Yankee Atomic Electric Company
Research And Development Program

THE STARTUP EXPERIMENT PROGRAM
FOR THE YANKEE REACTOR

By

J. M. Gallagher, Jr.
Plant Development
H. W. Graves, Jr.
Reactor Development
D. Hunter
Nuclear Power Services
WESTINGHOUSE ELECTRIC CORPORATION

J. E. Howard
YANKEE ATOMIC ELECTRIC COMPANY

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ABSTRACT

A series of experiments was performed with the Yankee Reactor Plant to determine the characteristics of the reactor and primary system following initial criticality. These included reactor physics experiments, control rod drive tests, and primary system transient response studies. This report describes the experiments and evaluates the results from the standpoint of previous analysis. Agreement with calculations was found to be very good.
1.0 INTRODUCTION

The reactor which provides the heat source for the Yankee Atomic Electric Company power plant was subjected to an extensive test program between July 1960 and February 1961. This report describes the series of experiments which were performed on the reactor plant before it was put into service as an operating power plant. This reactor is of the pressurized light water type, with rod shaped fuel elements of slightly enriched $UO_2$ contained in stainless steel. The reactor design was the object of a research and development program sponsored by the U. S. Atomic Energy Commission.

The program of startup experiments described herein was the final phase of the program of research, development, and design of the Yankee reactor and plant which began in mid 1956. At that time the reactor was the first large power reactor to use the combination of slightly enriched $UO_2$, stainless steel, and light water together. In addition to the novel combination of materials, the system of reactivity control used for the Yankee core is also a departure from that used in any other reactor, in that a neutron absorber, dissolved in the moderator-coolant is used to provide cold shutdown of the reactor. The neutron absorber is boron in the form of boric acid ($H_3BO_3$), and the concentration is varied by means of both a bleed and feed system, and an ion exchange system. At operating temperature sufficient reactivity control is provided with 24 cruciform shaped neutron absorbing (Silver-Indium-Cadmium alloy) control rods and the chemical control system is not required.

Because there was very little experimental physics information available at the beginning of the Yankee project, a series of part core critical experiments was carried out at the Westinghouse Reactor Evaluation Center in Waltz Mill. These experiments were performed with fuel rods very similar to those now being used in the Yankee power reactor with the primary difference being enrichment (2.7% vs. 3.4%). The object of these experiments was to develop as much experimental information as possible to help in determining fuel enrichment (or fuel loading), control rod worths, and, to some extent, kinetic parameters. These experiments were used to provide experimental verification of the analytical methods used to specify the fuel loading for the Yankee reactor in February 1959. In addition to the difference in operating
temperature between Waltz Mill experiments and the operating reactor, a major extrapolation in size was required since the largest core studied at Waltz Mill consisted of roughly 4000 fuel rods, 4 feet long. The Yankee reactor consists of slightly more than 23,000 fuel rods 7-1/2 feet in length; a factor of ten in size extrapolation. Because no full scale critical experiments had been run, a detailed series of experiments at the plant site was planned to determine the reactor characteristics.

The purpose of the startup program can best be summarized by referring to the Yankee Hazards Report which states "The primary purpose of these experiments is to check the analytical evaluation of $k_{eff}$, control rod and boron worth, reactivity coefficients, and other parameters which affect the operational safety of the plant." A secondary purpose, which is of more interest to the nuclear designer, is to evaluate the analytical methods used to determine the nuclear characteristics of the first Yankee core.

Core loading was begun on July 15, 1960 and criticality was first achieved in the fully loaded core on August 19. The first series of experiments was conducted during very low power operation (0 to 3 MWt) at approximately 100°F. These experiments established the worth of boron and banked control rods in the cold core, and the boron concentration required for 5% shutdown. Control rod drop times and mechanism operation were also evaluated in the cold reactor. In the ensuing series of experiments, the moderator temperature coefficient with high boron concentration (1050 to 1550) was evaluated between 100°F and 530°F during heatup of the primary loop. The worth of boron, of banked control rods, and of the control rod groups in the control rod program withdrawal sequence were then evaluated at low power. Control rod drop times with the primary loop at operating temperature were also determined. The final phase of the experimental program was concerned with determining the response of the reactor to various power conditions, such as load changes, xenon transients, and varying main coolant flow conditions. At the completion of this phase, the reactor was taken to its initial rating of 392 MW for an extended power run. A chronological outline of the test program is given in Table 1.1.

In general, the program started from the known reactivity point established by initial criticality and was carried out in such a manner that,
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<td>Loss of Load</td>
</tr>
<tr>
<td></td>
<td>Loss of Coolant Flow</td>
</tr>
<tr>
<td></td>
<td>Cooling by Natural Circulation</td>
</tr>
<tr>
<td></td>
<td>Xenon Worth</td>
</tr>
<tr>
<td></td>
<td>Instrumentation and Control Response</td>
</tr>
<tr>
<td></td>
<td>Dropped Control Rod Test</td>
</tr>
<tr>
<td>1/17/61</td>
<td>500 Hour Run at 392 Mw</td>
</tr>
<tr>
<td></td>
<td>Loss of Coolant Flow</td>
</tr>
<tr>
<td></td>
<td>Cooling by Natural Circulation</td>
</tr>
</tbody>
</table>
in each case, information was developed which would be required in succeeding phases. For example, the low power test work provided incremental control rod worth and temperature coefficient data which were required during the power tests to establish power coefficient, xenon worth, and the transient behavior of the reactor.

The moderator poisoning system was extremely valuable in establishing the nuclear characteristics of the core. In any reactor, the nuclear engineer is interested in determining the excess reactivity and the worth of control rods. In a core without a chemical poison system measurements can be made only at, or very near, one criticality point. An accurate determination of these parameters is therefore very difficult to make in most reactors since excess reactivity and rod worth are equal and opposite and of relatively unknown magnitude when the reactor is critical. A moderator poisoning system introduces a supplemental reactivity control which permits a variation of control rod position over a very wide range. Even though another unknown is introduced, this unknown can be evaluated experimentally, and the information which can be obtained on excess reactivity and rod worths is increased many-fold.

In general, the agreement between experimental results and analytical predictions was very good. No serious discrepancies were uncovered in excess reactivity values, control rod worth, boron worth, or kinetic characteristics. The excess reactivity of the clean core at operating temperature was roughly 1% higher than was predicted, but total control rod worth was sufficient to achieve the 3% shutdown required for that condition. Although some discrepancies existed in the predicted worth of individual groups of control rods, their magnitudes were compensating from the standpoint of total control rod worth. This total calculated was roughly 5% greater than the measured value. This difference has little effect on plant operation, and is primarily of interest in evaluating methods of reactivity calculations.

The measured values of all kinetic parameters of importance to operation (moderator, pressure, and power coefficients) compared favorably with calculated values. The measured value of moderator coefficient at high boron concentration was less negative than predicted, but the value was still substantially negative. Since initially the plant will not be operated at power with any appreciable boron concentration, this is not a serious discrepancy.
The transient portion of the startup test program confirmed the safe operation of the plant for accident transients such as loss of load and loss of two pumps at power followed by reactor scram and the subsequent loss of the two remaining pumps. In addition, the controlled response of the plant for normal rates of loading and unloading was demonstrated to be very stable.

While a direct comparison between predicted and actual responses is difficult because the analytical studies were based upon the most adverse combination of reactivity coefficients, in general, the agreement was satisfactory. The major discrepancy arose because of the assumptions employed in the steam generator model. In brief, the feedwater transients required to readjust the water level in the steam generator following a change in power were not included. Subsequent studies confirmed that these feedwater transients increased the stability of the plant for responses to load variations. However, for the condition of turbine trip and reactor scram, the experiments indicated that transients in feedwater flow caused more severe primary loop temperature transients than predicted. Consequently, additional control in the form of an automatic cutoff of feedwater flow for a scram and turbine trip transient was installed.

The pressurizer transient response showed that for rapid volume insurges the steam process was essentially isentropic initially. However, there was more heat removal from the steam than was predicted. Consequently, for slow transients the pressurizer process more closely corresponded to a saturated process.
2.0 DESCRIPTION OF REACTOR AND INSTRUMENTATION

2.1 Reactor Description

The Yankee pressurized light water reactor plant is rated to produce 485 MW of heat and 136 MW of net electrical power generation. The initial core, however, is designed to produce 392 MW of heat which provides approximately 110 MW of net electrical power generation.

The Yankee reactor has been described in detail in a number of previous reports \[3,4,5\]. Table 2.1 contains a summary of the pertinent design data of the core based on the final mechanical design, and the "as-built" fuel composition.

The reactor is fueled with slightly enriched uranium dioxide in the form of pressed and sintered cylindrical compacts stacked in stainless steel tubes in a core approximately 7.7 ft. in height and 6.3 ft. in diameter. The reactor is cooled and moderated by pressurized light water. The basic UO\(_2\) filled tubular fuel rod is bundled into fuel assemblies, containing either 304 or 305 rods. These assemblies are loaded vertically upon the core support plate to form a uniformly enriched, nearly cylindrical core of 76 fuel assemblies. The total UO\(_2\) mass is approximately 23,700 kg or 52,300 lbs. The initial fuel enrichment in the U-235 isotope is 3.4 wt. % to provide an estimated core life before refueling of 10,000 hours at 392 MWt. The cold volumetric composition of the reactor core is 34.7% fuel (UO\(_2\)), 49.1% water, 12.0% stainless steel, 2.2% control rods or control rod followers, 0.7% fixed shim rods, and 1 3% voids within the fuel rods. The average heat flux for the initial core design is 86,300 Btu per sq. ft-hr.; the maximum-to-average value for heat flux used as a design basis is 5.17.

Water in the main coolant system is normally maintained at a system pressure of 2,000 psig. The inlet water temperature to the 392 MW core is 490°F, and the average core outlet temperature is 532°F. These conditions yield 525 psig saturated steam at the outlet of the steam generator. The maximum fuel cladding surface temperature, based on the design value of maximum-to-average heat flux, is 663°F, causing some local boiling of the subcooled liquid within the coolant channel. No bulk boiling occurs under steady state conditions of this reactor.
The reactor is controlled at operating temperature by 24 neutron absorbing control rods. In order to control excess reactivity in the cold reactor, boric acid is dissolved in the coolant moderator.

2.2 Startup Rate Measurement

Several channels of nuclear instrumentation were available for measurement or observation of neutron flux level, and rate of change of flux level or startup rate. The channels provided for normal operation of the reactor consist of two startup range channels ($BF_3$ proportional counters with log count rate and startup rate indicators), two intermediate range channels (compensated ion chambers with log current and startup rate indicators), and 3 power range channels (uncompensated ion chambers with linear power level indicators). The two intermediate range channels also incorporate a current meter in the high voltage cable to the detector, which provides linear indications in the power range. A third compensated ion chamber is provided which has a linear current meter in the high voltage cable for additional indication in the power range. The compensated signal from this detector is not used during normal operation, but was connected to a micro-micro ammeter with a strip chart recorder, and used for startup rate measurement during the test program.

During the early test work, the startup rate measurements were made using the following channels:

a. The compensated ion chamber and micro-micro ammeter channel described above.

b. The two intermediate range log flux level channels with output recorded on a two pen strip chart recorder.

c. Another channel similar to (a) above, but having an uncompensated ion chamber.

d. Two special channels using $BF_3$ proportional counters with digital readout of counts for successive increments of time.

Data from these six channels were analyzed to determine the variation in the equilibrium startup rate obtained from the channels. The standard deviation on the average of all channels was found to be approximately 2%
of the average value of the startup rate. Based on the analysis of these measurements, it was decided to rely on the startup rate measurements made with the channel described in (a) above. It is felt that the startup rate measurements made in this manner have absolute accuracy of ± 3%, which includes an error of approximately 1% for reproducibility of the channel. Startup rate to reactivity conversion was accomplished using the methods and data given in Appendix B.

2.3 Control Rod Position Indication

Two methods of control rod position indication are available on the control board. The primary rod position indication is the same for each control rod and consists of thirty 3" transformers mounted around the pressure housing which contains the control rod drive shaft. The secondary winding of each transformer has a 2 volt incandescent bulb across it and these bulbs are mounted in a vertical column for each rod. As the control rod is withdrawn from the reactor, the control rod drive shaft passes up through the transformer windings and in so doing increases the magnetic coupling between the primary and secondary transformer windings which causes the bulbs to light in succession. This method indicates the position of the control rod drive shaft in the pressure housing within ± 3 inches. During the test work, this indicating scheme was used only to check that all the control rods in the same group were moving together.

The secondary rod position indication consists of a synchronous transmitter connected to the drive shaft which programs the sequence of coil operation for the magnetic jack rod mechanism. One revolution of this drive shaft results in a sequence of coil operations in each mechanism connected to this power supply so that the rod moves one 3/8 inch step. The rod mechanisms are connected so that they normally move in groups rather than individually and each of the six rod groups has a separate synchronous transmitter in its power supply with a receiver on the control board for digital readout of the group position. This system provides excellent accuracy for determining the distance which the rods move. Since the rods move in 3/8 inch steps, it is possible to determine the exact number of steps which the rods have moved. The mechanisms operate in such a way that a maximum ± 1/32 inch uncertainty exists in determining the amount of rod motion regardless of the number of control rod steps.
The initial position of the rods cannot be accurately determined with this system since it is sensitive to the amount that the rods should have moved and not to actual rod motion or actual rod position.

In order to indicate the magnitude of the uncertainty in absolute control rod position one should consider a typical sequence of operations involved in control rod worth measurements. These are outlined below:

1. After a scram, in the room temperature, atmospheric pressure condition, the maximum variation of the absorber sections with respect to one another is ±0.21 inches. In the fully inserted condition, position is 1-11/16 inches above the bottom of the average fuel column.

2. As the scram breakers are closed and the mechanisms are activated to move control rods out, the nature of the latching action and secondary position indication is such that the indicated rod position could vary plus one step minus 0 steps (+.375", -0") from the actual rod position.

Thus, the maximum uncertainty in the true control rod position compared to the indicated control rod position after several steps would be ±0.58 inch and -0.21 inch. This uncertainty existed whenever worth measurements were made. To this uncertainty one must add additional effects* if it is required to know the uncertainty in relative position between the rod tips and the bottom of the fuel.

The Yankee control rod consists of a 90 inch Ag-In-Cd section, a 3-3/4 inch stainless steel transition section, and a 94 inch zircaloy follower section. When the control position indication registers zero, the bottom of the Ag-In-Cd section is 1-1/16 inch above the bottom edge of the fuel and the stainless steel is 2-1/16 inches below the bottom edge of the fuel. In the figures of this report, the values of control rod height refer to the position indication, e.g. when control rod height is given as 25 inches, the bottom of the Ag-In-Cd section is at 26-11/16

* Pressurization from 15 to 2000 psig and increase in temperature from 70 to 514°F add +0.01 inch and +.14 inch respectively to the relative distance between rod tips and fuel bottom.
inches and the bottom of the steel transition piece is 23 inches from the bottom of the fuel. Since the "effective position" of a control rod with no transition piece will depend on the flux shape, no attempt has been made to define this position. The total control rod travel is 90 inches.

