Pumps Operational Test

The purposes of this test were as follows:

A. Obtain information on the operational characteristics of the reactor makeup blender in the "automatic makeup," "borate," and "dilute" modes of operation.

B. Provide a measure of the mixing characteristics of the reactor coolant system.

C. Determine the temperature rise in the boric acid storage tanks caused by the energy input into the system from the boric acid transfer pumps operating continuously in the recirculation mode.

Various amounts of reactor makeup water were dialed into the Veeter-Root integrater at different times and the blender system was energized. When the amount of reactor makeup integrated equaled the amount dialed, the "blend" system deenergized automatically. The amount of water delivered to the reactor coolant system was measured by calculating the displacement in the volume control tank. This was compared to the amount that had been set for and the amount integrated by the Veeter-Root integrater.

The "borate" mode was checked in the same manner with the exception that the boric acid was collected in calibrated containers at a sample point.

The flow rates of the "borate" and "dilute" modes were confirmed to agree with the rates set by the controller.

The "automatic makeup" mode of operation was checked for performance by injecting different concentrations of boric acid blend into the reactor coolant system and sampling the coolant at various time intervals at different sample points of the reactor coolant system.

Recirculating the boric acid storage tanks with the boric acid transfer pumps raised the temperature of the No. 1 tank 14°F in 8 hours 55 min and the No. 2 tank 15°F in 7 hours 55 min. The No. 1 tank cooled 7.50°F in 22 hours and the No. 2 tank cooled 9°F in 22 hours. The test was completed with satisfactory results.

14.6.1.2.10 Pressurizer Level Control Test

The objective of the pressurizer level control test procedure was to verify that the pressurizer level control system setpoints were properly set and that the control system functioned properly.[a]

With the reactor coolant system at the no-load temperature (547°F), and pressure (2235 psig) condition, and with all pressurizer controls in the automatic mode of operation, all pressurizer level indicators were checked and level indications recorded. Proper operation of channels was verified in this manner.

Preliminary values for the proportional band, rate time constant, and reset time constant for pressurizer level controller LC-428 F were given in the RG&E Reactor Control and Protection System Precautions, Limitations, and Setpoints Operating Instruction, P-1. These preliminary values were used during the initial checkout and calibration of the controller.

To determine how well the control system responded to system level and average T_{avg} variations, first the manual control setpoint on TC-401C (remote setpoint controller for LC-

a. The pressurizer level program setpoint is a function of T_{AVG} and varies from 19.5% at 547°F to 49% at 570°F for operation at 1300 MWt.

428C) was varied from 547°F to 570°F. This simulates a rise in average T_{AVG}. A rise in the pressurizer level followed. The level was lowered to the original value by reversing the above procedure. The levels on all level channels were recorded.

The level control system of the pressurizer was next checked by increasing the level of the pressurizer by manually controlling the charging line flow control valve HCV-142 and the charging pumps. As the pressurizer level increased it was verified that the pressurizer high-level alarm occurred at 70%, pressurizer heaters were energized at 70%, and pressurizer high-level reactor trip partial matrix alarm occurred at 92%.

The level was reduced in the pressurizer by the manual method described above and it was verified that pressurizer low-level alarm occurred at 5% below the level program setpoint, the low-level alarm, letdown isolation (LCV-427 closed) and heaters turned off at 11%, and a safety injection partial matrix alarm occurred at a level of 5%.

This test was successfully completed after the second attempt.

14.6.1.2.11 Pressurizer Pressure Control Test

The objective of this test was to first check the response, stability, and general control characteristics of the pressurizer pressure control system and make any adjustments to the controllers required to obtain proper operation, and secondly, to verify that all the various alarms and control setpoints are properly set and function as required.

With the reactor coolant system at the no-load temperature, and pressure conditions and pressurizer pressure controls on automatic, the proper operation of all pressurizer pressure indicators and recorders was verified. The control board pressure control controller was placed in the manual position and its signal varied to verify the following setpoints:

- Proportional heaters full off at 2250 psig.
- Proportional sprays begin at 2260 psig.
- High pressure alarm at 2310 psig.
- Relief valve PCV-431C opens at 2330 psig.

The reactor coolant system pressure was actually changed to verify the following:

- Relief valve PCV-430 opened at 2335 psig.
- High pressure reactor trip at 2400 psig.

With pressurizer heaters on automatic, the reactor coolant pressure was reduced by manually controlling spray water to the pressurizer. The following setpoints were verified during this mode of operation:

- Proportional heaters full on at 2220 psig.
- Backup heaters on at 2210 psig.
- Pressurizer Power Operated Relief Valve (PORV) interlock functioned at 2185 psig.

- Pressurizer low pressure alarm at 2185 psig.
- Safety injection could be manually blocked below 2000 psig.
- Low pressure reactor trip partial matrix alarm at 1720 psig.
- Safety injection partial matrix alarm at 1715 psig.

The pressurizer heaters had been turned off after verifying their operation points. They were then turned back on and put in the automatic mode after completing the pressure decrease portion of the test. This resulted in a gradual pressure increase allowing the following setpoints to be verified:

- Low pressure reactor trip partial matrix alarm cleared at 1725 psig.
- Safety injection unblocked at 1990 psig.
- Pressurizer Power Operated Relief Valve (PORV) interlock functioned at 2190 psig.

14.6.1.2.12 Steam Dump Test

The purpose of this test was to optimize the settings of the steam dump controller and to functionally test the system.

Some portions of this test could not be performed since steam dump to the condenser could not be sustained for any period of time without nuclear heat. Those portions omitted in this test were performed in the operational transient tests described in Section 14.6.1.7.

A simulated signal was fed into the steam bypass controller. This signal was varied until the turbine trip interlock was satisfied and the steam dump to the condenser valves opened. The test signal was decreased until the turbine trip interlock cleared and the steam dump valves to the condenser modulated close. Four of the eight condenser valves were set to open with the simulated signal for T_{avg} at 8°F, and the other four were set to open at 16°F. This procedure was repeated for the other condenser steam bypass controller with the exception that four of the valves were set to open at 12°F and the remaining four at 20°F.

The atmospheric steam dump system was functionally tested at this time for controller response and secondary system pressure control capabilities.

This system tested satisfactorily.

14.6.1.2.13 Radiation Monitoring System Operational Test

The purpose of this test was to provide an operational test of the complete radiation monitoring system to ensure that it would perform all the functions that are required of the system.

Figure 14.6-4 includes typical data sheets of the test results of one of the radiation monitoring system channels. The data explains the text objectives. The results of this test were acceptable.

14.6.1.2.14 Reactor Coolant System Flow Measurement Test

The purpose of this procedure was to provide a means of obtaining the necessary data to interrelate pump input power, elbow tap pressure, and steam generator delta P as an accurate measurement of flow rate. A description of the methods used to interrelate these parameters is contained in Section 4.2 of the original FSAR.

Having completed this test, the preliminary data analysis was completed and indicated the reactor coolant system flow rate for loop A to be 95,200 gpm or 106% of design flow, and for loop B to be 94,000 gpm or 104.8% of design flow.

14.6.1.2.15 Nuclear Instrumentation Test

This test provided a functional demonstration of the nuclear instrumentation system. Each of the 12 drawers (one for each nuclear instrumentation channel) was functionally operated and calibrated by simulating a detector signal to the first element after the detector in a channel. All trips and permissive signal setpoints generated by the nuclear instrumentation system were set, associated alarms were verified, and all remote meters and recorders were checked for proper operation and indication. The test results were satisfactory.

14.6.1.3 Safety and Relief Valve Tests

14.6.1.3.1 Pressurizer Safety Valve Test

This test was performed to verify the proper setting of the pressurizer safety valves.

To perform this test the system pressure was maintained between 1865 psig and 2110 psig. A Crosby Valve Company air-set lifting device was installed on PCV-434 (pressurizer safety valve) bonnet and an air supply attached to the device. By controlling the pressure regulator manually, the pressure was gradually increased to the lifting device. Pressurizer pressure readings and air pressure readings to the lifting device were taken continuously during this procedure. When the pressurizer valve began to open, simmer, or leak audibly, the air pressure on the lifting device was released. A differential pressure was determined from a curve for K2 orifice in Crosby Valve Company Instruction No. T-1652-1 for the air pressure reading of the air supply to the lifting device when the valve first opened. This differential pressure was added to the pressurizer pressure reading at valve opening to determine the actual pressure at which the valve would open. This test was repeated until the valve opened at 2485 psig ±1%. Figure 14.6-5 is a copy of the data sheet for the testing of both pressurizer safety valves.

