14.4 INITIAL TESTING OF THE OPERATING REACTOR

After satisfactory completion of fuel loading and final precritical tests, nuclear operation of the reactor began. The Phase III Plant Operational Testing Program continued through initial criticality, low power testing, and power level escalation. The purpose of these tests was to establish the operational characteristics of the reactor core, to acquire data for the proper calibration of setpoints, and to ensure that operation will be within license requirements. A brief description of the testing is presented in the following sections. Table 14.4-1 summarizes the tests which were performed from initial criticality to rated power.

In the development of the Phase III testing program, PSE&G complies with the intent of Regulatory Guide 1.68, dated November 1973, with the following exceptions:

1. Natural Circulation Test - This test is performed on a first-of-a-kind plant only, and therefore will not be performed on Salem Unit 1. Test results were submitted to the NRC for the Trojan 1 reactor plant, which satisfactorily performed a natural circulation test.

   PSE&G satisfactorily performed Natural Circulation testing on Salem Unit 2 between August 23 and August 29, 1980, as part of a Special Low Power Test Program required by conditions 2.C(6)b and c of Operating License DPR-75. The Test Program consisted of the following tests:

   SUP 90.1 Natural Circulation Test
   SUP 90.2 Natural Circulation with Simulated Loss of Offsite AC Power
   SUP 90.3 Natural Circulation with Loss of Pressurizer Heaters
   SUP 90.4 Effect of Steam Generator Secondary Side Isolation on Natural Circulation

SUP 90.5 Natural Circulation at Reduced Pressure

SUP 90.6 Cooldown Capability of Charging and Letdown

SUP 90.7 Simulated Loss of All Onsite and Offsite AC Power

A test Summary was transmitted to the NRC in a Letter dated September 8, 1980.

2. Loss-of-Flow Test - The reactor coolant flow coastdown test performed during Phase III testing will provide data on loss-of-flow characteristics. The plant trip test (see Table 14.4-1) performed during Phase III testing will provide additional Reactor Coolant System response data. Sufficient information will be made available from these tests so that a special loss-of-flow test at power will be unnecessary.

3. Vibration Measurements on Reactor Internals - Extensive vibration testing and a detailed vibration analysis are performed on each first-of-a-kind unit only and reported to the NRC. Indian Point Unit 2 has been established as the prototype for all four-loop plants. An extensive inspection of reactor internals will be performed in lieu of vibration measurements.

4. Pressure Coefficient of Reactivity Measurements - Direct measurements of the pressure coefficient of reactivity are not necessary, since the change in reactivity (PCM) over the entire pressure range is so small that such a test is of no practical value.

5. Axial Xenon Suppression Test (Part-Length Rod Effectiveness) - Determination of part-length effectiveness for controlling xenon transients is not necessary since use of part-length rods is prohibited and they will not be installed. Constant axial offset

control required by the Technical Specifications is the method of controlling xenon transients.

6. Turbine Trip - This test will be performed in conjunction with the generator trip test, which will be performed at 100 percent of rated thermal power. It is expected that a reactor trip and a turbine trip will result within a reasonable period of time following opening of the generator main breaker. During performance of this test, the time delay between the generator trip and the turbine trip, as well as any turbine overspeed, will be noted and recorded.

6a. Process Computer - This test will not be performed since the process computer performs no control or protection function. It serves only as a data logger.

7. Dynamic Rod Drop Test - The purpose of including this test in previous test programs has been to test the response of the automatic turbine runback feature. This control feature is not incorporated into the Salem design. Plant response to a dropped-rod casualty will therefore be tested by performing a rod drop test, which also will test the negative rate trip circuitry. The test will be performed as part of the at-power testing sequence (see Table 14.4-1).

Note: Salem NRC License Amendment 278-261 (Salem 1 and 2, respectively) approved the removal of the Negative Flux Rate Trip. This function was initially disabled by setting the setpoint to a greater value than the Maximum Negative Rate expected (per design change package (DCP) 80094424). The Negative Flux Rate Trip circuitry has been physically removed from both Unit 1 and 2 per DCPs 80097106 and 80099680.