For a more detailed description of the rod drive mechanisms and the control rod position indication see Section 213 in Volume 1 of the "Yankee Atomic Electric Company Technical Information and Final Hazards Summary Report."  

2.4 Boron Concentration Measurements

During physics testing boron samples of main coolant water were taken in the Primary Auxiliary Building sample room. From this point one can obtain samples from the pressurizer of any of the four main coolant loops through their respective drain lines. Whenever low main coolant temperature conditions existed, samples were taken from the shutdown cooling system sampling points also located in the Primary Auxiliary Building.

Boron concentration was determined by the volumetric-Mannitol method. Boric acid is a weak acid and cannot be titrated directly with sodium hydroxide; therefore, an organic acid, Mannitol, is added to form a stronger complex acid that can be titrated against sodium hydroxide. The accuracy of this type of boron analysis is ± 1% of the boron concentration in the sample.

The determination of small boron changes, as required for the measurement of the reactivity worth of boron, was achieved by two methods. The first method involved adding clean water directly to the main coolant system, using the charging pumps, and noting the change in pressurizer level to determine the amount of clean water added. Knowing the amount of clean water added, it is straightforward calculation to obtain the boron change. The second method was done by adding clean water to the low pressure surge tank and then noting the flow integrator readings to determine amounts added. This method required a greater mixing time for the boron concentration in the main coolant system to
reach equilibrium than the first method. Using these methods, it was possible to make 10 to 20 ppm changes in the boron concentration and evaluate the magnitude of the change to within ± 0.5 ppm.

2.5 Temperature Indication

The four main coolant loops are each provided with 3 resistance type temperature detectors. The location of these detectors is the same in all 4 loops, and is as follows: one in the steam generator inlet (hot leg) and two in the steam generator outlet (cold leg).

The detector in the hot leg is connected to a narrow range readout circuit indicating on the main control board covering a range of 510° to 540°F, with a specified accuracy of ± 0.9°F. One of the cold leg detectors is connected to a narrow range readout circuit and the other to a wide range readout circuit, both indicating on the main control board. The cold leg wide range circuitry covers a range of 70° to 600° with specified accuracy of ± 18°F and the narrow range circuitry covers a range of 485 to 515°F with specified accuracy of ± 0.9°F. During the test work it was discovered that these resistance thermometers have a resistance change corresponding to a temperature change of approximately 2° due to self heating of the resistance element using normal plant circuitry.

A precision Leeds and Northrup recording resistance bridge was used during test work for accurate measurements over wide ranges. The bridge has an absolute accuracy of ± 0.003 ohms and covers a range of 0 to 100 ohms, in 1 ohm steps using an automatic range changer. The accuracy of this resistance measurement corresponds to an absolute accuracy of ± 0.1°F over the entire operating range. It is possible to determine temperature changes to ± 0.03°F with this instrument. By means of a plug connector the instrument can be used to measure either the cold or hot leg temperature in any loop. Discrepancies of several °F (0.1 ohms) were found from one temperature detector to another under what appeared to be isothermal conditions with no apparent explanation. It is believed that loop temperatures are known to ± 2°F and changes in individual loop temperature to ± 0.2°F. In addition, the data indicate that a loop temperature drop through the steam generator was obtainable
with an accuracy between ± 1°F and ± 2°F. For dynamic temperature measurements, an instrumentation channel time constant of 2.5 sec was assumed based upon past tests with similar equipment.

The core is also equipped with 27 chromel-alumel thermocouples located at the outlet of selected fuel assemblies. All 19 assemblies in one quadrant of the core are instrumented with these thermocouples. The remaining 8 thermocouples are located throughout the 3 remaining quadrants of the core. Twenty-four of the thermocouples read out on a recorder in the control room having a selected range of either 0-600°F or 500°F-600°F. The other three thermocouples are not normally recorded. Absolute accuracy of the thermocouples is ± 4°F. Temperature changes can be measured with an accuracy of ± 0.25°F in the 500 to 600°F range and ± 1.5°F in the 0 to 600°F range.

2.6 Pressure Indication

The main coolant system has a pressure detector located between the hot leg valve and the reactor vessel in one of the main coolant loops. The detector is a bourdon tube that produces a voltage output from an integrally mounted differential transformer. This output is sent to a magnetic amplifier from which a signal is sent to the main control board. The main control board indicator has a range of 0 to 3000 psig with a specified absolute accuracy of ± 90 psi. Small pressure changes can be measured with a specified accuracy of ± 10 psi.

The pressurizer has a pressure detector similar to the one described above, but the output is sent to a magnetic amplifier circuit and control board indication having a range of 1750 to 2500 psi and has a specified accuracy of ± 23 psi.

During the startup test program, comparisons between the wide range and narrow range pressure readings showed that the narrow range pressure channel had considerable hysteresis, between 30 and 50 psi. Therefore only the wide range channel readings were used to determine variations in pressure for the transient tests.

A bourdon tube type Heise gage located in the Primary Auxiliary Building connected to the main coolant sample header is the most accurate
means of obtaining main coolant pressure. The Heise gage has a 0 to 3000 psi range with an absolute accuracy of ± 1.0 psi.

Main coolant pressure may be determined from the temperature of the steam phase in the pressurizer which can be recorded on the Leeds and Northrup precision bridge recorder. This method was not used extensively during testing due to the uncertainties that are connected with the thermodynamics of the water-steam phases present in the pressurizer.

2.7 Pressurizer Level Indication

The pressurizer has both a wide range and a narrow range water level detector. The wide range pressurizer level indication was presented both on a control board meter and the process recorder in a range from 20 inches to 360 inches with a specified accuracy of ± 10 inches and a specified reproducibility of ± 1.7 inches. The narrow range pressurizer level indication was presented at the control board on a continuous strip chart recorder with an attached direct visual readout. This indication was over a range from 40 inches to 160 inches with a specified recorder and direct reading accuracy of ± 3.6 inches and a reproducibility of ± 0.6 inches. During the course of testing, comparisons between the wide range and narrow range level readings showed that considerable hysteresis existed in the narrow range-channel; on the order of 10 inches. Consequently, only the wide range readings were used to determine changes in water level in the pressurizer.

2.8 Flow Indication

The flow indications were obtained from the pressure drop across individual steam generators. The static accuracy of the pressure drop measurement was ± 4% over 0 to 40 psi range. In general, at the time of the test program the readings appeared to be on the low side. Actually, the flow indication in lbs/hr could not be used directly because of an error in the flow scale on the meter; instead pressure drop was read and converted to flow by the expression

\[ \frac{\text{lb/hr}}{} = 2.09 \times 10^6 \sqrt{\text{psig}}. \]

A measure of the dynamic behavior of the flow instrument channels was not available.
### Table 2.1

**YANKEE REACTOR DESIGN DATA**

<table>
<thead>
<tr>
<th><strong>General</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Average Core Diameter (cold)</td>
<td>75.4 in.</td>
</tr>
<tr>
<td>Active Core Length (cold)</td>
<td>91.9 in</td>
</tr>
<tr>
<td>Number of Fuel Assemblies</td>
<td>76</td>
</tr>
<tr>
<td>$\text{U}_2\text{O}_2$ in the Core</td>
<td>52,334 lbs.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Fuel Design Data (Cold Dimensions)</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Pellet Diameter</td>
<td>0.294 in.</td>
</tr>
<tr>
<td>Pellet Length</td>
<td>0.6 in.</td>
</tr>
<tr>
<td>Avg. $\text{U}_2\text{O}_2$ Density</td>
<td>10.18 g/cc</td>
</tr>
<tr>
<td>Fuel Tube Material</td>
<td>Type 348 S.S.</td>
</tr>
<tr>
<td>Fuel Tube O.D.</td>
<td>0.340 in.</td>
</tr>
<tr>
<td>Fuel Tube I.D.</td>
<td>0.298 in.</td>
</tr>
<tr>
<td>Fuel Tube Center-to-Center Pitch - Normal</td>
<td>0.422 in.</td>
</tr>
<tr>
<td></td>
<td>- In Line with Control Vanes 0.454 in.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Control Rod Data</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of Movable Control Rods</td>
<td>24</td>
</tr>
<tr>
<td>Control Rod Material</td>
<td>80% Ag, 15% In, 5% Cd.</td>
</tr>
<tr>
<td>Control Rod Shape</td>
<td>Cruciform</td>
</tr>
<tr>
<td>Control Rod Span</td>
<td>7.865 in.</td>
</tr>
<tr>
<td>Control Rod Thickness</td>
<td>0.265 in.</td>
</tr>
<tr>
<td>Number of Non-Movable Shims</td>
<td>8</td>
</tr>
<tr>
<td>Shim Material</td>
<td>Zircaloy-2</td>
</tr>
<tr>
<td>Shim Shape</td>
<td>Cruciform</td>
</tr>
<tr>
<td>Shim Span</td>
<td>7.865 in.</td>
</tr>
<tr>
<td>Shim Thickness</td>
<td>0.265 in.</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Heat Transfer Data</strong></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Total Heat Output</td>
<td>392 Mw</td>
</tr>
<tr>
<td>Number of Coolant Loops</td>
<td>4</td>
</tr>
<tr>
<td>Operating Pressure</td>
<td>2000 psig</td>
</tr>
</tbody>
</table>
Table 2.1 continued

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total Coolant Flow Rate</td>
<td>$3.78 \times 10^6$ lbs/hr.</td>
</tr>
<tr>
<td>Coolant Flow Rate (Through Core)</td>
<td>$3.4 \times 10^6$ lbs/hr.</td>
</tr>
<tr>
<td>Average Coolant Temperature (in Core)</td>
<td>$516^\circ$F</td>
</tr>
<tr>
<td>Core Inlet Temperature</td>
<td>$499^\circ$F</td>
</tr>
<tr>
<td>Average Coolant Rise in Core</td>
<td>$33^\circ$F</td>
</tr>
</tbody>
</table>

**Nuclear Design Data**

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium Loading</td>
<td>20,926 kg</td>
</tr>
<tr>
<td>U-235 Enrichment</td>
<td>3.4%</td>
</tr>
<tr>
<td>Fuel Lifetime at Full Power (392 MWt)</td>
<td>10,000 hrs.</td>
</tr>
<tr>
<td>Core Materials (active length) 100°F</td>
<td></td>
</tr>
<tr>
<td>$\text{UO}_2$</td>
<td>142,400 in$^3$</td>
</tr>
<tr>
<td>$\text{H}_2\text{O}$</td>
<td>201,700 in$^3$</td>
</tr>
<tr>
<td>Stainless Steel</td>
<td>49,400 in$^3$</td>
</tr>
<tr>
<td>Zircaloy (or control rods)</td>
<td>12,000 in$^3$</td>
</tr>
<tr>
<td>Void</td>
<td>5,100 in$^3$</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>410,600 in$^3$</td>
</tr>
</tbody>
</table>
3.0 LOW POWER PHYSICS TESTS

3.1 Subcritical Measurements

3.1.1 Core Loading

Because of the location of the normal instrumentation relative to the core - separated by approximately 30 inches of steel and water - the monitoring of core loading and the subsequent controlled dilution was accomplished using temporary internal instrumentation. Three BF$_3$'s were positioned in the irradiation holes in the core baffle flange as shown in Figure 3.1-a. Their associated electronic gear (pre amp, amplifier and scalers) were contained in instrument racks positioned in the vapor container.

During the loading, the detectors were moved to follow the loading of fuel. After the completion of the core loading and before the reactor vessel head was installed (calling for removal of the temporary instrumentation), the fully loaded core, borated to 2100 ppm in natural boron, was subjected to a series of dilution steps. Figure 3.1-a shows the position of the detectors for the dilution steps. It was hoped that these steps would yield values of effective core multiplication to be used as an indication of the boron concentration required for 5% cold shutdown with all control rods inserted. The following paragraphs serve to summarize the results and the exact methods that were used.

Two of the BF$_3$ detectors were cadmium wrapped while the third was not; this wrapping was done in order to minimize changing thermal absorption effects on the multiplication data caused by changing boron concentration in the reflector. Also, a comparison of the data obtained from the wrapped and unwrapped detectors gives a measure of the magnitude of the boron effect.

The dilution work over the range of 2100 to 1600 ppm was linearly extrapolated to a critical boron concentration of 900 ppm.

* Use of an analytical boron worth also allowed an evaluation of the boron needed for 5% cold shutdown with all control rods inserted.
with all control rods inserted in the cold condition (without Cd cover, the indicated critical concentration was 1100 ppm). The increase in boron concentration would tend to increase the extrapolated concentration for criticality. However, the decrease in boron attenuation (source detector - geometry effect) caused by dilution would decrease the extrapolated concentration for criticality. Figure 3.1-b shows typical extrapolations when the two effects are considered separately. Unfortunately, sufficient data were not available at the time of the experimental work to allow anything but a linear extrapolation.

Data obtained at a later time in the experimental program showed that the linear extrapolation to a critical condition has overestimated core reactivity by approximately 3%; thus, the source detector geometry effect was overriding relative to the increase in boron worth with decreasing boron concentration.

3.1.2 Rod Drop Determination of the Shutdown Multiplication

During the course of both hot and cold rod worth measurements, control rod drop tests were used to evaluate the shutdown multiplication as boron dilution - rod worth measurements proceeded. Rod drops from both programmed and banked critical configurations were made. These data provided a working qualitative picture of the shutdown available at a given boron concentration. Thus, it allowed proceeding to the next dilution step when coupled with incremental worth measurements and the dilution work mentioned in 3.2.1. The method of evaluation of the $k_{\text{shutdown}}$ was as follows:

$$1 - k_{\text{shutdown}} \sim \frac{n_0}{n(t)} (\Sigma \beta_{i,\text{eff}} e^{-\lambda_i t})$$

For a complete description of the development of this equation, the reader is referred to Reference 13. Values of $\beta_i$ and $\lambda_i$ were taken from Keepin's data [6] while evaluation of the flux ratio, $n_0/n(t)$, from the nuclear instrumentation channels was carried out at 15, 30, 45, and 60 sec. An average of these
LOCATIONS OF THE TEMPORARY INSTRUMENTATION FOR CORE LOADING

Fig. 3.1-a

LOCATIONS OF THE TEMPORARY INSTRUMENTATION FOR CORE LOADING

Fig. 3.1-b

SUBCRITICAL BORON DILUTION
four values was then used. As an example of the data obtained, the work done at 100°F can be summarized in the following table.