14.6.1.3.2 Main Steam Safety Valve Test

The setting of the main steam safety valves was accomplished in the same manner as was the setting of the pressurizer safety valves; however, the differential pressure was obtained from the curve for R orifice in Crosby Valve Company Instruction No. T-1652-1 of air pressure versus differential pressure. The set pressures for each of the main steam relief valves are as follows:

V-3508	1141 psig
V-3509	1130 psig
V-3510	1140 psig
V-3511	1140 psig
V-3512	1140 psig
V-3513	1138 psig
V-3514	1085 psig
V-3515	1078 psig

The requirements were to set V-3515 and V-3514 at 1085 psig ±1% and V-3508 through V-3513 to 1140 psig ±1%.

14.6.1.4 Waste Systems Tests

14.6.1.4.1 Liquid Waste Concentration Demonstration Test

The purpose of this test was to demonstrate the proper operation of the two major drumming processes.

A portion of this test demonstrated the process of drumming concentrated waste from the waste evaporator feed tank. This included the operation of the recirculation system from the evaporator feed tank to the dispensing header, and the vacuum operated dispensing valves and vacuum switches. The test also included testing the use of the drums, shields, vacuum pump, and manipulating tools.

The test was successfully run with water and has since been used frequently with waste concentrates with no major problems.

Another portion of this test demonstrated the process of sluicing spent resins from the storage tanks to the drums. The operation of the pressurization and agitation systems along with the instrumentation associated with these systems was functionally checked. The proper operation of the dispensing valves, drums, shields, vacuum pump, and manipulating tools for this mode of operation was demonstrated.

14.6.1.4.2 Waste Disposal System Gaseous Waste Test

This test was a functional test of the waste gas system to ensure that the system could adequately process or vent the gaseous waste emanating from the vent header. All alarms and instrumentation associated with the system were verified for proper operation as were all automatic functions. The waste gas system has since been operated under its intended normal radioactive conditions with satisfactory performance.

14.6.1.4.3 Liquid Waste Processing Test

The purpose of this test was to functionally test portions of the waste disposal system including the waste evaporator, and to demonstrate that the liquid waste disposal system can adequately dispose of the liquid waste products in a safe and reliable manner.

The test was run with satisfactory results with one exception. The decontamination factor across the waste evaporator deteriorated with rated flow of 2 gpm. A decontamination factor of 106 could be maintained with a flow rate of 1.5 gpm.

14.6.1.5 Reactor Coolant System Measurement Tests

14.6.1.5.1 Reactor Vessel Internals Measurement Test

The intent of this test was to obtain experimental data on the reactor vessel internals movements during the startup test program.

The instrumentation installed for the test was as follows:

A. Fourteen maximum displacement indicators on the thermal shield to measure relative motion between the core barrel and thermal shield.

B. Seven accelerometers on the vessel head to detect gross changes in internals response.

C. Thirteen strain gauges to three guide tubes to measure mean deflection and dynamic response imposed by the flow during operation.

Maximum Displacement Indicators

Maximum displacement indicators were designed and installed on the thermal shield at the locations shown in Figure 14.6-6. The measurement of the gap is indicated as DIM. "A" on the sketch following the hot functional test provided an indication of the maximum relative motion between the thermal shield and the core barrel resulting from a combination of thermal differential expansion, hydraulic forces and vibration.

The internal spring-loaded plunger was designed to follow the relative cyclic motion between the thermal shield and core barrel, thus causing the two stationary spring-loaded styluses to leave small markings on the plunger. These marks provided a direct indication of the magnitude of the vibratory motion.

With the exception of two locations, No. 13 at the top and No. 2 at the bottom adjacent to a flexure, the total displacements were relatively small and consistent, and were in close correlation with expected results based on extrapolated data from model testing and from previous measurements on other reactors.

Vibratory motion measured by all the indicators was also small. The maximum motion, as interpreted from the plunger markings, was as follows:

Top	Double Amplitude
Numbers 10, 13, 14	.010 (±.005)
Number 9	.012 (±.006)
Numbers 11, 12	.014 (±.007)

Bottom	**Double Amplitude**
Numbers 6, 7	.008 (±.004)
Numbers 3, 5, 8	.010 (±.005)
Number 1	.014 (±.007)
Numbers 2, 4	.004 (±.002)

If this motion is conservatively assumed to be thermal shield motion only, i.e., the shield motion is ±0.007 in. at the midpoint between supports, the stress corresponding to this motion is very small and within the allowable stress for infinite cycle loading allowed by the ASME Code by an order of several magnitudes.

Several facts are evident from a magnified observation of the scribe markings on the plungers from No. 2 and No. 13 indicators. The scribe marks on No. 13 are located in a position that indicates the plunger was projecting 0.080 in. (Dim. A + 0.080) when the vibratory motion occurred. This indicates that No. 13 indicator was (1) not completely inserted at installation, (2) slipped soon after installation, or (3) a local thermal realignment occurred between the thermal shield and core barrel. The complete lack of marking in the completely inserted position indicates that the large gap is probably the result of either (1) or (2).

Even if the gap occurred as a result of item (3), however, the deflection was well below the calculated allowable of 0.180 in. between supports.

Indicator No. 2 indicated that the vibratory marking occurred with the plunger in the completely compressed position (Dim. A approximately = 0), so that the 0.030 in. gap did not exist during the hot functional test. It is also significant that the spring-loaded plunger had been driven back into the displacement pin and jambed, so that it was not in contact with the core barrel after placement of the internals on the new storage stand. Although not completely conclusive, it appears probable that this gap was influenced by the initial impact with the storage stand. It is located almost directly opposite the initial point of impact.

Based on the very good condition of the internals, the lack of motion between components and the small vibratory motion of the thermal shield, the conclusion was that the Ginna internals as installed are in excellent condition and are adequate for their functional requirements.

Accelerometers

Seven accelerometers were mounted on the outside of the reactor vessel. Their locations and directions of sensitivity are shown in Figure 14.6-7. The accelerometers were mounted on the top of the reactor and clamped to the 4-in.-diameter head penetrations. Those on the bottom of the vessel were attached to the vessel wall by a magnetic clamp.

Signals from the (piezoelectric) accelerometers were amplified with charge amplifiers and recorded on a Visicorder and on magnetic tape. Data was taken when the desired reactor conditions occurred during hot functional and cold hydrostatic testing, during both one and two main coolant pump operation and with main coolant temperatures from approximately 120°F to 560°F.

Part of the intent of these accelerometers was to detect sharp transients or abrupt changes in vessel motion that might result in the event that a significant abnormality occurred in the flow or in internals vibrations during testing. No transients of this type were observed while the data were being taken or during subsequent analysis of the data. Further, no marked changes in the character of the signals were observed at similar reactor conditions during the test period.

Strain Gauges

Four active strain gauges were attached to the upper end of three guide tubes (one dummy gauge was mounted for noise measurement), in order to obtain measurements that indicated the static and dynamic deflections and loads imposed on the guide tubes during cold hydrostatic and hot functional testing.

The maximum mean strain measured was 50.4 pico-in./in. and the overall dynamic strain levels measured were ±11.88 pico-in./in. These measured strains indicate that the guide tubes have an adequate safety margin. Direct verification of adequacy has also been obtained by visual examination after the hot functional test. The strain gauges were left in place in order to observe the core effect on the guide tube and vessel dynamic response.

14.6.1.5.2 Reactor Coolant System Vibration Test

The main function of this test was to verify that the vibrations of the reactor coolant pumps and the reactor coolant system piping and equipment were within acceptable limits during the pump operation. The test also provided reference data for the future operation of the reactor coolant system. The data sheets of Figure 14.6-8 are the results of this test.

14.6.1.5.3 Preoperational Reactor Coolant System Leakage Test

The performance of this test was necessary to satisfy the Technical Specifications that the leakage from the reactor coolant system did not exceed 10 gpm from known sources or 1 gpm from unknown sources.

Prior to running the test, the system was thoroughly inspected for visible signs of leakage.

The reactor coolant system was maintained at constant temperature and pressure for zero power conditions for the 10-hour duration of this test. At the end of the test run, tank and pressurizer levels were compared to levels at test initiation and added or subtracted from the water inventory of the reactor coolant system. Makeup water to the system for the test period was measured. A mass balance of the system was made and total leakage calculated. The results of this test satisfied the Technical Specifications.

