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ABSTRACT

The 330-MW(e) Fort St. Vrain reactor, built by General Atomic for the Public Service Company of Colorado, is the first commercial High-Temperature Gas-Cooled Reactor (HTGR) employing the multihole block-type fuel element design. The Fort St. Vrain program has progressed through construction, preoperational testing, fuel loading, initial criticality, and operational testing at power levels up to 2% rated power. To date, all tests necessary before the rise to full power have been completed, and the rise-to-power program is expected to be resumed again in late 1975.

Major plant systems, including the prestressed concrete reactor vessel and circulators, have demonstrated adequate performance. Extensive tests on the reactor core at zero power and up to 2% power have demonstrated the accuracy in the design predictions of such core characteristics as critical rod position, control system worths, neutron flux distributions, and temperature coefficients. Gaseous fission product release measurements to date have confirmed the extensive analytical estimates.

Unforeseen schedule delays have centered around the control rod drives, circulator service water systems and problems arising from the introduction of water into the primary system, licensing requirements for

electrical cable separation, and seismic restraint systems for steam lines. However, none of the problems has indicated any inherent difficulties or design limitations in the HTGR concept.

1. INTRODUCTION

The 330-MW(e) Fort St. Vrain Nuclear Generating Station is the first commercial-size HTGR to employ the multihole hexagonal fuel block design developed and marketed by General Atomic. The plant is owned and operated by the Public Service Company of Colorado. When placed in commercial operation, Fort St. Vrain will have the highest net thermal operating efficiency of any nuclear generating station -- 39.2%.

Characteristically for the HTGR, the entire primary coolant system, active core, steam generators, and helium circulators are contained within a prestressed concrete reactor vessel (PCRV) (Fig. 1). The single reheat steam cycle (Fig. 2) operates at 2400 psig/1000°F/1000°F and uses a standard 3600-rpm tandem-compound turbine generator.

As of September 1975, operational testing of the reactor core and associated plant systems had been successfully completed through 2% of rated power. Since Fort St. Vrain is a commercial prototype, the testing programs have been extremely thorough. All systems and components have been systematically tested and evaluated to assure proper operation. When deficiencies have been discovered, each has been carefully investigated and a solution implemented.

The testing program emphasized data obtained on normal plant instrumentation but also included extensive instrumentation used only during the initial testing program. Most of the testing program has been carried out or supervised by members of the commercial HTGR design staff, so that experience gained in testing and startup operations at Fort St. Vrain is readily available to the designers of the commercial larger HTGRs.

Problems discovered in the low-power testing have been resolved, and the rise-to-power program is expected to be resumed in late 1975.

2. PLANT HISTORY

The construction permit for Fort St. Vrain was issued in September 1968. Construction was essentially complete in August 1971, consistent with the original established schedule. Subsequent schedule slippages to initial fuel loading resulted from mechanical and electrical deficiencies discovered during the extensive preoperational checkout and testing phase. In spite of these delays, the Fort St. Vrain schedule progress is comparable to the industry when compared with other "commerical" reactor projects. Of 19 nuclear plants which received their construction permit in 1968, only nine, including Fort St. Vrain, started loading fuel in 1974 [1]. To date, six, including Fort St. Vrain, have not reached commercial operation [2].

In May 1971, as construction work neared completion, difficulties were encountered in complying with new Nuclear Regulatory Commission (NRC) requirements for seismic design, cable segregation criteria, and environmental impact reports. The seismic requirements increased plant congestion, impeded the plant testing program, and increased the length of delays experienced.

In mid-1972, during the Hot Functional Tests, the helium circulators indicated a shaft-rub while running. The pre-nuclear water Pelton wheels, which were installed only for testing of the primary coolant system, were removed and found to be damaged by cavitation and erosion. Concurrent with replacement of the Pelton wheels, a method of pressurizing the Pelton wheel cavity was developed to suppress the cavitation. By May 1973, the circulators were reinstalled and the Hot Functional Tests were run successfully in parallel with the installation and checkout of the cavitation suppression system.

Also in 1972, a water leak was discovered in the core support floor cooling system. This leak was internal to the core support floor, with no water entering the core cavity. A new method of sealing tubes (embedded in concrete) with epoxy was developed, tested, and implemented within 4 months. The epoxy was tested for radiation life and found to be satisfactory.