Table 3.1

<table>
<thead>
<tr>
<th>Chamber</th>
<th>Location</th>
<th>Indicated Boron Worth</th>
<th>( k_{\text{shutdown}} ) at 1050 ppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Bottom of Core</td>
<td>3%</td>
<td>0.92 - 0.94</td>
</tr>
<tr>
<td>2</td>
<td>Bottom of Core</td>
<td>3%</td>
<td>0.93 - 0.97</td>
</tr>
<tr>
<td>3</td>
<td>Bottom of Core</td>
<td>5%</td>
<td>0.94 - 0.98</td>
</tr>
<tr>
<td>4</td>
<td>Top of Core</td>
<td>6%</td>
<td>0.96 - 0.99</td>
</tr>
</tbody>
</table>

The indicated boron worth was obtained by subtracting \( k_{\text{shutdown}} \) values. These values compare to an integrated boron worth, over the same range of boron concentration from critical measurements of approximately 7\% and a shutdown \( k \) of approximately 0.95 at 1050 ppm.

The range of \( k_{\text{shutdown}} \), above, represents the spread of the data for various control rod drops. Because of the problems involved in sorting dynamic flux distribution effect, as well as geometry from multiplication effects (such as noted in Section 3.1.1), no error analysis has been performed. As indicated by the results obtained, this method is a qualitative safety technique and should not be interpreted as anything better.

3.1.3 Summary

In summary, cold subcritical measurements in the Yankee reactor proved to be quite useful in providing guidelines. However, it was found that quantitative information could not be obtained due to the difficulties in evaluation and separation of geometrical effects for the particular instrumentation arrangements that were available and because of transient flux distribution effects.
3.2 Zero Power Reactivity Measurements

3.2.1 Experiments at Ambient Temperature

The first series of experiments in which criticality was maintained was run at a low reactor power level (< 3 MWt) and ambient temperature (~100°F). These experiments were performed following initial criticality. The objective of these experiments was to determine boron worth, banked control rod worth, zero power moderator temperature coefficient, and an estimate of the boron concentration required for 5% cold shutdown with all control rods fully inserted. The temperature coefficient measurements are discussed in Section 3.3. The following paragraphs describe the remaining cold core measurements.

Banked Control Rod Measurements

The experimental work was begun at the boron concentration required for criticality with all control rods removed from the core (1900 ppm). Data were obtained at a series of fixed boron concentrations starting from 1900 ppm. Values of the critical banked control rod position*, and the incremental reactivity worth of banked control rods (Δρ/Δh), were obtained in steps of decreasing boron concentration. The size of the steps varied between 50 and 100 ppm. Figure 3.2 shows boron concentration as a function of the critical banked control rod position, and Figure 3.3 shows (Δρ/Δh) banked as a function of the critical banked control rod position. Calculated values are also shown in the figures. It is seen that the agreement on Δρ/Δh is very good while the calculated boron concentration vs. just critical banked control rod height are higher than those determined experimentally. This difference appears to be primarily due to a difference in boron worth between theory and experiment, as will be discussed later in this section.

Split Group Control Rod Worth Measurements

During the boron reduction, critical control rod heights and values of Δρ/Δh were also obtained for the even control rod

* As mentioned in Section 2.3, rod positions given in this report refer to a zero rod height that is 1-11/16" above the bottom of fuel.
BORON CONCENTRATION FOR CRITICALITY AT 100°F WITH ALL CONTROL RODS BANKED

Fig. 3.2
INCREMENTAL ROD WORTH AT 100°F WITH ALL CONTROL RODS BANKED

Fig. 3.3
groups (2, 4, 6) with the odd control rod groups (1, 3, 5) at 90 in. (fully withdrawn), and the odd groups, with the even groups at 90 inches. These are referred to as the "split group" control rod measurements. The purpose of these measurements was to obtain a more accurate indication of total control rod worth in the cold core than would be available from the total control rod bank data because of the difficulty in extrapolating below the last critical control rod position. Figure 3.4 shows boron concentration versus critical split control rod group position for both the odd and even control rod groups. The even groups have a boron equivalent of 655 ppm, while the odd groups have a boron equivalent of 685 ppm. Figures 3.5 and 3.6 show values of \( \Delta \rho/\Delta h \) for the odd and even groups, respectively. The curves of \( \Delta \rho/\Delta h \) can be graphically integrated to yield a reactivity change during group insertion. The results of such an integration are shown in the following table

<table>
<thead>
<tr>
<th>Control Rod Groups</th>
<th>Total Worth ( % \Delta \rho )</th>
<th>( C_B ) initial ppm</th>
<th>( C_B ) final ppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>1, 3, 5</td>
<td>(5.53 ± 0.41)</td>
<td>1895 ± 19</td>
<td>1210 ± 12</td>
</tr>
<tr>
<td>2, 4, 6</td>
<td>(5.03 ± 0.35)</td>
<td>1895 ± 19</td>
<td>1240 ± 12</td>
</tr>
</tbody>
</table>

The above data show the core is shut down by at least 5% with 1200 ppm and all rods inserted*. The effect of a lower boron concentration and control rod interaction will both serve to increase this shutdown value. As discussed later in the report, control rod interaction effects will increase total rod worth by 2 to 3\% in \( \Delta \rho \) and the rod worth at 1200 ppm will be greater than at the boron concentrations measured because of the increase in

* Based on the hot rod worth data, a best estimate of the shutdown boron concentration is 1050 ppm. However, uncertainties in measured quantities increase this to 1200 ppm in order to be certain that at least 5% shutdown is available.
BORON CONCENTRATION FOR CRITICALITY AT 100°F FOR SPLIT CONTROL ROD GROUPS

Fig. 3.4
INCREMENTAL WORTH FOR SPLIT GROUP 135 AT 100°F

Fig. 3.5
INCREMENTAL WORTH OF SPLIT GROUP 2 4 6 AT 100°F

Fig. 3.6
thermal neutron diffusion length with a reduction in boron concentration. A detailed discussion of hot shutdown is included in Section 3.2.3.

Boron Worth Measurements

The reactivity worth of boron in the cold core was determined by two methods. Direct measurements were made in which the change in reactor startup rate caused by a small boron dilution step was converted to a reactivity change. The results of these measurements are shown in Table 3.3.

Mixing problems, and an accurate determination of the amount of clean water added to the primary system in initial measurements, indicated that it would be desirable to have an alternate method. Indirect determinations were also made by plotting the integral of $\Delta \rho / \Delta h$ for both the full rod bank and split groups as a function of boron concentration and determining the slope of the curve. Table 3.4 shows the results of these indirect boron worth determinations, the range of boron worth over which the measurement of integrated reactivity applied, and the average boron concentration.

Table 3.4

<table>
<thead>
<tr>
<th>Control Rod Configuration</th>
<th>Total Worth ($% \Delta \rho$)</th>
<th>Range of $C_B$ ppm</th>
<th>$C_B$ Avg. ppm</th>
<th>$\Delta \rho / \Delta C_B$ $\times 10^4$ $\rho / \text{ppm}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Full Bank 90°-17&quot;</td>
<td>6.21</td>
<td>1895 - 1090</td>
<td>1490</td>
<td>0.772 ± 0.052</td>
</tr>
<tr>
<td>Groups 1, 3, 5</td>
<td>5.52</td>
<td>1895 - 1210</td>
<td>1550</td>
<td>0.808 ± 0.030</td>
</tr>
<tr>
<td>Groups 2, 4, 6</td>
<td>5.03</td>
<td>1895 - 1240</td>
<td>1570</td>
<td>0.768 ± 0.030</td>
</tr>
</tbody>
</table>

Although the experimental data (both direct and indirect) do indicate a trend of increasing boron worth with decreasing boron concentration, the accuracy of the data is not sufficient to warrant a positive conclusion in this respect. Figure 3.7 which is
<table>
<thead>
<tr>
<th>Avg $C_B$ ppm</th>
<th>Rod Group Position Inches</th>
<th>$\Delta C_B$ ppm</th>
<th>$\Delta \rho / \Delta C_B \times 10^4$ ρ/ppm</th>
</tr>
</thead>
<tbody>
<tr>
<td>1680</td>
<td>40-4/8 40-4/8 40-7/8 40-7/8 40-7/8 40-7/8</td>
<td>12.1</td>
<td>0.797 ± 0.025</td>
</tr>
<tr>
<td>1530</td>
<td>30-6/8 30-6/8 30-6/8 30-6/8 30-6/8 30-6/8</td>
<td>10.6</td>
<td>0.820 ± 0.034</td>
</tr>
<tr>
<td>1090</td>
<td>15-3/8 15-6/8 15-6/8 15-6/8 15-6/8 90</td>
<td>10.2</td>
<td>0.762 ± 0.069</td>
</tr>
<tr>
<td>1800</td>
<td>90 44-2/8 90 90 90 90</td>
<td>10.0</td>
<td>0.668 ± 0.036</td>
</tr>
</tbody>
</table>

Table 3.3
DIRECT MEASUREMENT OF BORON WORTH AT AMBIENT TEMPERATURE
a comparison of the experimental values with calculations, indicates the variation of $\Delta \rho / \Delta C_B$ which might be expected. It also indicates that the experimental values are roughly 10% greater than is indicated by the calculations in the range where most data is available. The possible source of this discrepancy is discussed in Section 3.2.3.

3.2.2 Zero Power Reactivity Experiments at Operating Temperature

A series of experiments similar to those described in the previous section was also performed at a moderator temperature of 514°F. Essentially the same data were obtained as in the ambient temperature work with the addition of individual control rod group worth in the referenced control rod program sequence. Since rod worth increases with temperature, while core reactivity decreases with temperature, the boron concentration could be reduced to zero, and measurements were made over the entire range of boron concentration.

Banked Control Rod Worth Measurements

As in the 100°F work, measurements started at the boron concentration required for criticality with all control rods removed from the core (1707 ppm). Data were obtained at a series of fixed increments in boron concentration. Figures 3.8 and 3.9* show boron concentrations and $(\Delta \rho / \Delta h)_{BANK}$ as a function of critical banked control rod position. Calculated values of the same quantities are also shown on the figures.

Control Rod Group Worth Measurements

In addition to the full control rod bank data, measurements of boron equivalents and $\Delta \rho / \Delta h$ were also made for the individual control rod groups in the same group withdrawal sequence which will

* You will note that the total worth from the analytical prediction is lower than the experimental total worth although Table 3.10 indicates the reverse effect. This is caused by the fact that the analytical prediction is based on a one dimensional calculation that underestimates rod worth by 3-4% $\Delta \rho$. Section 3.2.3 contains a complete discussion of this problem.
BORON CONCENTRATION FOR CRITICALITY AT 514°F WITH ALL CONTROL RODS BANKED

Fig. 3.8
INCREMENTAL ROD WORTH AT 514°F WITH ALL CONTROL RODS BANKED

Fig. 3.9
be used in operation. The group arrangements for the 24 control rods are shown in Figure 3.10. The reference control rod program calls for withdrawal of the groups in the order 6, 4, 2, 3, 5, 1. Figure 3.11 shows boron concentration as a function of control rod group position in the reference withdrawal sequence. Figures 3.12 through 3.17 show the plots of $\Delta \rho/\Delta h$ vs. control rod group heights for groups 1, 5, 3, 2, 4, and 6. Since the reactor is subcritical at 514°F with group 6 inserted at operating temperature, the group 6 worth measurement was made with groups 2 and 4 removed. Estimates of group 6 worth with groups 1-5 inserted are discussed in a later section. Table 3.5 shows the worth of each control rod group as determined by graphical integration of the $\Delta \rho/\Delta h$ vs. h curves, as well as the boron equivalent of each group, and the average boron concentration over which the worth was measured. Two experimental worth values are given. The first was obtained directly from the curve of $\Delta \rho/\Delta h$ vs. control rod group height. The second was obtained by integrating the total control rod band $\Delta \rho/\Delta h$ curve over the boron concentration limits corresponding to full removal and full insertion of the group. The accuracy values are representative of the accuracy of the points on the $\Delta \rho/\Delta h$ curves in the region of highest incremental rod worth.

It is interesting to observe the interaction effects which exist in control rod worth values. Rod groups 1 and 3 are identical in location and number of rods to groups 2 and 4. Group 2 added to groups 1, 3, and 5 inserted has less than 60% of its worth when it is added to a core with no other rods inserted. In a similar manner, the worth of group 4 with groups 1, 2, 3, and 5 inserted is 10% less than its worth with only groups 2 and 5 inserted. This change in worth arises from the difference in the flux distribution in the core for the two cases.
CONTROL ROD GROUP POSITIONS IN THE YANKEE CORE

Fig. 3.10
BORON CONCENTRATION FOR CRITICALITY AT 514°F FOR ROD WITHDRAWAL IN THE REFERENCE CONTROL ROD PROGRAM

Fig. 3.11
Groups 2-6 @ 90°

Incremental worth of control rod group 1 at 514°F

Fig. 3.12
CRITICAL ROD POSITION (INCHES)

INCREMENTAL WORTH OF CONTROL ROD GROUP 5 AT 514°F

GR. 1 AT 0°
GR. 6, 4, 2 & 3 AT 90°

Fig. 3.13
INCREMENTAL ROD WORTH OF CONTROL ROD GROUP 3 AT 514°F

Fig. 3.14
INCREMENTAL ROD WORTH OF CONTROL ROD GROUP 2 AT 514°F

Fig. 3.15
INCREMENTAL ROD WORTH OF CONTROL ROD GROUP 4 AT 514°F

Fig. 3.16
INCREMENTAL WORTH OF CONTROL ROD GROUP 6 AT 514°F

Fig. 3.17
Table 3.5
CONTROL ROD GROUP WORTHS (\%\Delta_\rho) IN THE ZERO
POWER YANKEE CORE AT OPERATING TEMPERATURE

<table>
<thead>
<tr>
<th>Rod Group*</th>
<th>C_B Avg. ppm</th>
<th>\Delta C_B ppm</th>
<th>Integral %\Delta_\rho</th>
<th>Bank Integral %\Delta_\rho</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1569</td>
<td>276</td>
<td>1.74 ± 0.08</td>
<td>1.71</td>
</tr>
<tr>
<td>2</td>
<td>1276</td>
<td>319</td>
<td>1.98 ± 0.17</td>
<td>1.95</td>
</tr>
<tr>
<td>3</td>
<td>887</td>
<td>449</td>
<td>3.01 ± 0.15</td>
<td>3.11</td>
</tr>
<tr>
<td>4</td>
<td>586</td>
<td>153</td>
<td>.98 ± .10</td>
<td>.97</td>
</tr>
<tr>
<td>5</td>
<td>308</td>
<td>404</td>
<td>2.82 ± .18</td>
<td>2.86</td>
</tr>
<tr>
<td>6</td>
<td>502</td>
<td>321</td>
<td>2.19 ± .15</td>
<td>2.15</td>
</tr>
</tbody>
</table>

Group 6 Worth

Group 6 worth is of particular importance since it provides the information necessary to calculate total rod worth and the hot shutdown margin. Total rod worth is obtained by adding group 6 worth to the summation of the worths of groups 1 through 5. Shutdown is known from the rod worth "remaining" in group 6 from the critical position at zero boron, 514°F and 2000 psig to complete insertion.