14.6.1.5.4 Reactor Coolant System Thermal Expansion Test

The major objectives of this test were to verify that the reactor coolant system could expand unrestrained during the system heatup from the cold condition to operating conditions, and also to establish reference data for the expansion of reactor coolant system components which could be used for future evaluations.

Basepoint measurements were taken at various points around the reactor coolant system with the system in the cold condition. These measurements were compared to measurements taken at the same points under hot conditions. An analysis of the data revealed no restraining problem.

14.6.1.5.5 Flow Coastdown Test

The Ginna flow coastdown test was performed without incident on December 14, 1969. The data was analyzed and found to agree favorably with Figure 14.1.6-1 of the original FSAR. Curves of the reduced information are presented in Figure 14.6-9.

The signals for flow recorded during the tests were in the form of a differential pressure (delta P) measurement. Flow as a fraction of nominal is obtained by taking the square root of the normalized delta P value. Data was taken and reduced for the two-loop total loss of flow and for both single-loop coastdowns, all from full flow.

Figure 14.6-9, Sheets 1 through 3, shows the individual loop coastdown curves determined from the plant data. In order to make a comparison with the design curve, a total core flow was determined by averaging the individual loop flows. These comparisons are found in Figure 14.6-9, Sheets 4 and 5. The time to 50% flow for the two-loop coastdown was predicted at 12.3 sec while the plant was found to take more than 13.5 sec. The one-loop coastdown (Figure 14.6-9, Sheet 5) also shows the safety analysis predicted curve reaching half flow sooner than the actual, but the prediction has the slower flow up to that point.

It was therefore concluded that the plant coastdown rate is consistent and conservative with respect to the FSAR value in order that departure from nucleate boiling be prevented. Although the core flow for the one-loop loss of flow fell faster than predicted, the two-loop coastdown is the limiting case and it is in agreement with the FSAR design value.

14.6.1.5.6 Natural Circulation Test

The Ginna natural circulation test was completed successfully on January 18, 1970. The data was evaluated and found to be in excellent agreement with the predictions reported in the original FSAR, Section 14.1.12. The comparative information is presented in Figure 14.6-10.

A record of nuclear power, coolant average temperature, delta T, and pressure data taken from the control board instrumentation was checked against data trended on the plant computer during the test and found to compare favorably. A primary flow was then calculated, based on the power level (measured by the nuclear instrumentation system), and the other directly measured parameters. The data recorded for the flow calculation are shown in Figure 14.6-10, Sheet 1. Figure 14.6-10, Sheet 2, is a curve of the FSAR prediction with the flow points calculated from the measured data at two power levels of the test.

It is thus apparent that natural circulation does occur and that it is more than adequate for decay heat removal. Furthermore, the flows determined from the plant data are in excellent agreement with the predicted curve.

14.6.1.6 Miscellaneous Safety-Related Tests

14.6.1.6.1 Backfeed from the 115-Kilovolt Grid Test

The purpose of this test was to ensure that power can be fed back from the 115-kV grid through the main and auxiliary transformers to the station auxiliaries, in the event of an extended outage of the station auxiliary transformer as specified in the original FSAR. More specifically, this test was to prove that the 115-kV cable from station 13A to Ginna Station, the 115-kV to 19-kV main generator transformer, the 19-kV bus duct, and 19-kV to 4.33-kV unit auxiliary transformer could be energized from the 115-kV substation, without risk of damage to the transformer or the high voltage lightning arrestors.

The normal station and system safety holding rules were strictly adhered to during the test. To perform this test, it was necessary to disconnect the terminals of the main generator from the isolated phase bus duct by removing the flexible connectors in the lead box under the generator for the test duration. A visicorder for recording voltage transients was installed in the 19-kV potential transformer secondaries.

The normal feed to the station auxiliaries during the then existing plant conditions (MODE 5 (Cold Shutdown)), was from circuit 767 through auxiliary transformer No. 12. This circuit was deenergized and the main transformer was energized from the 115-kV system by energizing circuit 912. The results of the test follow:

Steady-state post switch voltage (no auxiliary load)

A ϕ	68.5 V	0.99/unit
B ϕ	68.5 V	0.99/unit
C ϕ	68.5 V	0.99/unit

Peak maximum transient voltage

A ϕ	1.22/unit
B ϕ	1.06/unit
C ϕ	1.06/unit

Cycles to near steady-state (clean sine wave)

1120 cycles

OCB pole closing angle (breaker 91202[a])	
A ϕ	0.7°
B ϕ	0
C ϕ	2°

a. This is not the only breaker that could be used for this test

There was no evidence of any significant dynamic envelope occurring during this test.

The magnitude of the peak transient voltage was less than expected. This could be due to a combination of two factors.

A. Location of the measuring point on the opposite side of a three-phase wye-delta transformer from the impinging transient. However, even though higher per unit transients would probably have been measured on the 115-kV side of the transformer, indications are, from our test results, that these were not excessive.

B. Extremely small OCB pole closing angles. This could change with the number of circuit breaker operations or the use of a different circuit breaker such as OCB No. 1G1372.

The harmonic disturbances on top of the 60-cycle fundamental lasted for a much longer time period than expected. However, they did not appear to be severe enough to cause any problems. The test was performed on September 21, 1969.

14.6.1.6.2 Blackout Test Without Safety Injection Test

This test was basically concerned with the ability of the diesel generators to supply emergency power to the 480-V buses in the event that normal outside power is lost. This includes the clearing of the buses of loading if outside power is lost so that the diesel breakers will not close to a fully loaded bus.

The loss of power was simulated by simultaneously tripping the 4160-V supply breakers, 52/12A and 52/12B. Since the individual switchgear and 480-V switchgear interlocks were tested previously in the diesel test, no problems occurred.

Because they were not covered in other tests, the logic and opening and closing of the steam supply valves to the steam-driven auxiliary feedwater pump were tested and results were satisfactory.

14.6.1.6.3 Main Steam Isolation Valve Test

It was the intent of this test to demonstrate that the main steam isolation valves function by simulating a high-high containment pressure alarm and that the valves close in the prescribed amount of time.

To prove that the valves would function by simulating a containment high pressure necessitated modifying the test procedure slightly to allow the test to be done in two steps because of

plant status. The relays driven by the containment pressure transmitters were tripped manually to demonstrate steam-line isolation. At a later date, the containment high pressure was simulated at the pressure transmitters to demonstrate that the relays would trip. When this portion of the test was done, the valves were prevented from operating by removing the control air from the valve operator.

The timing of the valves was performed with satisfactory results since specified maximum closing time is 5 sec and the valves actually closed in 1 sec. The maximum opening time of the valves was observed to be 3 sec. The operation of the main steam valve isolation function was included in other tests and all of the logic trains that actuate main steam isolation were verified to perform as intended in the Reactor Trip System (RTS) operational test (Section 14.6.1.6.6).

14.6.1.6.4 Fire Service Water Test

This test was a functional test of the fire protection system intending to verify the design criteria of the booster, and diesel-driven and motor-driven fire pumps, as well as to ensure that all fire detecting devices, alarms, and control functions performed as intended. The test procedure was revised to conform to the updated standards of Nuclear Energy Property Insurance Association (NEPIA). All criteria were met with satisfactory results.

14.6.1.6.5 Electrical System Logic Test

The purpose of this test procedure was to specify the operations necessary to operationally test the following systems:

- Turbine and generator protection.
- Emergency power system logic.
- Rod stop.
- Turbine load reduction.

Where applicable these tests involved a checkout of the analog system followed by logic train A and B checks. Also, where possible, the tests were performed with and without blocking and permissive circuits actuated. The actual tripping of circuit breakers, closing of valves, and starting of diesel generators was not demonstrated in this test, but rather the activating devices, relays, controllers, etc., were monitored with the final action blocked. The performance of this test was carried out over a long period of time which included many retests as is normal for a test of this type. All components and functions of the systems being tested were tested satisfactorily.

14.6.1.6.6 Reactor Trip System (RTS) Operational Test

This test procedure was very similar to the electrical system logic test-testing to provide the operations necessary to operationally check out the reactor trips of the Reactor Trip System (RTS). The checkout involved a test of the analog system tripping followed by logic train A and B testing. The logic system testing was done with and without covering manual or blocking circuits.

It was first demonstrated that the reactor trip breakers would open automatically and then for the remainder of the test the trip breakers were prevented from opening and the devices that actually tripped the breakers were monitored for performance. Once again, this was a long and complex test that was eventually completed with all objectives satisfactorily met.