On December 21, 1973, the Atomic Energy Commission (now NRC) issued a full-power, full-term operating license, and fuel loading began on December 27, 1973. Initial criticality was achieved on January 31, 1974, and the zero-power physics test program was completed one month later, giving excellent confirmation of all design predictions. The subsequent rise-to-power program has been interrupted several times by problems with some of the plant systems and components. These are described below.

2.1. Moisture ingress

The test programs were interrupted twice by the detection of water in the primary system. In August 1974, tests were being completed on the fuel handling machine which required some helium circulation. During these tests, a plant protection system module in the control room was removed for inspection. The module deactivated the helium circulator shutdown seals, allowing water from the common bearing water system to enter the lower PCRV cavity through one of the circulators which had been shut down. Helium containing moisture was then circulated throughout the PCRV by the operating circulators.

As a result of ensuing surface rusting conditions on the core helium flow control valves, the control rod drive assemblies were removed from the core, completely cleaned of rust, and tested. While the control rod drives were being cleaned, the moisture was removed from the vessel. All control rod drives were reinstalled by November 1974, and the hot physics tests were performed.

Indication of a second moisture ingress became apparent in January 1975, as discrepancies in the cold critical rod position indicated some irregularity. As two circulators were put into operation, the changes in the critical rod position indicated that the core graphite was absorbing moisture. An orderly shutdown was promptly initiated. Subsequent investigation indicated that moisture had leaked into the vessel via the circulator bearing H_2O system while the circulators were shut down.

The removal of the moisture from the helium was accomplished utilizing the helium purification system. To speed the moisture removal from the thermal barrier, the entire core cavity pressure was reduced to 10 mm Hg (abs).

During this time period, the reserve shutdown material in one region was inadvertently released owing to malfunctioning of a pressure regulator during a periodic test of the system. After depressurization, the reserve shutdown material was easily removed by the standard vacuum cleaning equipment provided for this purpose.

Extensive tests on core components have clearly established that moisture ingresses have not adversely affected the fuel status or caused permanent damage to any other system components.

2.2. Circulators and Pelton wheels

During the conversion of two circulators from pre-nuclear to nuclear configuration, a helium circulator already in nuclear configuration developed an internal helium leak, which required disassembly and inspection. The leak was found to be a result of a ruptured shutdown seal bellows caused by overpressurization during testing. This rupture was conclusively determined by General Atomic and Public Service Company of Colorado not to be indicative of a generic problem.

The inspection of this circulator provided an opportunity to verify the adequacy of the cavitation suppression system. The Pelton wheel was removed and was found to be free of damage from cavitation or erosion. However, dye penetrant checks revealed the existence of cracks in the Pelton wheel buckets and in the coupling between the Pelton wheel and the steam wheel. Resolution of this problem required replacement of all Pelton wheels with a higher-strength material and a reduction in the operational speed of the Pelton wheel drives. Reduction of the Pelton wheel drive speed requires a change in the plant technical specifications, and an analysis was performed to demonstrate that the decay heat removal capability remained within limits specified in the Final Safety Analysis Report.

The cracks in the coupling between the Pelton wheel and the steam wheel have been determined to be a result of circulator testing at the Valmont Test Station. During these tests, the circulators were operated with steam conditions resulting in sonic flow and high acoustical pressures. These conditions cannot occur in the Fort St. Vrain plant.

3. PLANT SYSTEMS

Many of the plant systems, with the exception of the steam system, have been extensively tested. Highlights of these tests are summarized below.

3.1. Hot Functional Tests

The basic objectives of the Hot Functional Tests were to evaluate the performance of the primary coolant system and the internals of the PCRV of the Fort St. Vrain HTGR. These tests included operation of the four helium circulators under various steam conditions and a determination of the PCRV thermal barrier insulation effectiveness. They further provided a means of testing all PCRV internal components at essentially full-load, cold-side primary coolant temperature (625°F).

The four helium circulators were driven by special oversize Pelton water turbines (non-nuclear) during the Hot Functional Tests so the heat of compression could provide the required energy input to raise the primary coolant temperature to approximately 625°F. These tests also served to evaluate the acoustic response of the system and to determine that no flow-induced vibration problems exist in any of the various areas of the steam generator modules.