The experimental determination of this worth was carried out in the following manner:

1. Incremental worth measurements were made from complete removal to 51.4" of withdrawal with supplemental control provided by boron. This rod position is critical at 514°F, 2000 psig and zero boron.

2. By lowering main coolant temperature, additional incremental worth data were obtained to 27" of removal. ** Integration of the incremental worth data from 1 above, plus this information, gave rod worth from 27 to 90".

* All groups measured in programmed sequence (1, 5, 3, 2, 4, 6) except 6, which was measured with groups 2 and 4 removed.

** This rod position corresponds to the temperature for which calculations supplemented by experimental data indicated 3% shutdown.
3. Based on the methods outlined below, worth was obtained from 27" to 0".

Figure 3.18 shows the results of the incremental worth determination while Table 3.6 summarizes the findings. Before proceeding with a discussion of the methods, it should be pointed out that the region from 0 to 27" on group 6, with groups 1-5 inserted, is a region where no critical measurements can be made due to safety or shutdown margin considerations.

Several techniques are available for the determination of this remaining rod worth. The most obvious technique is to use data measured in a subcritical reactor. Although these determinations are handicapped by a dependence on detector geometry, they do provide an indication of rod worth. Since previous experience with the rod drop technique (See Section 3.2) indicated that dynamic subcritical measurements were not applicable, only static determinations were made. One such subcritical multiplication measurement involved movement of the entire group 6 and yielded 4.0% $\Delta \rho$ for the worth and 3.2% $\Delta \rho$ for shutdown. Another involved a subcritical approach on one rod of group 6 (rod 20) with all other rods inserted, from which a shutdown of 3.2% was inferred.

A second method involves the use of the shapes of previously measured incremental worth curves. This assumes that the percentages of rod worth above and below the peak incremental worth are amenable to a simple functional relationship with total worth. This function, called the shape factor, is defined by:

$$
F = \text{Shape Factor} = \frac{\int_{0}^{90} \frac{\Delta \rho}{\Delta h} dh}{\int_{0}^{90} \frac{\Delta \rho}{\Delta h} dh} = \frac{\text{worth above maximum } \Delta \rho/\Delta h}{\text{total worth}}
$$

* Because of the large rod interaction effects one cannot balance group 6 against another group and hope to get meaningful results. This was verified by measurements made by balancing group 6 against groups 1 and 5. This technique yields 1.4% $\Delta \rho$ for group 6 worth while group 6 worth with 1, 3, and 5 inserted was known to be worth 2.19% $\Delta \rho$. 

-50-
Incremental Rod Worth of Control Rod Group 6 at 514°F

**Fig. 3.18**
where \( x \) is the location of the peak incremental rod worth. The measured group rod worths formed the basis for the evaluation of the functional relationship between \( F \) and the total rod worth. The initial groups inserted (1, 3, and 5) have a relatively large percentage of the worth above the peak. Those added to 1, 3, 5 have lower percentages above the peak. Since it is desired to obtain the worth of 6 added to 1-5, we infer a functional relationship as shown in Figure 3.19 - biased toward data obtained with 1, 3, 5 inserted. If the assumption is made that the peak incremental worth occurs at the last measured point on group 6 with groups 1-5 inserted, the integral from \( x \) to 90 of \( \Delta \rho / \Delta h \) for group 6 is defined, equal to 2.65% \( \Delta \rho \). A trial and error solution, involving the shape factor, was then performed. This yielded a shape factor of 0.66 and a total rod worth of 4.0% \( \Delta \rho \) for group 6.

If the peak incremental worth occurs at a lower point in the core, a larger total worth would result. Measurements performed on group 6 with group 4 at 12" and all others at 0" indicate that the incremental worth is increasing at 23" (see Figure 2.18). Since the removal of group 4 to 12" should decrease the value of group 6 incremental worth due to a flux shift away from the moving group, these data provide the information necessary to estimate group 6 worth at 23". This was done by using the measured \( \Delta \rho / \Delta h \) with 4 at 12" and a known ratio of the incremental worth of group 6 with 1-5 at 0 to the increased worth of group 6 with 4 at 12". Integration of the inferred group 6 incremental worth curve to 23 inches, coupled with a trial and error solution involving the shape factor, yielded 4.9% \( \Delta \rho \) for group 6 worth.

In addition to these extrapolations, which were based on data obtained during individual group worth measurements, split group and all rods banked data were used to extrapolate total worth, shutdown, and group 6 worth. The use of the hot split group data is based on the addition of the worth of 1, 3, 5 with 2, 4, 6 at 90 inches to 2, 4, 6 with 1, 3, 5 at 90 inches. A total worth of 13.6% \( \Delta \rho \) is obtained; adjustment for boron increases
NUMBERS ASSOCIATED WITH POINTS REFER TO THE ROD GROUP(S) THAT WAS MEASURED IN THE PROGRAM SEQUENCE (1, 5, 3, 2, 4, 6) OF INSERTION.

SHAPE FACTOR FOR VARIOUS CONTROL ROD GROUPS

Fig. 3.19
this to $14.0\% \Delta \rho$. In addition, an interaction correction must be applied to the total worth obtained in this manner. This interaction factor was obtained by a comparison of the total worth with all rods banked, to the summation of the split group worths.

$$f = \text{Interaction factor} = \frac{\left[ x \int_{90}^{\text{Bank}} \frac{\Delta \rho}{\Delta h} \, dh \right]}{\left[ x \int_{90}^{\text{1,3,5}} \frac{\Delta \rho}{\Delta h} \, dh \right] + \left[ x \int_{90}^{\text{2,4,6}} \frac{\Delta \rho}{\Delta h} \, dh \right]}$$

Calculations of this value at the lowest measured critical position with all rods banked (9.6") yielded an interaction factor of 1.02. This leads to $14.3\% \Delta \rho$ for total rod worth, $3.1\% \Delta \rho$ shutdown, and $3.8\% \Delta \rho$ for group 6 worth. These values of total worth are minimum values since the interaction factor, $f$, was increasing rapidly as the all rods banked critical position decreased towards the last measured point. However, quantitative information is not available to show how much larger this factor should be.

Finally total worth, shutdown, and group 6 worth can be estimated from the all rods banked data. The functional relationship involving the shape factor cannot be extrapolated to the total worth with any confidence so this method was not used. Rather, the extrapolation was based on data obtained by balancing either, groups 1-5 against group 6, or the banked against a rod of group 6. The interaction effects that make this technique invalid for programmed rod patterns are smaller, though not negligible, for the all rods banked condition. Use of this method yields a total worth of $15.3\% \Delta \rho$ when done against group 6, and $14.5\% \Delta \rho$ when done against a rod of group 6. Group 6 worths and shutdown margins may be obtained from these total worth values by making use of critical rod position data for the banked and programmed rod configurations, as well as hot banked rod integrals. Table 3.6 provides a complete tabulation of results.
Table 3.6

SUMMARY OF GROUP 6 WORTH, SHUTDOWN MARGIN, AND TOTAL ROD WORTH

<table>
<thead>
<tr>
<th>Method</th>
<th>Gr. 6 Worth</th>
<th>Shutdown</th>
<th>Total Worth</th>
</tr>
</thead>
<tbody>
<tr>
<td>A. Inferred from programmed rod measurements</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Subcritical measurements</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>a. Movement of Gr. 6</td>
<td>4.0</td>
<td>3.2</td>
<td>14.5</td>
</tr>
<tr>
<td>b. Movement of 1 rod in Gr. 6</td>
<td>---</td>
<td>3.2</td>
<td>---</td>
</tr>
<tr>
<td>2. Shape Factor Analysis</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>a. with peak $\Delta \rho/\Delta \theta$ of Gr. 6 at 27&quot;</td>
<td>4.0</td>
<td>3.2</td>
<td>14.5</td>
</tr>
<tr>
<td>b. with peak $\Delta \rho/\Delta \theta$ of Gr. 6 at 23&quot;</td>
<td>4.9</td>
<td>4.1</td>
<td>15.4</td>
</tr>
<tr>
<td>B. Inferred from split group</td>
<td>3.8</td>
<td>3.0</td>
<td>14.3</td>
</tr>
<tr>
<td>C. Inferred from all rods banked measurements</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1. Balancing rods of Gr. 6 against Gr. 1 - 5</td>
<td>4.8</td>
<td>4.0</td>
<td>15.3</td>
</tr>
<tr>
<td>2. Balancing 1 rod of Gr. 6 against Gr. 1 - 6</td>
<td>4.0</td>
<td>3.2</td>
<td>14.5</td>
</tr>
</tbody>
</table>
In summary, several methods have been used to determine group 6 worth, shutdown, and total worth. Most of them involve assumptions which are not easily verified, however, one can say that the split group estimate provides minimum values. The shape fit, as dictated by Figure 3.19, with a peak group 6 $\Delta \rho/\Delta h$ at 23" appears to fit all of the experimental facts that are available and, therefore, must be considered as yielding best values.

**Split Group Control Rod Worth Measurements**

Measurements of the odd and even numbered control rod groups were performed in the core at operating temperature in the same manner as at ambient temperature (see previous section). These measurements are shown in Table 3.7. Comparison with values obtained at ambient temperature may be made to obtain the increase in control rod worth due to increasing temperature.

**Table 3.7**

<table>
<thead>
<tr>
<th>Control Rod Group</th>
<th>$C_B$ Avg. ppm</th>
<th>$% \Delta \rho$ (514°F)</th>
<th>$% \Delta \rho$ (100°F)</th>
<th>Ratio of Hot to Cold Rod Worths</th>
</tr>
</thead>
<tbody>
<tr>
<td>1, 3, 5</td>
<td>1185</td>
<td>6.96 ± 0.42</td>
<td>5.52 ± 0.41</td>
<td>1.26</td>
</tr>
<tr>
<td>2, 4, 6</td>
<td>1201</td>
<td>6.64 ± 0.39</td>
<td>5.03 ± 0.35</td>
<td>1.32</td>
</tr>
<tr>
<td>Full Bank (90&quot; to 17&quot;)</td>
<td>1083</td>
<td>8.1 ± 0.46</td>
<td>6.2 ± 0.42</td>
<td>1.31</td>
</tr>
</tbody>
</table>

These values compare favorably with a calculated control rod worth ratio of 1.23 obtained from banked control rod calculations in both the cold and hot cores.

**Boron Worth Measurements**

Both direct and indirect measurements of boron worth were obtained in the core at operating temperature using the same procedures as those described previously for the ambient temperature measurements. Tables 3.8 and 3.9 list the direct and indirect...
Table 3.8

DIRECT MEASUREMENT OF BORON WORTH AT OPERATING TEMPERATURE

<table>
<thead>
<tr>
<th>Avg $C_B$ (ppm)</th>
<th>Rod Group Position Inches</th>
<th>$\Delta C_B$ (ppm)</th>
<th>$\Delta \rho/\Delta C_B \times 10^4$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1686</td>
<td>78-3/8 90 90 90 90 90</td>
<td>8.25</td>
<td>0.602 ± 0.030</td>
</tr>
<tr>
<td>1678</td>
<td>72-0/8 90 90 90 90 90</td>
<td>7.65</td>
<td>0.630 ± 0.025</td>
</tr>
<tr>
<td>1095</td>
<td>0 90 77-5/8 90 0 90</td>
<td>10.7</td>
<td>0.654 ± 0.021</td>
</tr>
<tr>
<td>652</td>
<td>0 78-3/8 0 90 0 90</td>
<td>11.3</td>
<td>0.701 ± 0.035</td>
</tr>
<tr>
<td>652</td>
<td>0 81-0/8 0 90 0 90</td>
<td>11.3</td>
<td>0.716 ± 0.035</td>
</tr>
</tbody>
</table>

Table 3.9

INDIRECT BORON WORTH DETERMINATIONS AT 514°F

<table>
<thead>
<tr>
<th>Control Rod Configuration</th>
<th>Range of $C_B$ (ppm)</th>
<th>Avg $C_B$ (ppm)</th>
<th>$\Delta \rho/\Delta C_B \times 10^4$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bank 90&quot; → 11.8&quot;</td>
<td>1707 - 0</td>
<td>854</td>
<td>0.671 ± 0.049</td>
</tr>
<tr>
<td>Split Group, 1,3,5</td>
<td>1707 - 663</td>
<td>1185</td>
<td>0.667 ± 0.043</td>
</tr>
<tr>
<td>Split Group, 2,4,6</td>
<td>1707 - 695</td>
<td>1201</td>
<td>0.656 ± 0.043</td>
</tr>
<tr>
<td>Programmed Rods</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>1707 - 1431</td>
<td>1569</td>
<td>0.630 ± 0.062</td>
</tr>
<tr>
<td>5</td>
<td>1431 - 1112</td>
<td>1272</td>
<td>0.621 ± 0.066</td>
</tr>
<tr>
<td>3</td>
<td>1112 - 663</td>
<td>887</td>
<td>0.670 ± 0.040</td>
</tr>
<tr>
<td>2</td>
<td>663 - 510</td>
<td>586</td>
<td>0.641 ± 0.078</td>
</tr>
<tr>
<td>4</td>
<td>510 - 106</td>
<td>308</td>
<td>0.698 ± 0.053</td>
</tr>
<tr>
<td>Gr. 6 Added to 1,3,5</td>
<td>663 - 342</td>
<td>502</td>
<td>0.682 ± 0.058</td>
</tr>
<tr>
<td>Gr. 6 Added to 1 - 5</td>
<td>106 - 0</td>
<td>53</td>
<td>0.755 ± 0.081</td>
</tr>
</tbody>
</table>

*All boron concentrations are known to ± 1% of the concentration.
measurements. The range of boron concentration values listed in table 3.9 defines the fully withdrawn and fully inserted critical boron concentrations for the control rod configurations listed. The integrated group worth values at operating temperature are shown in Figure 3.20. Figure 3.21 compares these values with calculated results. Since measurements over a much wider range were performed, it was possible to observe a definite increase in the value of $\Delta \rho / \Delta C_B$ with decreasing boron concentration. The same order of discrepancy (~10%) between calculation and experiment as was observed in the ambient temperature work is also present at operating temperature, and will be discussed in the following section.

3.2.3 Discussion of Results

A comparison of the experimental results presented in the previous paragraphs with calculation indicates very good agreement for the most part. None of the deviations from predicted results gives rise to any significant changes in plant operation procedure or performance predictions. In attempting to assess the source of the observed discrepancies, problems arise in quantitatively separating out the effects of the many variables involved, particularly where deviations are not large.