14.6.1.6.7 Reactor Coolant System Hydro Test

The function of this test was to verify the integrity and leaktightness of the reactor coolant system and the high pressure portions of the auxiliary systems at 3105 psig (1.25 times the design pressure). All the necessary precautions were taken before the start of the test in that the system had been flushed with high-grade water, the water volume of the reactor coolant system was within the chemical Technical Specifications for cold conditions, pressure-relieving devices were set to relieve at 3120-3170 psig, the participating systems were aligned, no visible leaks were apparent, all possible safety precautions had been taken, and the temperature of the reactor coolant system was above that necessary to pressurize the system.

The reactor coolant system was then pressurized to 1000 psig, with the charging pump and HCF-123 (excess letdown pressure control valve). Upon reaching 1000 psig, the pressure was maintained constant while inspection parties investigated the systems involved for leaks. This procedure was followed for 1500, 2000, 2500, and 3110 psig with only minor problems encountered. This test was successfully completed March 1, 1969, at 1800 hours.

14.6.1.6.8 Ventilation Systems Test

Several tests were written and performed on the various ventilation systems of Ginna Station. The primary purpose of these tests was to functionally check out the systems and to ensure that design flow rates were achieved without overtaxing components and to ultimately balance the systems for flow. These tests were performed by Thomas & Young Associates with RG&E personnel in attendance. Upon completion of these tests RG&E engineers spot checked the various systems using RG&E test equipment to verify the test data.

14.6.1.6.9 Preoperational Containment Vessel Leak Rate Test

The object of the initial preoperational integrated leakage rate test was to establish the degree of leaktightness of the reactor containment building, penetrations, and isolation valves at the design pressure of 60 psig and to establish a reference test for subsequent retests at 35 psig. The allowable leakage of 0.2% per day was defined by the design-basis accident applied in the safety analysis for Ginna Station in accordance with the site exposure guidelines set forth in 10 CFR 100.

The allowable integrated leakage rates for the test were as follows:

Conditions	Allowable Integrated Leak Rate (%/day)[a]
Accident, 60 psig at 286°F	0.1000
Test, 60 psig at 93°F	0.0731

Conditions	Allowable Integrated Leak Rate (%/day)[a]
Test, 35 psig at 93°F	0.0597

a. See item (B) in this section.

During the test period of 6.50 days, the structural integrity test on the reactor containment structure was also conducted. A maximum internal pressure of 69 psig (1.15 times 60 psig design pressure) was used for the structural integrity test. The leakage rate data was gathered over a period of at least 24 consecutive hours after conditions were stabilized at each pressure.

Following each 24-hour period, a controlled leakage rate was superimposed on the reactor containment building to verify and validate the test instrumentation.

A. The reactor containment structure leakage rate at 59.9 psig and 93.2°F was found to be 0.0219 ±0.0168% per day.
B. The leakage rate at 35.1 psig and 93.8°F was -0.0059 ±0.0180% per day. The negative value indicates that the leakage rate was less than the instrumentation sensitivity and ability to react in a relatively short (24-hour) period. With a longer test time, the reduction in error would have led to a better averaging and more definition of the finite rate. When a controlled leakage rate of 4.9 lb/hr was superimposed on the vessel at 35 psig, the calculated rate of 5.05 lb/hr demonstrated the satisfactory performance of the instrumentation.
C. Primary boundary leaks were noted in six penetrations during the test. The resulting leakage was, of course, a part of the overall leakage rate.
D. Comparison of test instrumentation calibrations before and after the test was made and negligible differences were noted.
E. It was not necessary to superimpose a fixed leakage rate at both pressure levels; one was considered sufficient, preferably at the retest condition.

Figure 14.6-11 describes the actual containment vessel pressure versus time. Figure 14.6-12 describes the pressurization system.

14.6.1.6.10 Structural Integrity Test

The Gilbert Associates Report, GAI 1720, Structural Integrity Test, presented the results and observations made on the reactor containment during the structural integrity test on April 11, 1969, to April 14, 1969, and during subsequent depressurization which was concluded on April 18, 1969. The conclusions of the structural integrity test were obtained from the interpretations of test data and responses of the reactor containment when subjected to a maximum internal pressure of 69 psig (115% of design pressure-60 psig).

Most of the structural integrity test instrumentation performed well and their recorded data are regarded as being valid. Some discrepancies in the data were noticed and the significant

discrepancies were noted and discussed. The number of discrepancies was small compared with the amount of data recorded.

The results of the structural integrity test showed the stresses, strains, and displacements were within the limits as defined in the original FSAR and the GAI predicted results. The whitewash areas revealed crack patterns and spacings in good agreement with the GAI prediction; no horizontal cracks were found in dome concrete except for construction joints. The base shear restraint was stiffer than anticipated. The strains and displacements of the cyclinder wall, the discontinuity of dome and cylinder wall, and the dome revealed that the structural stiffness of the containment is greater than anticipated.

The structural capacity of the containment met and exceeded its imposed criteria.

14.6.1.6.11 Reactor Trip System (RTS) Operation Time Response Test

The intent of this test was to determine the response time from the time the plant protection parameters reach their trip setpoints until the tripping time of the reactor trip breakers. In the procedure, the reactor trip time from the deenergizing of the undervoltage coil to the actual tripping of the breaker was recorded, and thereafter in succeeding tests the time from trip setpoint to operation of the undervoltage coil was recorded. From this information the total time from trip setpoint to breaker trip was determined for each of the trip parameters. The trip response time limits as specified in Chapter 14 of the original FSAR were proven to be conservative by the results of this test.

14.6.1.7 Operational and Transient Tests

The tests of this category were designed to test the reactor control and protection systems response.

14.6.1.7.1 Ten Percent Load Swing Test at Thirty Percent Power

The purpose of this test was to introduce a 10% load decrease and verify the nuclear plant transient response including automatic control systems performance and then introduce a 10% increase in load and verify the response and performance again.

The power level was 113 MWe when the test began and 70 MWe after the 10% reduction of load. The load increase was from 70 MWe to 113 MWe. In either case the control system brought the nuclear plant smoothly to the new power level, and there was no measurable amount of nuclear power overshoot. No alarms were observed on the 10% load decrease. Alarms were observed on both steam generators on the load increase.

In both cases rods moved at full speed for about 30 sec and then for a short period at low speed. The rods remained stationary for some 8 min after which they moved two or three steps in the reverse direction at about one or two steps per minute. Steam generator control was smooth and no manual intervention was necessary. On the decrease in load the level in the steam generators decreased about 5% and on load increase the level rose about 5%. T_{AVG} swing was limited to about 2°F. The pressurizer pressure swings were limited to about 20 psi.

14.6.1.7.2 Generator Trip Test

The objective of this test was to verify the ability of the automatic control system and the secondary plant to sustain interaction between systems and accommodate a net electrical load loss from below a 50% power level. The test results were to be evaluated to determine possible changes in control setpoints in order to improve the transient response based on actual plant operation.

The initial power level of the plant was 110 MWe. The main transformer high-side circuit breakers were opened to achieve loss of load. The final power level after the trip was 12 MWe, enough to sustain the plant auxiliary load. The control system responded smoothly and equilibrium conditions were reached in 15 min after loss of load. Controlling rod control bank D moved into the core from 194 steps out of the core just prior to loss of load to 65 steps out of the core at equilibrium conditions. This test was successful.

14.6.1.7.3 Ten Percent Load Swing Test at Seventy-Five Percent Power Level

This test procedure verified the nuclear plant transient response, including automatic control systems performance, when step load changes were introduced at the turbine generator. This test had been performed at a 30% power level previously. The plant load at initial conditions of this test was 348 MWe. A step change to 291 MWe was introduced. After equilibrium conditions were reached, a step change back up to 348 MWe was introduced.

No problem was incurred with either step change. The control system brought the plant to the new power level in approximately 3 min. There was no noticeable overshoot of any major variable. The rods stepped into the core at a rate of 72 steps per minute for 35 sec on the load decrease and stepped out at a rate of 72 steps per minute for 40 sec on the load increase.

Alarms on Load Decrease

1. Steam generator level deviation, loop A.
2. Steam generator level deviation, loop B.
3. Pressurizer low pressure.
4. Feedwater heater and drain tank level.

Alarms on Load Increase

1. Steam generator level deviation, loop A.
2. Steam generator level deviation, loop B.
3. Pressurizer low pressure.
4. Feedwater heater and drain tank level.
5. Charging pump speed.