Parallel operation of the circulators demonstrated that there was minimal interaction between circulators. Over a wide range of speeds, the circulator discharge valve closed smoothly whenever the circulator speed was reduced to the stall point and reopened with only a small increase in speed above that of the other operating machines.

In addition, the procedure for restarting one circulator with the other three operating to achieve balanced flow in both primary coolant loops was demonstrated. The data obtained from these tests showed that such a restart could be accomplished with a minimum perturbation of the overall plant process control system.

3.2. Vibration survey

Acoustical energy incident upon internal structure was measured during the Hot Functional Tests using microphones placed at strategic locations inside the primary coolant system. In addition, more than 100 thermocouples and strain gages were located on the steam generator module shrouds, inlet bellows, steam tubing, gas guides, and shroud and insulation cover sheets. A step-wise increase to maximum helium flow was established using two circulators in one loop. During this test, responses from the microphones and strain gages were monitored over a range of frequencies using high-speed magnetic tape recorders and a spectrum analyzer. Results from these tests revealed that acoustically induced stresses in the PCRV internal structures and components were within design criteria and that no flow-induced vibrations of any significance developed.

3.3. PCRV cooling

One of the purposes of the Hot Functional Tests was a determination of heat losses from the primary coolant into the surrounding PCRV liner and concrete.

The flow data were obtained from an automatic temperature scanner/printer, analyzed by a computer program, and then used to determine the proper piping configuration and inlet valve adjustments required to obtain a uniform temperature rise across all cooling tubes.

The liner cooling system was designed as two independent loops, each of which is capable of removing 100% of the design heat load passing through the liner. Results from the Hot Functional Tests confirmed the design capacity of the system.

3.4. Feedwater dynamic stability

A series of dynamic response tests were performed on the feedwater/preboiler secondary coolant system to determine the dynamic characteristics and initial settings of the control system components.

The tests included step changes in flow demand, simulated dump of one loop with the attendant two-phase flow, and the rapid (0.2-sec) isolation of one of the two feedwater loops. The only changes made as a result of the test were minor modifications to the boiler feed pump speed controllers. The tests confirmed the adequacy of the control system of the secondary coolant.

4. FUEL AND REACTOR CORE

The initial core for Fort St. Vrain was manufactured in the General Atomic Fuel Fabrication Facility at San Diego. Even though the core has only been operated to about 2% power, a great deal of production experience and operational information has been gained from fuel manufacturing and the testing programs. Nuclear and performance characteristics of the core have been determined, and close agreement between measurements and design predictions has been obtained.

Detailed results of the tests performed on the reactor core have been discussed in various reports and papers [3-6]. Only the highlights of the test results will be discussed here.

4.1. Core fabrication

During fabrication of the initial core, changes in the calibration of the heavy metal contents in the fuel rod standards used to assay fabricated fuel rods were necessary part way through the fabrication and caused reevaluation of the metal loadings of some already assembled elements. As a result, special loading adjustments were required in the remaining elements to satisfy both power distribution and reactivity requirements, ultimately leading to a specification of specific core locations for some of the initial core fuel elements. Later determinations of initial core criticality and flux distribution measurements confirmed that these adjustments were performed properly without degradation of core performance.

Table I compares as-built values, determined from production quality control measurements, and the initial core fuel specifications. Both fuel loading and contamination satisfied the specifications.

4.2. Fuel loading and handling

The Fort St. Vrain reactor core was loaded in air using special hoists on the refueling floor to insert fuel and reflector elements into the PCRV. Personnel in the core cavity positioned the elements into the correct core locations. This method of fuel loading was demonstrated to be extremely efficient. The 1482 fuel elements, 1292 side and top reflector elements, and more than 100 temporary poison rod sections were installed within 3 weeks. Data-taking in the control room for determining core multiplication changes was the time-limiting factor.

Testing of the normal fuel handling equipment was accomplished with core temperatures at refueling conditions after completion of the zero-power physics tests. During the fuel handling test, a complete refueling region was removed from the core and reinserted. The fuel handling machine performed flawlessly through the entire test. The only problem encountered involved the sticking together of fuel elements caused by temporary marking paint which had been used to identify the fuel elements.