14.4.1 Initial Criticality

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

Successive stages of control rod assembly group withdrawal and of boron concentration reduction were monitored by observing changes in neutron count rate as indicated by the regular source range nuclear instrumentation as functions of group position during rod motion and, subsequently, a reactor coolant boron concentration and primary water addition to the Reactor Coolant System during dilution. Throughout this period periodic samples of the primary coolant boron concentration were obtained and analyzed.

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 control rod assembly group withdrawal and during reactor coolant boron dilution. The rate of approach was reduced as the reactor approached extrapolated

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criticality to ensure that effective control was maintained at all times.

Written procedures specified alignment of fluid systems to allow controlled start and stop and adjustment of the rate at which the approach to criticality was expected, and identify chains of responsibility and authority during reactor operations.

14.4.2 Lower Power Testing

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

The measurements were made at lower power and primarily at or near operating temperature and pressure. Measurements which were made included verification of calculated values of control rod assembly group reactivity worths, is thermal temperature coefficient under various core conditions, differential boron concentration reactivity worth and critical boron concentrations as functions of control rod assembly group configuration. In addition, relative power distribution measurements were made. Concurrent tests were conducted on the instrumentation including the source and intermediate range nuclear channels.

Detailed procedures were prepared to specify the sequence of tests and measurements to be conducted and the conditions under which each was to be performed to ensure both safety of operation and the relevancy and consistency of the results obtained.

14.4.3 Power Level Escalation

When the core performance characteristics of the reactor were verified by the low power testing, a program of power level escalation in successive stages brought the unit to its full rated power level. Both reactor and unit operational characteristics were closely examined at each stage and the relevance of the safeguards analysis verified before escalation to the next program level was effected.

Measurements were made to determine the relative power distribution in the core as functions of power level and control assembly group position. These measurements supplied additional core performance data in terms of heat flux and margins to departure from nucleate boiling.

Secondary system heat balances ensured that the several indications of power level were consistent and provided bases for calibration of the power range nuclear channels. The ability of the Reactor Control System to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operation was verified.

The dynamic response characteristics of the reactor coolant and steam systems were evaluated at prescribed power levels. The responses of system components were measured for 10-percent reduction of load and recovery, 50-percent reduction of load, plant trip, and trip of two control rods.

Adequacy of radiation shielding was verified by gamma and neutron radiation surveys inside the containment and throughout the station 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.

14.4.4 Post Startup Surveillance and Testing Requirements

Post startup surveillance and testing requirements were designed to provide assurance that essential systems, which include equipment components and instrument channels, are always capable of functioning in accordance with their original design criteria. These requirements can be separated into two categories:

1. 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 response.

2. Reliability is maintained at levels comparable to those established in the design criteria and during early station 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 confirmed. Verification of operation of complete systems is checked at refueling intervals. Individual checks of components and instrumentation are made at more frequent intervals as outlined in the Technical Specifications.

The techniques used for testing the instrument channels included a preoperational calibration which confirmed values obtained during factory test programs. These reconfirmed calibration values became the reference for recalibration maintenance at refueling intervals during station life. Periodic testing, as defined to the Technical Specifications, includes the insertion of a predetermined signal that trips the channel bistable. Indication of the operation is confirmed and recorded.

Testing of components is initiated through manual actuation. If response times are of importance, they are measured and recorded. The capability to deliver design output is checked with instrumentation and compared against design data. Allowable deviations have been established in the Technical Specifications. The component is operated a sufficient length of time to allow equalization of operating temperatures in bearings, seals, and motors. These parameters are checked periodically. The component is surveyed for excessive vibration and readings are recorded.

Public Service Electric & Gas believes that testing in accordance with the program described above 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.

14.4.5 Safety Precautions

The test operations during low power and power escalation were similar to normal station operation at power, and normal safety precautions were observed. Those tests which required special operating conditions were accomplished using test procedures which prescribed necessary limitiations and precautions.

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