14.6.1.7.4 Fifty Percent Load Reduction from Seventy-Five Percent Power Level Test

The purpose of this test was to verify the ability of the automatic control system and the ability of the secondary plant to sustain a 50% load rejection from 75% of full power, and the

interaction between the systems. Particular attention was paid to the operation of the steam dump system. Figure 14.6-13 shows some of the more interesting recordings of process variables. The 10% load swing test at 75% power preceded this test by a short time and the variations of the process variables for both tests can be seen in Figure 14.6-13. The test was begun with a power level of 347 MWe and control rods of the controlling bank D at 215 steps out of core. Following the 50% load reduction, the plant leveled off at equilibrium conditions in 17 min and 138 MWe and a bank D position of 35 steps out of core.

The turbine power was run back smoothly during the reduction. Margin to delta T trips increased smoothly. Rods moved in at maximum speed for 1 min and 12 sec. Delta T setpoint 1 dropped to 48°F while actual delta T dropped faster. Six steam dump valves opened and gradually modulated down to two valves open and oscillating slowly but acceptably.

Six min after test initiation rods were at 69 steps out of core on bank D. Pressurizer level rose from 41% to a peak of 48%.

Alarms That Functioned During Load Reduction

- Steam generator level deviation A and B.
- Hotwell level - high.
- Steam generator high feedwater flow loop A.
- Nuclear instrumentation system power range upper detector high flux deviation.
- Pressurizer low pressure.
- High feedwater flow loop B.
- Steam generator low level loop A single channel alert.
- Feedwater heater and drain tank level.
- Nuclear instrumentation system power range lower detector high flux deviation.
- Average T_{AVG} minus T reference deviation.
- Steam generator high level loop A channel alert.
- Steam generator high level loop A.
- Feedwater pump seal-water low differential pressure.

It can be seen on the "A" steam generator feedwater flow recording that there was instability of flow. The control system was allowing the "A" steam generator level to reach 68%, where the feedwater isolation scheme was activated, which accounted for the sharp decrease in flow. The sudden increase in flow after an isolation occurrence was caused by the automatic resetting of the feedwater valve control whereby the valve was allowed to open again. This situation was corrected by changing the "A" steam generator feedwater valve controller response characteristics.

14.6.1.7.5 One Hundred Percent Power Level Transient Tests

Ten percent and 50% load swing tests were performed at the 100% load level that were identical to the same load swing tests at 75% power level. The results of these tests were

satisfactory and similar to those at the 75% level. On March 14, 1970, a plant trip test from 100% power level was successfully conducted. The purpose of the test was to verify the ability of the primary and secondary plant to sustain a trip from 100% power and bring the plant to a MODE 3 (Hot Shutdown) condition in an orderly manner. The test was initiated by pushing the manual turbine trip button on the main control board. The following was verified:

A. The turbine and reactor trips did occur.

B. The steam dump valves did open.

C. Pressurizer safety valves and Main Steam Safety Valves (MSSV) did not open.

D. The safety injection system did not operate.

E. All control rods were inserted in the core.

<u>One Hundred Percent Trip Alarm Annunciations</u>

- 2200 hours
- Manual turbine trip.
- Reactor trip.
- Number 1 generator voltage regulator field forcing.
- Turbine valves single-channel alert.
- Turbine valves auto stop.
- Air to extraction dump valves tripped.
- Feedwater heater and drain tank level.
- Condenser hotwell level.
- Condensate header pressure.
- Feedwater pump seal water low differential pressure.
- Feedwater pump low suction pressure.
- Feedwater pump light load.
- Feedwater pump seal water filter line.
- Auxiliary feedwater pump light load.
- Reactor coolant low T_{AVG} loop A and B.
- Reactor coolant T_{AVG} deviation.
- Average T_{AVG} deviation.
- Pressurizer low pressure.
- Pressurizer safety valve high temperature.
- Nuclear instrumentation system power range upper-high-flux deviation.
- Nuclear instrumentation system power range lower-high-flux deviation.
- Nuclear instrumentation system power range rod stop-rod drop.

- First out indication annunciator.
- Turbine auto-stop.
- Turbine valves.
- Steam generator low-low feedwater level loop A.
- Steam dump armed.
- Steam generator level setpoint deviation A and B.
- Rod bottom rod stop.
- Rod control urgent failure rod stop.
- 115-kV panel.

2210 hours

- Pressurizer liquid high temperature.

2215 hours

- Condensate storage tank (CST) level.

The plant functioned as expected with no major deviation from design intent.

14.6.1.7.6 Operational Dynamic Rod Drop Test

The purposes of this test were as follows:

A. Demonstrate the operation of power range rod drop detection circuits and to provide a basis for the optimum adjustments of setpoints.

B. Demonstrate the operation of turbine runback controller and blocking of automatic rod withdrawal.

C. Evaluate the plant transient response following a dropped rod and demonstrate the adequacy of the dropped rod recovery procedure.

Plant power level was 40% at test initiation. The selected rod J-10 was dropped by removing the fuse from the rods stationary gripper coil circuit. The nuclear plant control system responded smoothly during this transient, but the turbine runback system did not reduce turbine load sufficiently to compensate for the reactivity decrease caused by the dropped rod. The turbine runback was completed manually.

The dropped rod detection was successful as shown in Figure 14.6-14. The four recordings in the figure are of each of the nuclear power channel signals. The two traces of each recording are the signals of the upper and lower ion chambers of a channel.

The dropped rod was detected by the rod position indicator on the main control board and the illumination of the rod bottom light. Verification of the dropped rod was made by the rod position digital voltmeter on the main control board and in-core thermocouple temperature computer printout. Verification could also have been made by running a flux map. On December 10, 1969, while at 30% power, rod J-7 was dropped to the bottom of the core and a

flux map taken at that time confirmed the satisfactory detection of a dropped rod by flux mapping.

The following alarms were actuated during the transient:

AA. Nuclear instrumentation system power range upper detector high flux deviation or auto defeat.

BB. Nuclear instrumentation system power range channel deviation.

CC. Nuclear instrumentation system power range rod drop rod stop.

DD. Rod bottom rod stop.

EE. Steam generator level deviation loop A.

The dropped rod recovery procedure was proven adequate in this test.

This test was successfully rerun the following week after the initial attempt with a satisfactory turbine runback performance.

14.6.1.7.7 Delta T Zero Power Alignment and Delta T Channel Span Adjustment Tests

The delta T zero power alignment test provided instructions for the zero alignment for all four delta T channels. The normal resistance temperature detector inputs into the Dana amplifier (first amplifier of the reactor control and protection system) were disconnected and precision decade boxes were connected to the input of the Dana amplifier and a direct reading voltmeter connected to the output of same. A linearity check of the amplifier was made using the resistance values provided by the test procedure. With the plant at MODE 3 (Hot Shutdown) conditions the amplifier was adjusted to produce an output corresponding to 0.0°F.

The delta T channel span adjustment test provided a curve of amplifier output versus plant load. Upon reaching approximately 75% power, a calorimetric was performed to determine actual level. The Dana amplifiers of each of the protection channels were span adjusted for the actual power level to provide an output as dictated by the linear curve of amplifier output versus plant load.

14.6.1.7.8 Nuclear Instrumentation Calibration and Reactor Coolant System Flow Confirmation

The purpose of this procedure was to specify the requirements for obtaining data for nuclear instrument calibration and reactor coolant system flow confirmation and to check the performance of the nuclear instruments by:

A. Obtaining a plot of anode voltage versus source range instrument output for use in setting source range anode voltage.

B. Obtaining nuclear instrument channel overlap data during increases and decreases in power.

C. Plotting power range detector currents to verify linearity of detector outputs.

D. Determining operational settings of instrument compensating voltages and test current values.

E. Obtaining a plot of detector voltage versus output for intermediate and power range output for use in setting detector voltage.

A plot of source range detector (B10) anode voltage versus detector output in counts per second was obtained for each source range detector as follows:

AA. Prior to core loading and prior to initial criticality, data was obtained for anode voltage plot using startup source.

BB. One to 2 hours after shutdown from power operations of at least 500 MW days. (These plots were performed with neutron flux resulting from gamma-neutron reactions in the core and a significant gamma field incident on the detector).

With the source range channel adjusted per the NIS Instruction Manual, anode voltage was varied in 25-V steps over its adjustable range, without exceeding the maximum allowable operating voltage of 1000-V dc. Data was obtained of anode voltage versus counts per second (cps), and plotted for each of the two conditions specified above. The anode voltage setting was determined from the plot using the criteria that the voltage should be set at a point above the start of the plateau, corresponding to one third of the voltage plateau length. The anode voltage was set and recorded on a data sheet.