The most straightforward method of comparison is to perform calculations on critical configurations which were achieved experimentally and evaluate the deviation of $k_{\text{eff}}$ from unity. This gives an accurate evaluation of the gross difference between experiment and calculation, but presents difficulties in establishing the source of this difference. In comparing calculation with experiment for dynamic measurements in which reactivity effects resulting from changes in one core parameter are measured, uncertainties in the startup rate measurements and startup rate-to-reactivity conversion relationship, as well as computational diffusion theory codes in which the reactivity effect is a small difference between large numbers, prevent accurate quantitative
INTEGRATED ROD WORTHS IN THE PROGRAMMED WITHDRAWAL SEQUENCE

Fig. 3.20
comparisons. The following paragraphs present a discussion of comparisons which have been made for boron worth, control rod worth, and excess reactivity, where both static and dynamic experiments are considered.

Boron Worth

The major discrepancy between calculation and experiment which appeared in the experiments is that exhibited by boron worth. Figures 3.7 and 3.21 indicate the deviation to be roughly 10%. A similar effect was also observed during the proof test critical experiments for the BR3 reactor, which has very similar nuclear characteristics.

The experiments indicate a greater neutron absorption rate in boron relative to uranium than would be indicated in the calculations. The most likely source of the discrepancy is the assumption, made in the calculations, that the spectrum of neutrons to which the fuel and moderator are exposed is the same. Account is taken of the neutron flux change, but not the spectrum change which occurs between fuel and moderator. A twenty percent difference in the average neutron energy between fuel and moderator would be sufficient to account for the difference between calculation and experiment, and a difference of this magnitude would not be unreasonable. The variation in $\Delta \rho/\Delta C_B$ with boron concentration appears to be predicted within the accuracy limits of the measurements.

Control Rod Worth

The agreement between the calculated and experimental $\Delta \rho/\Delta h$ vs. banked control rod position curves shown in Figures 3.3 and 3.9 appears to be very good. The calculated curves are based on one-dimensional diffusion theory which includes two approximations which are known to involve some error, but the errors are expected to be compensatory in the range of banked control rod heights studied experimentally. The assumptions involved are:
a. Representation of the cruciform shaped control rod by a slab control rod, with the same perimeter and cross sectional area. This assumption tends to overestimate control rod worths.

b. Representation of the distribution of the control rods in the core as uniform, whereas in the actual core, there are more rods in the central region of the core and fewer rods in the peripheral region of the core. This assumption tends to underestimate control rod worth.

At low control rod bank positions, the second assumption predominates while at high banked positions, the first assumption is more important. These effects, as well as the underestimate of excess reactivity and boron worth in the core at operating temperature, may cause the cross-over observed in the critical boron concentration vs. banked control rod position curve at operating temperature.

In both the ambient and operating temperature experiments, the calculated boron worth is low, so that the predicted boron concentration for full bank removal is higher than observed in the experiment. This difference becomes less pronounced as boron concentration is reduced, causing the two curves to come together. At intermediate control rod heights the effect of the increase in radial leakage as rod height is reduced counteracts the underestimate of boron worth. Since a given change in banked control rod height produces a greater reactivity change, the slope of the curve $C_B$ as a function of banked control rod height is also greater. The changing magnitude of these two effects in the calculations may explain the "crisscrossing" observed between experiment and theory. In view of the assumptions of the calculations the agreement is excellent, since the reactivity difference between the curves is less than about 1% over the entire range of boron concentration studied.
The calculated worths for individual control rod groups in the core at operating temperature were performed with two-
dimensional calculations which treat the control rods explicitly. Table 3.10 shows a comparison of these calculations with exper-
imental values for several control rod groups and combinations of
groups. Since the calculations were performed for unborated cores, the experimental worth values are adjusted to account for the reduction in measured worth due to the presence of boron. Calculated values for large numbers of control rods are in excellent agreement, but discrepancies exist for individual group worths where insertion occurs in cores which contain 10 or more control rods.

Table 3.10

<table>
<thead>
<tr>
<th>Configuration</th>
<th>Measured $C_B$ Avg ppm</th>
<th>Measured Worth - $%\Delta\rho$ Uncorrected</th>
<th>Measured Worth - $%\Delta\rho$ Corrected</th>
<th>Calculated Worth - $%\Delta\rho$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1569</td>
<td>1.74 ± .08</td>
<td>1.84 ± .08</td>
<td>- - - - - -</td>
</tr>
<tr>
<td>5</td>
<td>1276</td>
<td>1.98 ± .17</td>
<td>2.06 ± .18</td>
<td>- - - - - -</td>
</tr>
<tr>
<td>3</td>
<td>887</td>
<td>3.01 ± .15</td>
<td>3.12 ± .16</td>
<td>3.07</td>
</tr>
<tr>
<td>2</td>
<td>586</td>
<td>0.98 ± .10</td>
<td>1.01 ± .10</td>
<td>0.75</td>
</tr>
<tr>
<td>4</td>
<td>308</td>
<td>2.82 ± .18</td>
<td>2.86 ± .18</td>
<td>2.61</td>
</tr>
<tr>
<td>$6 \text{ /a}$</td>
<td>---</td>
<td>4.90 ± .45</td>
<td>4.90 ± .45</td>
<td>6.10</td>
</tr>
<tr>
<td>$6 \text{ /b}$</td>
<td>502</td>
<td>2.19 ± .15</td>
<td>2.24 ± .15</td>
<td>2.79</td>
</tr>
<tr>
<td>1,5</td>
<td>1410</td>
<td>3.72 ± .16</td>
<td>3.92 ± .17</td>
<td>3.88</td>
</tr>
<tr>
<td>1,5,3</td>
<td>1185</td>
<td>6.73 ± .22</td>
<td>7.04 ± .23</td>
<td>6.95</td>
</tr>
<tr>
<td>1,5,3,2</td>
<td>1108</td>
<td>7.71 ± .24</td>
<td>8.05 ± .25</td>
<td>7.70</td>
</tr>
<tr>
<td>1,5,3,2,4</td>
<td>906</td>
<td>10.53 ± .32</td>
<td>10.93 ± .35</td>
<td>10.31</td>
</tr>
<tr>
<td>1,5,3,2,4,6</td>
<td>---</td>
<td>15.43 ± .77</td>
<td>15.83 ± .80</td>
<td>16.4</td>
</tr>
</tbody>
</table>

$\text{/a}$ Out of programmed sequence (1,5,3,6) instead of (1,5,3,2,4,6)

$\text{/b}$ Inferred from other measurements
Apparently, the flux distribution changes which occur with control rod insertion are calculated properly for insertion of groups 1, 3, and 5 since these worth values compare very favorably with experiment. On insertion of groups 2 and 4, however, in which calculated group worths are low, the flux distribution changes may not be treated properly. Too great a shift in fission density to the outer region of the core would give the worth values shown in the table. The group 6 worths, for which calculated values are higher, illustrate the same effect.

Figure 3.22 also illustrates the same effect. It is a plot of calculated values of $k_{\text{eff}}$ (two-dimensional) for various critical cores with different numbers of fully inserted control rods as a function of critical boron concentration. Because of the underestimate of boron worth in the calculations, a trend of increasing values of $k_{\text{eff}}$ with increasing boron concentration is expected. Such a trend is definitely observed in the last four points, which correspond to a calculation for the unrodded core and successive insertion of control rod groups 1, 5, and 3. Addition of rod groups 2 and 4 gives less reactivity reduction than might be expected, and addition of group 6 to groups 1, 3, and 5 gives a greater reactivity reduction. Since an appreciable amount of resonance absorption occurs in Ag-In-Cd control rods, the discrepancy may be the result of an inaccurate distribution of resonance and thermal absorptions. For example, if the ratio of resonance absorption to thermal absorption were increased and rod worth maintained constant, agreement might be improved.

**Excess Reactivity and Temperature Defect**

The calculated value of $k_{\text{eff}}$ at operating temperature is 1.113. This corresponds to an excess reactivity of 10.2%. The excess reactivity inferred from the integration of differential control rod worth values is 11.4 ± 0.3, which indicates a difference slightly greater than 1%. Although this increase is expected to
CALCULATED VALUES OF $k_{eff}$ FOR SEVERAL CRITICAL CONTROL ROD CONFIGURATIONS

Fig. 3.22
increase core life by roughly 1500 hours, the net result is an increase in the expected core life at 392 Mw thermal from 8500 hours to the original design objective of 10,000 hours.

Since one of the uncertainties in the calculation of reactivity is the temperature defect (cold-to-hot reactivity swing), it is also of interest to make a comparison between experiment and calculation for this parameter. An estimate of the experimental value can be obtained by (a) evaluating the excess reactivity at a temperature, and (b) integration of the temperature coefficient as a function of temperature. Table 3.11 compares these values with calculation.

Table 3.11

<table>
<thead>
<tr>
<th>Boron Concentration (ppm)</th>
<th>Experiment $%\Delta \rho$</th>
<th>Calculation $%\Delta \rho$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1700</td>
<td>1.6 ± 0.2</td>
<td>2.0</td>
</tr>
<tr>
<td>1100</td>
<td>3.7 ± 0.4</td>
<td>4.4</td>
</tr>
</tbody>
</table>

Although the number of interest is that for an unborated core, the cold shutdown requirement precluded any measurement in the core at ambient temperature at concentrations less than 1100 ppm. At both boron concentrations in Table 3.11, the calculated reactivity drop is greater than that inferred experimentally. The implication is that at least some of the difference between calculation and experiment for the excess reactivity at the operating temperature of the core results from improper treatment of temperature effects on reactivity. Since experimentally determined pressure and temperature coefficients in the unborated core at operating temperature compare well with calculated values, and since the main component of these coefficients at operating temperature is due to moderator density changes, the discrepancy may be due to spectral effects.
3.3 Zero Power Reactivity Coefficients

Experiments were performed in order to evaluate the effects of moderator temperature and pressure on core reactivity with various control rod configurations, boron concentrations, and temperature at essentially zero power (less than 1% power). In addition to these measurements, the effects of main coolant flow rate on core reactivity were checked in order to verify the fact that the physical positions of the various core components did not shift when main coolant flow rate changes. Calculations based on the analytical techniques presented in YAEC-136 were carried out prior to the experiments to serve as guides for the experimental program. Table 3.12 shows a comparison of the calculated and experimental data for several conditions.

The low power kinetic parameters were determined by period measurements. Two general types of measurements were used; dynamic and static period measurements. During the dynamic measurements, the temperature or pressure was continually varied monotonically. During this variation, period measurements at a given rod position were repeated. Thus, the variation in core reactivity - as inferred from the period to reactivity conversion - was obtained as a function of temperature or pressure. Hence, the coefficient was readily evaluated. Similarly, static measurements were made by doing a period measurement changing the pressure or temperature a discrete step and repeating the period measurement at the same rod position. Static measurements were found to be more reliable for pressure coefficient, while static and dynamic measurements were of comparable reliability in the determination of the temperature coefficient. The following sections present in detail the reactivity coefficient measurements made at essentially zero power.

3.3.1 Moderator Coefficient

Three discrete "sets" of coefficients were determined for the Yankee reactor; (1) ambient to operating temperature at high boron concentrations - 1050 to 1550 ppm, (2) at operation temperature over the boron concentration range of 1700 to 0 ppm, and (3) at zero boron over the temperature range from 470 to 550°F.
These results are shown graphically in Figures 3.23, 3.24, and 3.25. It is interesting to note that the data obtained at high boron concentrations had much less scatter than at approximately zero boron. Moreover, scatter becomes worse as temperature is increased from operation temperature to 550°F. The difficulty in the "zero" boron measurements was caused by not having the pressurizer completely void of boron.

3.3.2 Pressure Coefficient

Measurements of the pressure coefficient were carried out at high boron concentration (1200 to 1400 ppm) over a pressure range of 400 to 2000 psig and at zero boron over a range of approximately ± 300 psig about 2000 psig. Figure 3.26 shows the results of these measurements. In connection with the measurements at high boron concentration several comments apply:

(1) Temperature effects cannot be ascertained within the accuracy of the data, i.e. the pressure coefficient at 514°F, 1000 psig appears to be the same as the pressure coefficient at 100°F, 1000 psig.

(2) Boron mixing problems account for the large uncertainties of ± 50% on the measurements. The boron concentration in the pressurizer must be very close to equilibrium with the main coolant when trying to measure a 0.05% reactivity change for a 500 psig pressure swing (less than a 10 ppm change completely masks the pressure effect) at a time when pressurizer equilibrium is being upset by the change in pressure.

3.3.3 Flow Coefficient

No measurable change in core reactivity was noted in either the hot or cold reactor at zero power when the flow rate was varied from shutdown cooling flow of 1000 gpm to 4 pump flow of ~10^5 gpm in any step sequence. A flow coefficient of reactivity at power was noted (see Section 5.2.3) and is associated with the increase in reactivity loss due to the axial variation in the moderator temperature in going from four to three pump operation at a given
MODERATOR TEMPERATURE COEFFICIENT \( \Delta \rho/\Delta T \cdot 10^{-4} \) (\( \text{f/F} \))

MAIN COOLANT TEMPERATURE (\( ^\circ \text{F} \))

INDICATED VALUES HAVE AN UNCERTAINTY OF \( \pm 15\% \)

- MEASURED
- ESTIMATED

\( C_B = 1050 \)
\( C_B = 1190 \)
\( C_B = 1300 \)
\( C_B = 1450 \)
\( C_B = 1540 \)

MODERATOR TEMPERATURE COEFFICIENT OVER THE TEMPERATURE RANGE 70-514\(^\circ\)F AT HIGH BORON CONCENTRATION

Fig. 3.23
Moderator coefficient with variable boron concentration at operating temperature

Fig. 3.25
power level. The flow coefficient can result in only negative reactivity insertion since the main coolant pumps cannot be started when the reactor is at power because of the primary loop temperature interlocks.

3.3.4 Discussion of Results

To calculate temperature coefficients, a graph of $k_{eff}$ versus moderator temperature was constructed. The slope of such a curve is the moderator coefficient. A combination of boron and banked control rods was provided to maintain criticality in the calculations. The agreement between calculation and experiment (Table 3.12) is quite good without boron. With boron in the main coolant system, the agreement is not as good. This discrepancy may be the result of an underestimate of the neutron absorption rate in boron relative to uranium. This is the same effect as had been observed in the comparison of predicted and measured boron worth. This table also shows the strong positive effect of adding boron or removing control rods, e.g., calculations indicate that fully inserted control rods add $-0.7 \times 10^{-4}/°F$ to the zero boron temperature coefficient while boron adds $+0.9 \times 10^{-7}/°F$ -ppm of natural boron.

Table 3.12

| COMPARISON OF CALCULATED AND EXPERIMENTAL KINETIC COEFFICIENT AT ZERO POWER |
|----------------------------------|------------------|------------------|
|                                 | Calculated       | Experimental     |
| Moderate Coefficient            |                  |                  |
| (in $10^{-4}/°F$)               |                  |                  |
| $100°F$, 1200 ppm               | -0.3             | -0.5 ± 0.09      |
| $250°F$, 1200 ppm               | -0.9             | -0.6 ± 0.09      |
| $514°F$, 1200 ppm               | -2.0             | -1.3 ± 0.2       |
| $514°F$, 600 ppm                | -2.6             | -2.2 ± 0.3       |
| $514°F$, 0 ppm                  | -3.2             | -3.1 ± 0.2       |
| Pressure Coefficient            |                  |                  |
| (in $10^{-6}/\rho$/psig)        |                  |                  |
| $514°F$, 1200 ppm               | +1.3             | +1.0 ± 0.5       |
| $514°F$, 0 ppm                  | +2.5             | +2.7 ± 0.7       |

*Calculations indicate the temperature coefficient with banked and programmed rods to be very nearly equal. Experimentally, it was impossible to differentiate one from the other within the errors in the data.
The calculated pressure coefficient was obtained from the moderator coefficient by separating water density effects from neutron spectrum and fuel temperature effects, thus obtaining a rate of change of $k_{\text{eff}}$ with moderator density. This, combined with the pressure coefficient of density, yields the pressure coefficient of reactivity. The calculated and experimental coefficients are in excellent agreement for all conditions.
4.0 CONTROL ROD MECHANISM TESTS

Following the completion of the core loading, a check of the control rod mechanism operation and scram delay times at ambient temperature was completed in order to assure a safe approach to initial criticality and the ensuing test program. Three types of tests were performed: functional operation of the mechanism, control rod drop time determination, and scram of instrumentation delay time.

Functional operation was checked by observing the current from operating coils and comparing them to the normal or expected behavior\(^7\) both in terms of sequence and type of behavior. Several problems were uncovered during these tests; malfunction of some load transfer coils, poor settings of a few of the cam angles and the fact that the pull-down coil should not be energized\(^*\) when the rod is driven down to or past zero. All of these troubles were alleviated before proceeding to the next step in the test program, i.e. rod drives were all functional.

The second type of test performed was a determination of control rod scram times. These tests involved the measurement of four separate parameters; signal initiation to rod breaker opening time, rod mechanism release or breakaway time, time from the beginning of motion at 90\(^\circ\) to entry into the dashpot at 6\(^\circ\), and finally the dashpot closure time.

In the ambient temperature condition\(^**\) all control rods were dropped twice with measurement of the parameters listed above. The average breakaway time was found to be 0.206 sec. with a variation of approximately ± 0.04 sec.; the average drop time was 1.1 sec. with a variation of ± 0.2 sec. (see Figure 4.1 for a distribution of these drops); and the average dashpot closure time was 0.098 sec. with a variation of ± 0.045 sec. All of these values are approximately the design values\(^7,8\) , within the errors of the experimental data.

\(^*\) The danger lies in jamming the drive in the down position as caused by latch finger - dashpot spring reaction.

\(^**\) Flow was 1000 gpm from shutdown cooling with pressure in the range of 150-250 psig.
CONTROL ROD DROP TIME AT 100°F

Fig. 4.1
At the operating temperature, a more comprehensive series of control rod scram time determinations was performed in that one rod was dropped 30 times; a second rod, 15 times; four rods, 5 times; and the remainder, twice. If one uses the scram time determinations on the rod dropped 30 times in carrying out a statistical evaluation* of the rod drops, the following results are obtained:

a. Scram time** for the rod used as a reference was 1.758 ± .026 (1 σ) seconds.

b. 15 rods dropped in less than 3 σ limits.

c. One rod had a scram time above the 3 σ limits.

d. The drop time decreases as a rod is raised, dropped, raised and dropped again. This decrease is 2.5% from the first 5 drops to the 15th. It is believed that this decrease is due to the increase in temperature of the water within the mechanism housings. Repeated rod withdrawal and scram action causes high temperature water to be circulated from the vessel upper plenum area into the rod housing area. The effect of the high temperature, low density water then tends to reduce impeding forces.

e. All rods scrammed in less than 2 seconds.

It is obvious from the above results that the statistical distribution of scram time does not exist; rather that several statistical families exist. Figure 4.2 serves to point out the cause of this effect, i.e. rods toward the outside of the core take longer to drop than those in the center. This is caused by the fact that with four pumps running, as they were in these tests, the cross flow of water as it exits from the core and passes to the outlet nozzles imparts a horizontal force component to the control rods which increases drag and hence

* It is granted that the statistical evaluation of 23 rods from a sample whose performance is known for only 30 operations is not good from a "randomness" standpoint; however, these results are given since they do allow a logical presentation of data.

** Defined as breaker open to rod bottomed
CONTROL ROD DROP TIME AT 514°F FOR FULL FLOW AND NO FLOW

Fig. 4.2
drop time. The effect is greatest at the outlet nozzles; that is revealed by the inner group 6 rods which are directly in "front" of the nozzles. These rods, which on the average drop fastest at zero flow, have one of the longest drop times with all pumps running. Further evidence of this force component, substantiated by calculation, is given by an analysis of the relative drop times for zero and full flow for the other rods in the core (see Figure 4.2).

The third type of performance test that was carried out was a determination of the initiating scram circuit firing times. These are given in table 4.1 and are self-explanatory.
### Table 4.1

**CIRCUIT FIRING TIMES**

(Time From Input Signal Received To Rod Breaker Open)

<table>
<thead>
<tr>
<th>Input Signal</th>
<th>Milliseconds Before Rod Breaker Open</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Turbine Board Push Button</strong></td>
<td>26</td>
</tr>
<tr>
<td><strong>S-29 Push Button</strong></td>
<td>23</td>
</tr>
<tr>
<td><strong>S-28 Push Button</strong></td>
<td>22</td>
</tr>
<tr>
<td><strong>High Flux Channel</strong></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>290</td>
</tr>
<tr>
<td>7</td>
<td>281</td>
</tr>
<tr>
<td>8</td>
<td>252</td>
</tr>
<tr>
<td><strong>High Startup Rate Channel</strong></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>217</td>
</tr>
<tr>
<td>2</td>
<td>196</td>
</tr>
<tr>
<td>3</td>
<td>233</td>
</tr>
<tr>
<td>4</td>
<td>223</td>
</tr>
<tr>
<td><strong>Loss of Pressure, 1900 psig</strong></td>
<td>797</td>
</tr>
<tr>
<td></td>
<td>Pressurizer</td>
</tr>
<tr>
<td></td>
<td>609</td>
</tr>
<tr>
<td><strong>1800 psig</strong></td>
<td>858</td>
</tr>
<tr>
<td></td>
<td>Main Coolant</td>
</tr>
<tr>
<td><strong>2000 psig</strong></td>
<td>599</td>
</tr>
<tr>
<td><strong>Loss of Flow - Loop 1</strong></td>
<td>865</td>
</tr>
<tr>
<td></td>
<td>868</td>
</tr>
<tr>
<td></td>
<td>885</td>
</tr>
<tr>
<td></td>
<td>737</td>
</tr>
<tr>
<td><strong>Generator Differential Relay</strong></td>
<td>92</td>
</tr>
<tr>
<td><strong>Unit Differential Relay</strong></td>
<td></td>
</tr>
<tr>
<td><strong>No. 1 SST Differential Relay</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Generator Loss of Field Relay</strong></td>
<td>125</td>
</tr>
<tr>
<td><strong>Generator Overcurrent Relay</strong></td>
<td>From 500 to 16000 depending on the magnitude of the fault.</td>
</tr>
<tr>
<td><strong>No. 1 SST Overcurrent Relay</strong></td>
<td></td>
</tr>
<tr>
<td><strong>No. 4 SST Overcurrent Relay</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Generator Ground Relay</strong></td>
<td>2 hours</td>
</tr>
</tbody>
</table>
5.0 POWER PHYSICS TESTS

5.1 Reactor Power Level Determination

Reactor power level is determined by relating generator gross electric output to the reactor output. Secondary plant calorimetrics (energy balances on the steam generators) were obtained at selected stable gross generator outputs and these calorimetric data were analyzed to obtain the total BTU/HR removal rate from the steam generators. After taking into account primary plant thermal losses and main coolant pump thermal inputs, a plot of generator gross Mwe versus reactor Mwt was established, (see Figure 5.1).

For subsequent test work, the stable reactor power level was determined by measuring generator gross electric output and converting to reactor thermal power using the plot described above. Generator gross electric output was determined from a control board mounted Mwe meter having an accuracy of ± 1.0% or from measurements of the disk speed of the generator watt-hour meter from which generator output can be determined with an accuracy of ± 0.3%. Using this method to determine reactor power level requires that stable load conditions exist for a period long enough to assure that generator Mwe and reactor Mwt are in equilibrium. In order to assure that equilibrium conditions existed, data were taken only after the main coolant temperatures had been stable for a period of several minutes.

5.2 Reactivity Measurements

5.2.1 Incremental Rod Worth

Two types of measurements were carried out in an attempt to define incremental control rod worth curves for use during power operation. One involved direct rod worth measurements during xenon decay following reactor shutdown. The other was carried out by coupling an experimentally determined change in main coolant temperature induced by control rod motion (ΔT/Δh) with an analytically determined temperature coefficient. The following paragraphs describe the results of these measurements for the first 400 equivalent full power hours.
CALORIMETRIC DETERMINATION OF REACTOR POWER LEVEL

Fig. 5.1
The direct measurements after shutdown were made using the rod motion - period technique described fully in section 3.2. Corrections were made for xenon decay whenever necessary. Figure 5.2 shows the results of the experiments for group 5. For this rod group, there is surprisingly little deviation of the rod worth data from the zero power data. Only at the peak incremental worth is the deviation of any significance and, even there, the scatter in the data is such that a "redefined" group 4 curve cannot be drawn. Similar results were obtained for group 6.

Data obtained at power using the second technique that was mentioned above gave incremental worth values for group 4 that were in qualitative agreement with the low power data. Most of the values obtained were within ± 10% of the low power worths.

5.2.2 Xenon and Samarium Worths

During the step-wise approach to full power operation at 392 Mw, measurements were made of the reactivity tied up in xenon. The experimental procedure involved the use of rod position as a function of time in conjunction with the low power rod worth data discussed above. From this information, the transient and equilibrium xenon behavior was obtained as a function of power output. Figure 5.3 shows the results of these evaluations of equilibrium xenon poisoning; in addition, data are presented for peak xenon following shutdown from various levels.

The calculated values $\bar{x}$ for equilibrium and peak xenon are in excellent agreement with the experimental data as Table 5.1 shows. Following shutdown from various power levels and subsequent complete decay of xenon, the net loss in reactivity due to samarium and burnup were determined.
INCREMENTAL ROD WORTH OF CONTROL ROD GROUP 4 AT 514°F
DATA OBTAINED DURING XENON DECAY

- LOW POWER, XENON DATA

INCREMENTAL ROD WORTH (\(\Delta p/\Delta n\) - 10^{-4} p/inch)

CRITICAL ROD POSITION (INCHES)

LOW POWER: ZERO XENON

Fig. 5.2
REACTIVITY IN XENON POISONING

Fig. 5.3
Table 5.1
XENON AND SAMARIUM POISONING AT 392 MWT

<table>
<thead>
<tr>
<th></th>
<th>Calculated $%\Delta P$</th>
<th>Experimental $%\Delta P$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Xenon, equilibrium</td>
<td>2.5</td>
<td>2.5 ± 0.2</td>
</tr>
<tr>
<td>Xenon, peak</td>
<td>3.0</td>
<td>3.0 ± 0.2</td>
</tr>
<tr>
<td>Samarium, equilibrium</td>
<td>0.87</td>
<td>-----</td>
</tr>
<tr>
<td>Samarium, peak</td>
<td>1.07</td>
<td>-----</td>
</tr>
</tbody>
</table>

Unfortunately, insufficient data were obtained to allow an evaluation of the samarium reactivity loss at full power. Rather, dependence is placed on the calculation for a determination of the effect.

5.2.3 Power Coefficient Determination

The power coefficient was determined in two ways, both of which are dependent on the zero power data of Section 3.2. One involves the determination of the rod motion necessary to compensate for a given change in gross electrical output while maintaining a constant main coolant average temperature; the other entails the determination of the change in main coolant average temperature necessary to compensate for a given Mwe change at constant rod position. Thus, one is dependent on the rod worth data and the other on temperature coefficient information.

Ramp load increases and decreases as well as step load decreases were used in the measurement of the coefficient. For proper initial conditions all methods are of comparable value in determining the coefficient. For ramp increases the best measurements are made from a xenon-free condition; for ramp or step decrease, the proper initial condition is with equilibrium xenon. These are preferable initial conditions since they reduce the xenon corrections necessary to arrive at a power coefficient. Although all methods are of comparable value, practical limitations dictate the use of the ramp changes in load.
In determining a power coefficient in this manner, the changes in reactivity associated with the non-uniform radial and axial moderator density effect are included with the change in reactivity induced by the fuel temperature change (Doppler effect) $\frac{1}{\Delta}$.

Indications that the non-uniform moderator temperature contributes to the coefficient are best exemplified by the reactivity loss in going from zero power to a given power level with two flow rates. A change from four pump flow to three pumps increases the reactivity loss by 15% as the $\Delta T$ across the core increases. This change is predicted analytically which tends to increase the confidence in the analytical prediction of the separate effects - moderator and fuel - that make up the measured coefficient.

The measurement of the power coefficient is handicapped by uncertainties in the parameters needed to evaluate it. These are

1. The uncertainties referred to in 5.1.1 are introduced in converting from Mwe to Mwt since the reactivity loss due to a change in thermal or reactor output is desired.

2. Changes in rod positions as they affect the statistical weight function are expected to cause the major change in the power coefficient throughout life (the factor increases from approximately 1.3 to 1.9 throughout life) $\frac{1}{\Delta}$. However, for the range of rod positions encountered in the measurements referred to here, this is a small effect.

3. Statistical uncertainties in quantities used to convert rod position (rod worth) and temperature (temperature coefficient) changes to reactivity. Both are based on low power data. The low power incremental worths are, for the most part, accurate at low power with no poison ($\pm 6\%$ or better). However, as noted in 5.2.1, the data obtained at power deviate from the low power data by $\pm 10\%$. The low power temperature coefficient has a $\pm 7\%$ uncertainty associated with it.

4. Uncertainties in any xenon corrections that are applied.
Because of the large number of systematic errors not amenable to statistical analysis, an error analysis, per se, was not performed. Instead, a best value based on conditions where the xenon correction is negligible is quoted along with the observed spread in data.