Immediately after anode voltage data was obtained for the condition 2 above, and after setting the anode voltage, the discriminator voltage was varied in 0.2-V steps over the operating range and data was obtained to perform a plot of discriminator voltage versus cps. Discriminator voltage was adjusted to a point determined from the plot.

The four power range nuclear instrument channels were calibrated based on a calorimetric measurement of the secondary system. The power delivered by each steam generator was determined by measurement of feedwater flow, feedwater temperature, and steam pressure. A second method of determining the power delivered by the reactor was by measuring the delta T across each reactor coolant system loop and the reactor. The delta T measurements were used to verify the feed flow method and were also used as a means of verifying loop flow. Measurements of feed flow were made by venturi meters installed in the feed flow lines to each steam generator. Differential pressure instruments installed across the venturi meters indicated differential pressure which was used to determine reactor power from a curve of feedwater temperature versus the square root of differential flow pressure. Percent reactor power was determined for each power level and a calorimetric calibration was performed by summing the power being delivered by each steam generator as determined from the curve (less net thermal input due to pump operation, radiant heat loss and letdown) and dividing by the design full power output. Percent power was computed as follows:

Power % = [(Power loop A + Power loop B - Power heat gains) x 100] / 4437 x 10^6 Btu/hr

In performing a calorimetric calibration, plant power was increased to the approximate level as indicated by the feed flow differential pressure detector and as a backup by the watt meter in the main generator output. In increasing power to the levels specified in the tabulation below, the feedwater flow differential readings and watt meter readings as indicated below were not exceeded for specified power.

Final Approximate Power Level (%)	Feedwater Flow D/P Meter Reading	Watt Meter Reading (MWe)
30	(Obtained from curve)	150
50	(Obtained from curve)	240
75	(Obtained from curve)	360
100	(Obtained from curve)	460

Once the nuclear instrument calibration data had been taken, the reactor power calculations were performed by feed flow, and by reactor and loop delta T methods. Using the results of these calculations, the gain of the power range channel indicating closest to the calculated power by feedwater flow was adjusted. Following this gain adjustment, the gain of the other three channels was adjusted as necessary to match this channel. Prior to adjusting the gain of power range instruments, an examination of the flux maps and out-of-core flux (current) readings was made for the power at which the calorimetric data was taken, for any asymmetrical flux pattern that could explain any difference in out-of-core indication.

Reactor power determined from delta T measurements was used for informational purposes.

Reactor power was computed using loop and reactor delta T measurements as follows:

Loop Delta T (Spare Resistance Temperature Detectors Th-Tc) Method

$$Percent\ ReactorPower = \frac{FullDesignFlow\,\frac{lb}{hr}\,x\,\frac{deltaTA + deltaTB}{2} - P_{netheatgains}\,x100}{FullDesignPower\,\frac{Btu}{hr}}$$

$$= \frac{68.0x10^{6}\,\frac{lb}{hr}\,x\,\frac{(deltaTA + deltaTB)}{2} - P_{netheatgains}\,x100}{4437x10^{6}\,\frac{Btu}{hr}}$$

(Equation 14.6-1)

A plot of average power range detector current versus power to determine degree of linearity was made.

A plot of power range detector current versus detector voltage at near full power condition to determine operating voltage (twice voltage for 90% of saturated current condition) was made.

An in-core flux map was made for each steady-state power level for obtaining nuclear instrumentation calibration data.

An approximation of design reactor flow was computed using differential temperature measurement and reactor power as one means. Both loop and reactor differential temperatures were used in making these computations.

14.6.1.7.9 Ex-Core In-Core Calibration Test

It was the function of this test to establish a relationship between in-core and ex-core generated axial offset and delta flux.

The results of this test were later used to calibrate the upper and lower detector channels and to align the axial offset signals to the delta T setpoints.

With the part-length control rods inducing an axial offset by virtue of their position in the core (10 steps from the bottom) and with the plant electrical load maintained constant, a flux and thermocouple map was run under these conditions and the ex-core detector voltages were recorded periodically. This same procedure was followed with the part-length rods located 85 and 160 steps out of core. Two more runs were made with bank D positioned on the bottom of the core as opposed to about 15 steps from the bottom, as was the case in the first three runs.

This test was again run at 75% power to verify the channel settings and to further refine settings for extrapolation to 100% power.

This test did establish the fact that there was a fairly linear relationship between the in-core and ex-core axial offset and a linear relationship between offset and power level.

14.6.1.8 Startup Physics Testing

14.6.1.8.1 Introduction

An extensive physics testing program was conducted to see if the core reactivity characteristics and power peaking were close to design calculations and conservative with respect to assumptions used in the safety analysis. Measurements were made to determine:

A. Core reactivity parameters, including reactivity coefficients and control rod bank worths.

B. Power distributions, from zero to full power, with and without control rod bank insertion.

C. The effects of abnormal rod configurations, including individual rods fully withdrawn, fully inserted, and intermediate out-of-position configurations.

D. The adequacy of the ex-core instrumentation to monitor core performance for both normal and abnormal control rod configurations.

The conclusions drawn from the physics program results were as follows:

AA. The core performance was quite close to the design predictions.

BB. The measured values for physics parameters required for safety analyses were less restrictive than the assumed values.

CC. The core instrumentation system was successful in monitoring the core power distribution and sensing power asymmetry.

14.6.1.8.2 Power Distribution Measurements

The power distribution measurement results were documented separately in WCAP-7542-L, September 1970, Topical Report: Power Distribution Monitoring in the R. E. Ginna PWR. The responses of the ex-cores, thermocouples, and incore movable detectors to both normal and abnormal power distributions are discussed therein.

14.6.1.8.3 Zero Power Critical Boron Concentrations

A summary of key reactivity measurements made during the initial physics tests is presented in Table 14.6-2. These zero power measurements are in excellent agreement with predicted values with the exception of the stuck rod configuration which has all rods but one inserted. The measured boron concentration is less than the predicted value, indicating the reactor had a greater total rod worth than predicted for the limiting stuck rod configuration.

14.6.1.8.4 Reactivity Coefficients and Shutdown Margin

In Table 14.6-3 the isothermal temperature coefficients are in good agreement with the predictions. It should be noted that the positive coefficient does not exist with the normal rod configuration at zero power or at any other power level. This all-rods-out case was achieved only for the purpose of the test program and was specifically permitted by the Technical Specifications.

The measured doppler coefficients shown in Table 14.6-3 are larger than predicted. The shut-down margin calculated from measured data is greater than the design value by roughly 0.3% reactivity. The greater measured doppler defect is overcome by the greater measured rod worth with one stuck rod for a small gain in shutdown margin.

14.6.1.8.5 Ejected and Dropped Rod Worths

The statically ejected and dropped rod worths are listed in Table 14.6-4. In the safety analyses, the ejected rod for the zero power case was assumed to be worth 1% reactivity. The measured value (0.75% reactivity) shows this assumption was conservative. The measured ejected rod for the limiting full power configuration was 0.30% reactivity, compared to 0.365% assumed in the safety analysis. The measured power peaking factors (documented in WCAP-7542-L) for these two rod configurations are compared to the values assumed in the safety analyses at the bottom of Table 14.6-4.

The dropped rod reactivities presented in Table 14.6-4 are not directly related to any safety concern; no minimum or maximum limit was used in any original FSAR safety or accident analysis.

14.6.1.8.6 Xenon Oscillation Test

An xenon oscillation test was performed to determine the dampening characteristics of the 12-ft core. The oscillation was induced by bank D insertion for 4 hours, then the bank D was withdrawn. The part-length bank was held at the midplane throughout the test. The initial oscillation had the following characteristics:

Period	28 hours
Amplitude (t + 1/2 period)/Amplitude (t)	0.5

The oscillation was again induced, but axial symmetry was maintained using part-length rod movement to counteract the xenon oscillation. The part-length rods successfully held axial offset at 0%.

14.6.2 POWER TEST PROGRAM TO 1520 MEGAWATTS THERMAL

14.6.2.1 Test Description

Rochester Gas and Electric Corporation obtained an amendment to the operating license for the R. E. Ginna Nuclear Power Plant on March 1, 1972, which authorized an increase in the plant output from 1300 to 1520 MWt. The testing program during the initial escalation in power to 1520 MWt was performed in March and April 1972. The objective of the program was to ensure a well-documented transition from 1300 to 1520 MWt. The core average burnup was 14,800 MWd/metric ton uranium.