Based on these assumptions, the power coefficient at full main coolant flow is \((-0.33 \pm 0.10) \times 10^{-4}/\text{Mwt}\) over the range of 80 Mwt to 392 Mwt. For lower power levels there is a trend toward a larger value \((\sim -0.5 \times 10^{-4}/\text{Mwt})\); however, greater uncertainties are present in Mwt determination.

The measured value is in reasonable agreement with the calculated power coefficient of \(-0.41 \times 10^{-4}/\text{Mwt}\). This is a surprisingly good comparison when one considers the problems associated with the calculation and measurement of power coefficients.

If the differences observed are due to calculational rather than experimental techniques, one would suspect the thermal model more than the nuclear model used in the calculations. The fuel surface temperature varies directly with the temperature drop in the gas gap between pellet and clad; the power coefficient is directly proportional to the rate of change of this temperature with power. Thus, uncertainties in calculation of the temperature drop as caused by unknown gas composition, gap size and contact resistance are reflected directly in the calculated power coefficient.

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* In order to determine whether the quality of the data could be improved by a detailed xenon correction, a power history (100 hrs) with 14 changes in power level was examined. Xenon corrections were made and the power coefficient evaluated. No improvement in the data was found.
6.0 TRANSIENT TESTS

The transient tests performed on the Yankee reactor showed that the plant behaved very well in both the controlled and uncontrolled mode of operation. The plant transient response to load changes with rods on automatic control was very stable. The primary system could take a rapid drop in load on the order of 60 MWe without excessive transient variations, although the secondary plant was a more limiting factor. Trends that indicated differences between actual and predicted performance were noted. These were primarily caused by the simplified steam generator model employed for the Yankee simulator studies.

6.1 Loss of Load Tests

The objectives of these tests were to determine the capability of the plant to handle a loss of load on manual control, without reactor scram, and to provide data to be used in the determination of the power coefficient. Test results showed that for the most pessimistic combination of experimental values of power and temperature coefficients which existed at the time of the tests the plant should be capable of a load drop from 90 MWe to 3 MWe without operation of the first set of secondary side safety valves. This was considered to be a limit for these tests because of the undesirability of operating such valves unless absolutely necessary. Actually no tests were performed for full load drops at higher than 60 MWe.

For the 60 MWe test the reactor did not scram and no limits were reached. Furthermore, the plant parameters recorded changed as expected and the pressurizer spray functioned as required.

6.1.1 Test Procedure and Conditions

Prior to the load drop, equilibrium plant and core conditions were established at the desired power level and the reactor control was put into the manual mode. The load was then
removed by manually tripping the two 115 Kv oil circuit breakers. The responses of the reactor and plant to loss of load transients were determined by recording the following operational parameters at frequent intervals.

a. Cold-leg and hot leg temperatures for each main loop and \( T_{avg}\).
b. Pressurizer pressure
c. Pressurizer level
d. Gross electrical output of the turbine generator.
e. Secondary steam pressure.

During the stepwise approach to full power several loss of load tests were conducted. The first two tests were run for a 30 to 3 MWe load drop. In the first run, feed and bleed were left in operation. This made the evaluation of the pressurizer volume surge caused by loop temperature and pressure changes difficult. Also steam dump was actuated, with the result that the primary system was cooled down and an elevated primary loop temperature condition did not exist at the end of the test. The second test was run without feed and bleed and steam dump. On this test the turbine was tripped automatically because of high level in the turbine moisture separators. This turbine trip caused the reactor to scram so that again the desired equilibrium condition was not obtained. A modification to the turbine moisture separators was made which consisted of installing a larger vent line between the moisture separators and the heater drain tank.

Analysis of data from these two tests showed that operation was acceptable for a higher power load drop so that, after the modification, the test was run for a load drop from 60 to 3 MWe.

When the 60 MWe test was performed it was observed that the indication of the water level in the moisture separators rose, went off scale and was thus close to the turbine trip level. This meant that step load changes greater than 60% power might produce a turbine trip and a reactor scram; thus, loss of loads studies above this power level were not performed.
6.1.2 Results of 60 MWe Load Drop

With a loss of electrical load, there is a sharp decrease in steam flow from the steam generators to the turbine and, consequently, a decrease in heat removal from the steam generators with an accompanying steam pressure and temperature increase. Since the reactor power lagged the power demanded, the energy contained in the primary loop increased which increased both the hot and cold leg temperatures as shown in Fig. 6.1. As the steam temperature rose, the heat removing capacity of the steam generator fell and the cold-leg approached the hot-leg temperature. Both temperatures, and therefore the average temperature of the coolant, rose causing a decrease in power output of the reactor. This decrease was caused by the moderator temperature coefficient of reactivity. This caused the rate of temperature increase to drop, until equilibrium was reached between heat addition by the reactor and heat removal through the steam generator and primary coolant piping. Equilibrium in temperature was reached at a value higher than initial because of the power coefficient of reactivity. The maximum cold-leg temperature of 534°F and the maximum hot-leg temperature of 534.5°F was reached approximately 5.5 minutes after loss of load. The hot-leg temperature was well below the maximum allowable value. The maximum steam generator steam temperature of 529.5°F which corresponds to a steam pressure of 865 psig was reached approximately 6.25 minutes after loss of load. Because of this increase in primary system temperature the water density was decreased which caused a volume surge into the pressurizer. As measured by the wide range level detector, Fig. 6.2, this volume surge peaked at about 82 ft³ in about 7 minutes and then decreased to an equilibrium value of 76 ft³. This time shift between the peak in loop temperatures and the peak in volume surge is an indication of the amount of mixing delay between various water regions within the reactor vessel. A similar effect was observed in changing the boron concentration in the primary loop although for that case the pressurizer provided an additional mixing lag.
HOT-LEG TEMPERATURE, COLD-LEG TEMPERATURE, AND APPROXIMATE SECONDARY STEAM TEMPERATURE AS A FUNCTION OF TIME FOR LOSS OF LOAD FROM 60 MW(E) (250 MWe)

Fig. 6.1
PRESSURIZER PRESSURE AND SURGE VOLUME AS A FUNCTION OF TIME FOR LOSS OF LOAD FROM 60 MW(E)

Fig. 6.2
The pressure transient as measured by the wide range pressure detector is also given in Fig. 6.2. The rate of volume surge was such that spray was actuated at the set point of 2300 psig twice and became effective at a slightly higher pressure causing the pressure to drop to the spray off set point of 2250 psig with a slight amount of undershoot in pressure. The continued decrease in pressure after the volume surge had reached equilibrium was caused by steam condensation in the pressurizer as a result of pressurizer heat losses.

The analytical model for the pressurizer assumed that during the compression cycle the steam phase followed an isentropic process with constant mass until spray was actuated. As shown in these data this assumption was valid only for the initial portion of the transient. Above 2150 psig the actual transient deviated from this process which again is an indication of steam condensation caused by heat losses. On the second pressure increase the deviation from the isentropic process is even greater. The plot of the pressure vs. surge volume given in Fig. 6.3 further substantiates this conclusion. In this figure a comparison is made between saturation and isentropic compression for no condensation but with a new steam mass estimated for each condition where the pressure was at a minimum by assuming saturation conditions at that time. This plot shows that the compression cycle is less than a saturation process for constant mass which could be the case if condensation were present. Without further data no judgment of the process other than that of initially isentropic can be made. However, as demonstrated in the following discussion, if an isentropic process with condensation were assumed, the amount of steam condensed is within experimental error as concerns the measurement of volume surge.
PRESSURIZER PRESSURE AS A FUNCTION OF PRESSURIZER STEAM VOLUME FOR LOSS OF LOAD FROM 60 MW(E)

Figure 6.3
The change in mass of water in the primary loop from initial conditions to the temperature and pressure conditions at 7 minutes is approximately 3,060 lbs. At hot leg conditions this corresponds to a volume change of 64 ft$^3$ as compared to an indicated change of 82 ft$^3$. However, if an isentropic compression curve is assumed, calculations based upon the indicated pressure and volume transients show that 365 lbs. of steam is condensed in 7 minutes. This corresponds to an additional water volume of approximately 10 ft$^3$ which yields a total surge of 74 ft$^3$ as compared to 82 ft$^3$ indicated. The error of 12 ft$^3$ is within the instrumentation errors for temperature and level changes.
6.2 Loss of Flow and Emergency Cooling by Natural Circulation Tests

The principal objective of these tests was to demonstrate the ability of the main coolant system to provide adequate heat transfer from the reactor for a sequential loss of main coolant flow. The sequential loss was that which would be caused by a loss of the two outside power lines followed approximately one minute later by removal of the generator field excitation which causes a loss of the two remaining pumps.

6.2.1 Conditions of Test

The tests were carried out at two levels of operation, namely 60 MWe and 120 MWe. Before each test, the reactor was in operation at equilibrium power for a sufficient length of time to build up decay heat. These times were 119 hours and 136 hours for the 60 MWe and 120 MWe tests, respectively.

Actually, the 60 MWe test was initiated from an equilibrium condition with three loop operation so that the pump loss sequence was 3-2-0 instead of 4-2-0 as in the case of the 120 MWe test.

The reactor was scrammed and the turbine tripped automatically when the flow decayed to 80% of full flow.

The 120 MWe test ran about three hours. During this time, several special operating conditions were established to provide additional transient data. In general, these affected the operation of the pressurizer and the steam generator level controllers. The feed and bleed system was put into manual control and the pressurizer level was allowed to decrease to 75" before being manually charged to 100" and again secured. Consequently, the pressurizer level variations give a good measure of changes in loop average temperature. In a similar manner the steam generator feed flow was shut off for a part of the transient to obtain the change in level caused by evaporation of the secondary water. These level variations are shown in detail in Fig. 6.4. The rate of change of level was then used to determine the heat input to the secondary plant. This heat flow added to the loop heat losses, was needed to determine the decay heat generation.
Boiler No. 3

- Feed water to boilers
- Feed to No. 3 boiler
- Feed to No. 2 boiler
- Feed to all boilers
- Feed stopped to all boilers

Boiler Level as a Function of Time for Loss of Flow from 120 MW(e)

Fig. 6.4
The results from these tests showed that temperature had a tendency to stabilize with time. As no excessive temperatures were recorded, this tendency indicated that the ability of flow by natural circulation to remove decay heat from the core was safe and adequate.

6.2.2 Discussion of Results

The predicted results for this test were that the core outlet temperature should rise quickly to a peak, drop to a minimum and finally stabilize. For the analysis of the actual core outlet temperature, reference is made to Figs. 6.5 and 6.6 which show averaged readings from the in-core thermocouples as well as a measure of the spread in the readings (see Section 2.0 for description of the thermocouples and resistance thermometer indication). The following paragraphs describe the chronological sequence of events for the actual test.

In the initial portion of the tests, an immediate slowdown of flow* caused the coolant to absorb more heat, raising the core outlet temperature to a peak. From Fig. 6.5 it is estimated that this peak is in the range of 530°F to 540°F. Since a maximum core outlet temperature equal to or above saturation was considered to be the danger point, the initial core outlet temperature peak is well within safe limits. To further substantiate this statement, reference is made to Fig. 6.7 which shows the hot-leg and cold-leg temperatures measured by the loop thermometers vs. time. An initial hot-leg temperature peak of 534°F is shown when the readings were corrected for a time lag of 2.5 secs. With a bypass flow through the core of approximately 10%, the average core outlet temperature could have reached no higher than 540°F. Thus, the resistance thermometer data checks the thermocouple data quite well. Since at this time the inlet core temperature was still 500°F the

* The flow coastdown in loops 1 and 4, as taken from the movie film of the control board during the transient, are shown in Fig. 6.7-a.
AVERAGE CORE OUTLET TEMPERATURE, AS MEASURED BY THERMOCOUPLES AS A FUNCTION OF TIME, FROM 0 TO 350 SECONDS FOR LOSS OF FLOW FROM 120 MW(e)

Fig. 6.5
Core outlet temperature, as measured by thermocouples, as a function of time from 30 to 110 minutes, for loss of flow from 120 MW(e)

Fig. 6.6
NOTE: $t = 0$ IS ARBITRARY REFERENCE

(A) MAIN COOLANT FLOW

(B) HOT-LEG AND COLD-LEG TEMPERATURE

PRIMARY PLANT PARAMETERS AS A FUNCTION OF TIME FOR LOSS OF FLOW FROM 120 MW (e)

Figure 6.7
average enthalpy rise across the core was 48 Btu/lb. With a design hot channel factor of 3.35, which at the time of the test was known to be high, the maximum enthalpy out of the core was 648 Btu/lb. This corresponds to a saturation pressure of 1800 psia or 1785 psig which is about 100 lbs below the minimum pressure reached during the transient.

As the core stored heat decreased while flow remained above 50% of full flow, the core outlet temperature fell to a minimum of 510°F for the 120 MWe test. This temperature remained essentially constant until the second two pumps were cut out. This minimum was reached in about 40 seconds and remained for about 30 seconds.

With the loss of the second two pumps and as the total flow decreased to a minimum, thermal energy was again added to the coolant causing the core outlet temperature to rise to another peak, 190 seconds from test start. This second temperature peak was quite close to the first, registering 537°F.

In conjunction with this second increase, natural circulation was established. An examination of the loop temperatures best shows the sequences following the initiation of natural circulation. As is shown in Fig. 6.8, the outlet temperature of the steam generator decreased because the water, previously stagnant in the steam generator tubes, has reached the cold-leg pipe. As this cold water flows through the core, the outlet temperature of the core is decreased. At some time later the inlet temperature to the steam generators was also decreased. In interpreting the temperature curves in Fig. 6.8, one must remember that for a value of 2% flow the total loop flow delay becomes on the order of 10 minutes.

Once natural circulation had been established the loop temperatures began to rise because the decay heat generation rate was greater than the heat removal rate. This was caused by the
PRIMARY PLANT PARAMETERS AS A FUNCTION OF TIME FOR LOSS OF FLOW FROM 120 MW(e)

Fig. 6.8
special operating procedures followed on the steam generator level control. Under normal conditions steam dump would have been employed as required to maintain the desired equilibrium temperatures.

In the Final Hazards Summary Report the reactor cold leg temperature was predicted to drop immediately, followed by a rise to approximately 514°F. The initial drop, caused by the continued full power withdrawal of steam from the boilers during initial flow decay, did take place reaching a minimum of 500°F. With the turbine trip-out, the steam flow from the boilers decreased and caused the cold-leg temperature to increase. According to the prediction in the above reference, the next maximum should be slightly greater than 514°F, however, a high of only 508°F was recorded. As decay heat was decreasing, the cold leg temperature then decreased to a stabilized value. Feedwater then entered the boiler sending the temperature quickly down. The minimum temperature expected was that corresponding to the saturated steam pressure in the boiler. For the 120 MWe test, this temperature was 469°F which compares well to the minimum of 467°F recorded. For the 60 MWe test this temperature was 486°F, somewhat above the 481°F recorded. With the start of natural circulation, coolant heated by decay heat reached the cold leg recorder resulting in a cold-leg temperature rise. As previously mentioned, a difference between decay heat and heat removal caused a gradual increase in cold-leg temperature throughout the region of established natural circulation. This increase subsides as decay heat diminishes.

6.2.3 Determination of Flow

In any discussion of natural circulation, it becomes important to determine the magnitude of the flow rate that is established. Several methods are available to do this. One involved the use of transit times from the cold-leg temperature indicator to the hot-leg indicator while the other is dependent on the core heat source and ΔT across the core.
To apply the first method, attention is directed to Fig. 6.8 which gives the temperature responses for the 120 MWe test. The transient 115 to 120 min. from test start, was caused by cold water injection to the boilers. From the transient, delay time for a cold-leg change to appear in the hot-leg was estimated. From the volumes of the various parts of the primary system, a volume flow was determined. Considering this as constant throughout the natural circulation region and taking specific volumes determined from average temperature of the coolant, one estimate of flow with respect to time can be made (see Fig. 6.10).

From the delay time, $\Delta T$ across the core was determined. The decay heat curve $\frac{Q}{C_p} = Q_D$, is shown in Fig. 6.9. With the relation $W = \frac{Q_D}{C_p \Delta T}$, another estimate of flow was obtained. As Fig. 6.10 shows, the flow rates determined by the two methods are in good agreement. However, at this time there is not sufficient data to put a qualitative estimate of error on either the decay heat calculation or the flow calculations.

6.3 Dropped Control Rod Test

One of the areas of uncertainty during the design of the Yankee Plant were the conditions that would prevail following a dropping of a control rod. A major question was the relationship between the worth of a dropped rod and the resultant distortion in flux distribution. The worth of the rod would determine whether or not the existing process instrumentation would cause a scram and thus protect the core from a return to power with a distorted distribution sufficient to cause core damage.

During the design phase the decision was made to wait until the testing program and determine by experiment the worth of a dropped rod and the resultant power distribution. Based upon the results of this test a contact meter set at 80% of full load on the power range instrumentation was installed so that a sudden decrease in power to 80% on any channel would automatically reduce turbine generator load to 80%. This circuit provides adequate protection against a dropped rod accident.
DECAY HEAT AS A FUNCTION OF TIME FOR LOSS OF FLOW FROM 120 MW(e)

Fig. 6.9
CALCULATED HEAT AND FLOW BEHAVIORS AS A FUNCTION OF TIME FOR LOSS OF FLOW FROM 120 MW(e)

Fig. 6.10
6.3.1 Conditions of Test

In order to determine how serious the redistribution of power would be, a controlled "rod drop" test was simulated at part load. With control rod groups 1, 2, 3 and 5 at zero, group 6 at 90° and control with group 4 at approximately 40°, a rod in group 6 (worth ~ .25% $\Delta k/k$) was inserted into the core with its negative reactivity being balanced by group 4 withdrawal. Enthalpy rise distributions in the core were obtained from the in-core thermocouples. By first inserting a group 6 rod in the fully instrumented quadrant and then repeating the test for a diametrically opposed control rod, a full core enthalpy distribution was constructed.

This enthalpy map formed the basis for an estimate of the increase in maximum-to-average enthalpy rise. By referring to the distribution before the "drop", an increase in hot channel factors of 30% was noted. Based on this increase in hot channel factors, this transient test was done in order to evaluate whether adequate protection was available or whether additional circuitry should be installed.

Procedurally, the following steps were carried out in order to assure that a stringent test of the protection circuitry would be accomplished:

a. $T_{avg}$ was driven to the lower end of the control band so that when the synthesized control rod drop occurred, rod motion would commence without delay in an attempt to restore $T_{avg}$--decreasing the probability of reaching the low pressure scram setting. Fig. 6.11 shows this initial condition, $T_{avg} = 512.2$.

*This is not known to be a maximum increase. As the worth of the dropped rod increases, the redistribution becomes worse; however, the probability of low pressure scram increases.


**Fig. 6.11**

- 
  - PRIMARY SYSTEM RESPONSE TO DROPPED ROD
  - \( \Delta V \) - FT\(^3\)
  - PRESSURE psig
  - MWE LOAD
  - % CORE POWER
  - \( T_H \)
  - \( T_{AV} \) CONTROL
  - \( T_C \)
  - TIME IN MINUTES

- BACK UP HEATERS ON
- WR
- NR
- EXTRAPOLATED

\[ CH \ 5 \ 100\% \times 5.05 \times 10^{-5} \ AMPS \]
b. Pressure was set at the high end of the control band, 2010 psig. With the load at 60 MWe, and prior to the setting of the above conditions, group 2 was withdrawn to give an equivalent of 0.25% in reactivity.

The test was initiated by manually deactivating the stationary gripper coil holding group 2 with the plant in normal auto control on group 4, resulting in a rapid insertion of negative reactivity.

6.3.2 Results of Test

As shown in Fig. 6.11 core power level dropped immediately as did pressure and temperatures. However, the automatic rod motion induced by the $T_{\text{avg}}$ control signal going out of the dead band plus the pressurizer heaters coming on to restore pressure acted to counterbalance the reactivity and temperature effect. The load, temperature and pressure were returned to nominal by the auto control system without alarm or scram actuation. Thus, it was demonstrated that the dropping of rod worth 0.25% reactivity was not sufficiently protected against to obviate the maldistribution of power that would result.

As noted in Fig. 6.11 the core power level dropped sharply on the introduction of the 0.25% reactivity. The observed drop of 25% compares well to a predicted drop of 25% as obtained from a one group of delayed neutron model. This drop is to be expected for all dropped rod cases. Use was made of this sudden drop in the design of the protection circuit. In this figure the pressurizer level and pressure transients were taken from both the wide range WR and narrow range NR instrumentation to show the effect of hysteresis in the narrow range instruments.

*Group 4 was inserted for a "balance" of reactivity
6.4 Evaluation of Automatic Reactor Control

The automatic reactor control system is designed to maintain a constant loop average temperature as a function of load from 15 MWe to full power output. The inputs to the control system are the highest hot leg temperature and the highest cold leg temperature. These temperatures are obtained by auctioneering units which select the highest temperatures measured by the hot leg and cold leg resistance thermometers. The control system averages the two input temperatures and compares this average to the reference or set point temperature which is normally set for 514°F. If the difference between $T_{avg} - T_{ref}$ exceeds +3°F a bi-stable magnetic amplifier initiates rod motion "IN" which inserts the controlling rod group. This insertion is discontinued when $T_{avg} - T_{ref}$ is reduced to 2.5°F. If the difference is less than -3°F the rods out bi-stable is tripped and the controlling group is withdrawn until the difference exceeds -2.5°F.

A series of transient tests were performed to determine the response of the controlled plant to station load changes. The load changes were introduced by operating the turbine control valves to increase or decrease load at a rate of approximately 4 MWe per minute. This was done manually at the control board by the turbine operator. For these tests the reactor control was in automatic and all systems were in their normal mode of operation.

Of primary interest in this test was the transient behavior of the primary loop variables; in particular, loop temperatures and pressure, pressurizer level, and control rod position. From the results of these tests, the conclusion was drawn that all deviations from equilibrium are within reasonable limits, considering the sensitivity of the recording instruments. As indicated above, the $T_{avg}$ controller is adequate for the settings used in these tests, +3°F dead band and 1/2°F hysteresis. The controlled response was stable, that is, there were no rod reversals during the power loading tests.
6.4.1 Conditions for Test

The reactor had been operated in automatic control at lower power levels during loading and unloading and performed satisfactorily as determined by an examination of the data. For historical purposes the data presented here were obtained from the first loading to full power, 120 MWe, on January 17, 1961 at 3:25 P.M. Actually the loading on automatic was performed in three increments; 60 MWe to 93 MWe, 90 MWe to 105 MWe and 105 MWe to 120 MWe with several minutes of equilibrium operation between each loading.

In particular the loadings presented here were obtained from two power increases with the following initial conditions:

<table>
<thead>
<tr>
<th>Nominal Change</th>
<th>Initial Power</th>
<th>Initial Pressure</th>
<th>Initial $T_{AVG}$</th>
<th>Group 4 Rod Position</th>
</tr>
</thead>
<tbody>
<tr>
<td>MWe</td>
<td>MWe</td>
<td>psig</td>
<td>°F</td>
<td>Inches</td>
</tr>
<tr>
<td>60 to 90</td>
<td>61</td>
<td>2050</td>
<td>511.5</td>
<td>36&quot;</td>
</tr>
<tr>
<td>105 to 120</td>
<td>108</td>
<td>1975</td>
<td>512</td>
<td>53-2/8&quot;</td>
</tr>
</tbody>
</table>

At this time group 6 rods were at 90" and groups 5, 3, 2, and 1 were at 0". All systems such as feed and bleed and the pressure control were in normal operation during these tests.

6.4.2 Results of Test

The operational transients for the aforementioned load increases are presented in Figs. 6.12 and 6.13. In brief, these two figures show the following transient characteristics.

The average turbine generator loading rate was 3-3/4 MWe/min from 60 to 90 MWe and 3-1/4 MWe/min from 108 to 122 MWe which was on the order of the 4 MWe/min that was used as a design basis for the Yankee control system. The latter part of the load transient for the increase from 60 to 90 MWe was a 13 MWe change in 1-1/2 minutes effectively a step in generation from 78 to 91 MWe. This corresponds to a 10% step which was also a design basis for the Yankee control system. In both transients the desired turbine power was overshot by approximately 2.5 MWe.
PRIMARY LOOP PARAMETERS AS A FUNCTION OF TIME FOR A LOAD INCREASE FROM 60 TO 90 MW(e)

Fig. 6.12

PRIMARY LOOP PARAMETERS AS A FUNCTION OF TIME FOR A LOAD INCREASE FROM 105 TO 120 MW(e)

Fig. 6.13
The automatic rod withdrawal went smoothly, that is, there was no overshoot in rod position. The average reactivity rate per interval of motion was decreased from $6.0 \times 10^{-5} \Delta k/\text{sec}$ to $3.0 \times 10^{-5} \Delta k/\text{sec}$, with successive withdrawals in correspondence with the incremental worth curve given in Fig. 2.16.

The main coolant pressure transient was very well contained with no significant variation except for the rapid ramp increase in the latter portion of the 60 to 90 MWe transient. At that time, the pressure was decreased by 50 psi. Actually the plant system was not at equilibrium for the 105 to 120 MWe increase so that the initial behavior differs from the first set of data.

The pressurizer volume surges were also well contained with maximum deviations of $3.0 \text{ ft}^3$ positive and $2.5 \text{ ft}^3$ negative reached during the transients. Except for those periods of time during which the pressurizer heaters were on, or where feed and bleed was effective, calculations based upon coolant pressure vs. pressurizer surge volume show the pressure changes followed a saturation process.

All loop temperature readings showed little fluctuation during the power transient with rapid stabilization after the power increase. The $T_{\text{avg}}$ controller was very stable; $T_{\text{avg}}$ varied over a band of only slightly greater than two degrees during both tests.
ACKNOWLEDGMENT

The authors are indebted to a number of people from Westinghouse, Yankee, the Yankee Sponsor Companies, and Stone and Webster without whose help the experimental program could not have been carried out. Notable amongst these people are A. R. Collier of Westinghouse and W. J. Schmidt of Stone and Webster who provided significant technical contributions during the physics and control rod tests and W. Lyman of Westinghouse for a comparable contribution during the transient tests. The calculated values for the nuclear characteristics of the core shown in the comparisons were obtained by R. F. Janz, J. D. McGaugh, and C. G. Poncelet of WAPD. Finally, the able efforts of Mrs. Mary Jane Kelly and Joan Jesko in preparing both drafts and final version of the report is gratefully acknowledged.
APPENDIX A

The following tabulations serve to present in detail, the banked control rod data that was obtained during the experimental program. These data are shown in graphical form in Figures 3.2 and 3.3 (Table A-1) and Figures 3.8 and 3.9 (Table A-2).
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<th>Critical Position (in)</th>
<th>Boron Concentration (ppm)</th>
<th>Temperature (°F)</th>
<th>Incremental Worth ($\Delta \rho/\Delta h \times 10^{-4} \rho/\text{in}$)</th>
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APPENDIX B

PERIOD - TO - REACTIVITY CONVERSION RELATIONSHIP

The following formula has been used to calculate the reactivity ($\rho$) corresponding to a measured reactor period $T$. The delayed neutron parameters used in the experiments are also given:

$$\rho = \beta^{235} \frac{T}{\sum_{i=1}^{6} \left( 1 + \frac{(\beta \nu)^{238}}{(\beta \nu)^{235}} \frac{a_{i}^{238}}{a_{i}^{235}} \delta_{28} \frac{a_{i}^{235}}{1 + \lambda_{i} T} \right)}$$

where $T$ is reactor period in seconds,

- $\lambda_{i}$ is decay constant of $i^{th}$ delayed neutron group.
- $a_{i}$ is the yield of the $i^{th}$ delayed neutron group normalized such that $\Sigma a_{i} = 1$. Superscript represents the yield from either U-235 or U-238 fissions.
- $\beta\nu$ is the total delayed neutron yield per fission of either U-235 or U-238 as indicated by superscript.
- $\nu$ is the number of prompt neutrons per fission for either U-238 or U-235 as indicated by superscript.
  $$\nu^{25} = 2.47, \quad \nu^{28} = 2.62$$
- $\delta_{28}$ is the ratio of U-238 fissions to U-235 fissions in the lattice. It has a value of .071 in the core at ambient temperature and .088 at operating temperature.
- $I$ is a quantity expressing the ratio of the "importance" of delayed neutrons to prompt neutrons in the fission cycle. This importance is a function of the reactor period, but changes only by about 0.1 percent over the range of periods used (80 to 5000 sec) in the present measurements. The value of $I$ as a function of boron concentration was calculated for the core at both ambient and operating temperature.
The six delayed neutron group parameters used in the inhour equation are tabulated below:

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<tr>
<th>Group Number</th>
<th>Half-Life (sec.)</th>
<th>Thermal Neutron Fission of U-235 Relative Abundance $a_{25}$</th>
<th>Fast Neutron Fission of U-238 Relative Abundance $a_{28}$</th>
<th>Mean Energies of Delayed Neut. Spectra</th>
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<td>$54.51 \pm 0.94$</td>
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<td>$0.013 \pm 0.001$</td>
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<td>Calculated Fractional Delayed Neutron Yield per Fission, $\beta$</td>
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BIBLIOGRAPHY