A set of base conditions was measured at 1300 MWt before power escalation was initiated to serve as a basis for comparison with subsequent tests. These base conditions included chemical and radioactivity levels at typical locations, radiation measurements, power distribution measurements, and a core performance evaluation. These tests were repeated at intermediate power levels of about 1380 MWt and 1455 MWt, before going to the full power level of 1520 MWt.

Figure 14.6-15 displays the reactor power level as a function of time for the test period. There were several distinct phases to the uprating program. Following a 5-day plant shut-down, a number of reactor physics parameters were measured at hot zero power. While these zero power tests were not a necessary portion of the testing program, the shutdown did afford an excellent opportunity for obtaining end-of-cycle physics data for use with nuclear design calculations.

At the completion of the zero power testing, the power escalation program was initiated. As can be seen in Figure 14.6-15, this escalation was comprised of several discrete steps, from 0 to 1300 MWt, from 1300 to 1380 MWt, from 1380 to 1455 MWt, and in mid-April to 1520 MWt. After each new power level was reached, a number of tests and measurements were performed. These included flux and delta T measurements, containment vessel radiation surveys, and primary coolant activity level measurements. Data obtained at each power level were reduced and evaluated before the core power was increased. In addition, careful attention was paid to system components during all phases of the escalation program.

As power was increased to 1300 MWt, the power coefficient was measured and the power defect obtained. A review of test results at 1300 MWt, including a detailed check of the flux map results showed good agreement with the expected data. Data obtained at 1380 MWt also displayed this good agreement with predictions, thus justifying a further power increase to

1455 MWt. In addition to the tests outlined above, flux maps were obtained at 1455 MWt to facilitate the generation of the axial offset f(delta I) setpoints for operation at 1520 MWt.

Two phenomena caused further power escalation beyond 1455 MWt to be postponed. A higher than expected primary coolant activity was encountered and steam line vibration was noted. The results of all other tests were favorable and indicated that power could be raised to 1520 MWt.

On April 12, 1972, shortly before the cycle 1B MODE 6 (Refueling) shutdown, core power was increased to 1520 MWt for approximately 6 hours. The escalation from 1300 MWt proceeded at 1% power per hour and followed 4 days operation at 1300 MWt for primary coolant activity cleanup. Plant improvements had been implemented to remedy the steam line vibration problem. The purpose of the operation at 1520 MWt was to test the secondary system at 1520 MWt before the annual maintenance period and to test the fuel prior to conducting the fuel inspection. After completing all tests outlined above, the reactor was returned to 1300 MWt.

Testing performed during the power escalation program demonstrated that the plant could be operated at 1520 MWt. Core flux and delta-T maps showed that, as expected, there was very little change in assembly relative power levels as core power was raised from 1300 to 1520 MWt. Margins to the core safety limits remained large. For example, the measured peak FNQ, including a 5% measurement uncertainty, was 1.63. This may be compared with the Technical Specification limit at that time of 2.72. One reason for the low measured value is that a full cycle of depletion had taken place; peaking factors are expected to be largest at the beginning of a cycle.

The containment radiation surveys did not reveal any unexpected increases in radiation levels during or following the escalation program. The primary-coolant radioactivity levels did, however, increase more rapidly than expected, particularly for the shorter-lived isotopes such as Xenon-135, Krypton-87, and Krypton-88. The effect of the power escalation on fuel rod integrity can best be analyzed by comparing primary coolant activity following equilibrium operation at 1300 MWt prior to early March with the activity at 1300 MWt in early April. These data indicate that some additional fuel rods probably failed between early March and early April. The data obtained at 1520 MWt could not be evaluated in this fashion since several days of operation at a given power level is required to reach equilibrium coolant activity conditions. The small increase in coolant activity noted at 1520 MWt compared to the 1300 MWt levels did indicate the beneficial effect on coolant activity of increasing power level slowly.

A comprehensive testing program was established and followed to ensure an orderly power escalation. Care was taken to evaluate all available information before proceeding to new power levels greater than 1300 MWt. This care ensured that a safe and well-documented program was carried out which resulted in demonstrating that the core could be operated at 1520 MWt.

14.6.2.2 Steam Generator Moisture Carryover Tests

Steam generator moisture carryover tests were performed at each power level during the uprating test program and at 1520 MWt in April.

The results of these tests were as follows:

Power	Date	Percent Carryover
1300	March 11, 1972	0.02
1380	March 12, 1972	0.052
1455	March 13, 1972	0.21
1455	March 14, 1972	0.22
1520	April 12, 1972	0.519

The moisture carryover met the 0.25% requirement at the warranted power level of 1455 MWt.

14.6.2.3 Assembly Delta T Measurements

In conjunction with each flux map, a complete set of assembly temperature rise measurements was taken. Before proceeding to a new power level above 1300 MWt, the last set of thermocouple data was extrapolated to the new power level based on the power increase and on the expected assembly relative powers. While the temperature measurements are not as accurate as the flux measurements, they do provide a quick check of the assembly power levels. In general, the measured assembly exit temperatures were within 1°F of the expected values. (The temperature rise through the core at 1520 MWt is approximately 58°F.) In the few cases where the differences were larger than 1°F, the flux maps ensured that the assembly power levels were as expected.

14.6.2.4 Plant Radiation Surveys

Radiation surveys were made throughout the plant with portable survey instruments during the power escalation program. Gamma and neutron radiation levels were measured at a number of points on the operating floor, the intermediate floor, and the basement of the containment vessel. The measurements gave a rough estimate of the radiation levels in the containment. Accuracy of the measurements was limited since surveys at different power levels were taken by different people, a constant counting geometry could not be maintained at each survey station, and the high radiation levels gave only a short time in which measurements could be made. In addition, nonequilibrium effects and the changes in waste treatment system flow rates could have introduced errors into the measurement.

It was expected that the neutron radiation levels would be proportional to the power. The gamma radiation level should not, however, be proportional to power since it depends on waste treatment. A summary of the data is presented in Table 14.6-5. The values listed in the

table for radiation increase refer to the average of the surveys taken at a particular power level and are related to the average obtained at 1300 MWt.

14.6.2.5 Reactor Physics Measurements

14.6.2.5.1 Zero Power Measurements

Following the scheduled 5-day shutdown prior to the uprating program and while at a nominal hot zero-power level, a number of reactor physics measurements were performed. The results are primarily of benefit in reactor design and development and were not an important facet of the uprating program. These tests included:

- Critical boron concentration - All rods out.
- Isothermal temperature coefficient - All rods out.
- Bank D differential and integral worth.
- Critical boron concentration - Bank D inserted.
- Isothermal temperature coefficient - Bank D inserted.
- Bank C differential and integral worth (bank D inserted).
- Critical boron concentration - Banks C and D inserted.
- Isothermal temperature coefficient - Banks C and D inserted.

Basic results of the measurements are reported in Table 14.6-6.

The worths of bank D and bank C were less than those predicted and measured at the beginning of cycle 1B. This might have been expected since the relative power in the rodded assemblies decreased during the cycle. The plots of integral and differential worth for banks D and C are presented in Figures 14.6-16 and 14.6-17, respectively.

The isothermal temperature coefficient was obtained as a function of boron concentration by taking measurements of several different control rod insertion configurations. At a given rod configuration, the moderator temperature was varied about a nominal value to obtain the reactivity effect of such a change. The isothermal temperature coefficient was found to be a nonlinear function of the boron concentration, as can be seen in Figure 14.6-18. Nonlinear behavior was expected based on the curves presented in Section 3.2.1 of the original FSAR. The nonlinearity may have been due in part to the different control rod configurations employed. The changes seen in the nominal moderator temperature may have contributed to the nonlinearity due to the effect on the neutron diffusion length as a result of changing moderator density. The data at 616 and at 532 ppm of boron are about 10% less negative than predicted for the end-of-life by the cycle 1B design report, while the value at 422 ppm agrees well with the prediction.

14.6.2.5.2 At-Power Measurements

Upon conclusion of the zero power measurements, reactor power was increased to 1300 MWt in several steps. During this increase, the power defect was measured. The integral power defect (doppler, moderator temperature, and flux redistribution) from zero to 1300 MWt was

measured to be 1.33% delta P at the critical boron concentration of 420 ppm. The reactivity defect due to doppler and flux redistribution was obtained by removing the reactivity effect of increasing the moderator T_{AVG} and was found to be 1.00% delta ρ from zero to 1300 MWt. The values predicted at the end-of-life for the doppler defect and the power defect (not including flux redistribution) are approximately 1.18 and 1.70 delta ρ , respectively.

After correcting the data for variation in moderator temperature and xenon redistribution, the power coefficient as a function of power was obtained. These data are plotted in Figure 14.6-19 Data were not obtained between 1300 and 1520 MWt because it was decided not to subject the core to the rapid transient which would have been necessary. Power transients had been found to result in a temporary increase in primary coolant activity.

Upon reaching 1300 MWt, the main portion of the uprating tests began. The intent was to take core maps at 1300, 1380, 1455, and 1520 MWt. Each map was to be analyzed before proceeding to a higher power level. At 1300 MWt, a reference flux map was taken to serve as a basis for evaluation of the power distribution obtained at higher power levels. Excellent agreement was found between the measurements and the predicted power distributions.

A flux map was taken at 1380 MWt and three maps, for use in the f(delta I) setpoint calibration, were taken at 1455 MWt. Selected system parameters for these maps are given in Table 14.6-7.

In all cases, the agreement between measured and predicted power distributions was very good. Differences were typically less than 3%. The relative power distributions symmetry at 1300 MWt, 1380 MWt, and 1455 MWt are shown in Figure 14.6-20. For ease of presentation, the values listed in the figures represent the average for the four quadrants. These data demonstrated that the power distributions were well behaved and that there were no unexpected hot assemblies.

The assembly relative power distributions for the three flux maps taken at 1455 MWt are given in Figure 14.6-21. There were no major differences in the assembly power distributions of these three maps and the measurements agreed well with the predictions from PDQ calculations.

The power range detector output was monitored as a function of core power. In Figure 14.6-22, the output of ex-core detector NE-41 (sum of the top and bottom detectors) is plotted as a function of core power and a linear correlation is seen. A similar linear correlation was seen for detectors NE-42, NE-43, and NE-44. The correlation between ex-core detector response and the axial offset as calculated from the flux map data is presented in Figure 14.6-23 for detector NE-41. The linearity of detector response with axial offset was also found in the other three ex-core detectors. This linearity demonstrated that the detectors continued to accurately monitor core axial offset and that the data obtained at 1455 MWt could be used to generate the f(delta I) setpoints for operation at 1520 MWt.

The power escalation program was halted at 1455 MWt due to high coolant activity and steam line vibration. On April 12 the reactor was taken to 1520 MWt for a period of 6 hours so that system component behavior might be determined before the mid-April MODE 6 (Refueling). A core flux map was taken during the operation at 1520 MWt and the results of

that map are presented in Figure 14.6-24. The assembly relative powers could be directly compared with the earlier maps in the uprating program since the part-length rods were withdrawn from the core prior to increasing power to 1520 MWt. The difference between measurement and prediction was, typically, less than 2%.

Table 14.6-1a
ACCUMULATOR BLOWDOWN TEST RESULTS - OBSERVED RESULTS

	Low Pressure		High Pressure	
	Loop A	Loop B	Loop A	Loop B
Initial level, %	40	40	40	40
Initial gas volume, ft^2	750	750	750	750
Initial pressure, psig	300	300	740	720
Valve opening time, sec	10	9.7	10	---
Delay before fluid enters vessel, sec	---	---	4.5	4.5
Liquid blowdown time - total, sec	---	---	29	54
Gas blowdown time, sec	---	---	30	50
Water lost to loop seal, gal	640	875	1110	1125

Table 14.6-1b
ACCUMULATOR BLOWDOWN TEST RESULTS - PREDICTION OF FINAL PRESSURE

Run	Loop A Initial pressure	740 psig
	Initial gas volume	750 ft^3
	Final gas volume	1750 ft^3
Final pressure	Calculated (pv =C)	220 psig
	Measured - PT – 936	200 psig
	PT – 937	220 psig

Table 14.6-1c
ACCUMULATOR BLOWDOWN TEST RESULTS - PIPE RESISTANCE

Loop	Initial Pressure (psig)	Pipe Resistance[a] Equivalent Diameter (L/D)
A	740	305 Same
A	740	334 Run
A	300	371
B	300	388
B	740	1275

a. Resistance used in accident analysis - 530

Table 14.6-2
BEGINNING OF CYCLE ZERO POWER CRITICAL BORON CONCENTRATIONS

Zero Power Critical Boron Parameter	Measured (ppm)	Predicted (ppm)
All rods out	1608	1609
Bank D in	1526	1528
Banks C and D in	1365	1382
Banks B, C and D in	1253	1270
Part-length at midplane	1566	1566
Part-length out, 28 rods in	960 ±25[a]	1015

NOTE:—10 ppm = 100 pcm nearly equal to 0.1% reactivity

a. Inferred from subcritical state. Large uncertainty due to noncritical measurement.

Table 14.6-3
REACTIVITY COEFFICIENTS AND SHUTDOWN MARGIN

	Measured	Predicted
Isothermal temperature coefficient		
Zero power, all rods out	$+1.4 \times 10^{-5}$/°F	$+1.2 \times 10^{-5}$/°F
Zero power, bank D inserted	-2.4×10^{-5}/°F	-1.9×10^{-5}/°F
Doppler coefficient		
10% power	-40×10^{-5} / % Q	-27×10^{-5} / % Q
30% power	-22×10^{-5} / % Q	-16×10^{-5} / % Q
85% power	-10×10^{-5} / % Q	-6.5×10^{-5} / % Q
Doppler defect		
50% power	1.45%	1.00%
100% power	2.03%	1.40%
Shutdown margin		
Beginning-of-life	3.11%	2.85%
Estimated end-of-life	2.60%	2.27%

Table 14.6-4
EJECTED AND DROPPED ROD WORTHS

Rod	Bank	Position	Worth (ppm)	Bank Positions S	A	B	C	D	Power (%)
Ejected rods									
K-7	D	230	75[a]	230	175	5	5	5	0
K-7	D	230	30[b]	230	230	230	230	20	30
G-11	D	230	29	230	230	230	230	20	30
K-7	D	230	8	230	230	230	230	153	30
J-10	C	230	20	230	230	230	107	22	30
K-7	D	230	38[c]	230	230	230	107	22	30
G-7	C	230	18	230	230	230	107	22	30
Dropped rods									
G-7	C	0	45	230	230	230	230	20	30
K-9	S	0	33	230	230	230	230	20	30
G-7	C	0	23	230	230	230	230	180	30
I-7	B	0	22	230	230	230	230	167	30
J-10	C	0	20	230	230	230	230	169	30
F-12	A	0	20	230	230	230	230	149	30
K-7	D	0	10	230	230	230	230	141	30
K-9	S	0	18	230	230	230	230	153	30
I-7	B	0	40	230	230	230	230	22	30

NOTE:—10 ppm = 100 pcm nearly equal to 0.1% reactivity.

a. FSAR safety analysis zero power assumed rod worth: 100 ppm Fq (measured) = 7.71 Fq, (assumed) = 12.6

b. SAR safety analysis full power assumed rod worth: 36 ppm Fq (measured) = 2.58 Fq, (assumed) = 4.75

c. Bank C insertion not permitted at full power. Position of bank C corresponds to 40% power insertion limit.

Table 14.6-5
AVERAGE INCREASE IN CONTAINMENT RADIATION LEVELS DURING UPRATING PROGRAM TO 1520 MEGAWATTS THERMAL

	Date			
Reactor Power (MWt)	3/11/72 1300	3/12/72 1380	3/13/72 1455	4/12/72 1520
Percent increase	0	6.1	11.9	17
Neutron radiation percent increase	0	6.2	2.4	23
Gamma radiation percent increase	0	4.0	6.5	18

Table 14.6-6
SUMMARY OF MEASURED PARAMETERS AT HOT ZERO POWER PRIOR TO UPRATING TO 1520 MEGAWATTS THERMAL

Parameter	Measured Value
Control bank integral worth, pcm	
Bank D	839
Bank C	1176
Critical boron concentrations, ppm	
All rods out	616
Bank D in	535
Banks C and D in	425
Boron worth, pcm/ppm	-10.7
Temperature coefficients, pcm/°F	
All rods out, 548 ± 3°F, 616 ppm boron	-13.9 ± 0.4
Bank D inserted, 550 ± 3°F, 532 ppm	-14.8 ± 0.2
Banks C and D inserted, 547 ± 2°F, 422 ppm	-18.1 ± 0.1

Table 14.6-7
SELECTED DATA FOR FLUX MAPS

Map	Power (Mwt)	Measured Axial Offset (%)	Rod Position D	Rod Position P/L
93	1300	+0.6	213	83
94	1380	-3.1	210	75
95	1455	-0.6	211	67
96	1455	-11.1	211	84
97	1455	+10.3	212	33