The core geometry and gaps between the hexagonal core fuel columns were measured at various core elevations during the core loading. The gaps along several core traverses were measured, and no discrepancy between measured and design tolerances was found. These measurements confirmed that the core support, side reflector, and fuel elements had been manufactured and installed within design tolerances.

4.3. Physics tests

Extensive physics testing was performed after fuel loading. The zero-power physics test program was directed toward verifying the calculational methods employed during the core design. Highlights of the test results are shown in Table II and Figs. 3 and 4. Measurements of control system worths using pulsed neutron techniques were in excellent agreement with predictions, not only for the total shutdown margin but also for the worths of single rod groups. Considering the asymmetry of the heavy metal loading in the Fort St. Vrain core, the data obtained clearly indicated that the shutdown margins, maximum worth rods, and rod-group worth data can be predicted with the accuracy required for design and operation.

The flux distribution profiles shown in Figs. 3 and 4 are typical of the many axial traverses taken during the initial test phase using 12-in.-long, B^{10}-lined proportional counters. Calculational results shown were obtained from three-dimensional diffusion analyses using the measured fissile and fertile content of the specific fuel elements in each core location.

The hot physics tests with and without power operation allowed the determination of temperature defect and temperature coefficients (Fig. 5) and measurements of the radial power distribution.

Radial peaking factors (RPF) were measured for each refueling region at about 2% core thermal power. Heat balances were made on each region by measuring the region coolant temperature rise with the flow control valves set to provide equal mass flow per coolant hole to each region. Measurements of region RPF were made for four coolant flow rates between 7.4% and 12.7% of rated full-power flow and with flow valves set at two positions.

Figure 6 compares the measured RPFs and those calculated by two-dimensional analysis for the as-built core. Error bars on the data points indicate that the measurements have a standard deviation of 20% due to uncertainties in the measured parameters which results from the small values of region temperature rise at this low power level.

Although calculations lie within one standard deviation of the measured value in nearly every case, a more definitive demonstration that the power distribution meets all criteria must await more precise peaking factor measurements planned at higher power levels.

Measurements of average core coolant temperature rise indicated that flow rates used in the core design for core bypass flow through the side reflector, control rod channels, and interblock gaps are quite conservative for the initial core.

Overall, the core physics test programs to date have been extremely successful. Pertinent significant conclusions are:

1. The use of pulsed neutron techniques allowed a fairly accurate evaluation of the full core shutdown margin as well as the determination of the worth of single rod pairs and control rod groups in subcritical configurations. The theories developed for application of this test method to large graphite reactors were demonstrated to be sound. The use of on-line data acquisition and analysis proved to be effective and time saving.

2. Predicted core shutdown margins and control system worths agreed with the measured values to within 0.004 Δk or better for all 12 subcritical core configurations evaluated. The largest discrepancy occurred for cases where the core multiplication was less than 0.95.

3. The discrepancy between actual and predicted core criticality was 0.003 Δk. Subsequent analysis of the as-assembled core configuration showed agreement between theory and test of 0.001 Δk. This agreement corresponds to less than 2 ft of discrepancy in the position of one control rod pair (out of 37) and to less than 1% in fissile or fertile material loading.

4. Axial flux distributions were measured in selected core channels. These locations were chosen to be in the vicinity of partially and fully inserted rod pairs because these are most difficult to predict. Agreement between calculations and measurements was again excellent, indicating that multidimensional physics analysis has been developed to a state which enables the detailed behavior of complicated systems to be predicted with good accuracy.

5. The temperature defect and temperature coefficient measurements (Fig. 5), over the temperature range tested to date, show excellent agreement with predictions. Thus, the important inherent HTGR safety feature of a negative temperature coefficient can be accurately predicted.

The close agreement between predicted and actual physics results has clearly demonstrated that the nuclear characteristics of the HTGR are well understood. The powerful computational methods that have been developed during the last 19 yr, supplemented by the extensive nuclear data banks which form the bases for predicting the nuclear behavior, can be employed with confidence in HTGR core design.

4.4. Fission gas release measurement

Fission gas inventories were measured in the primary coolant during low-power operation. These inventories are a sensitive measurement of the quantity of uranium contamination in the fuel elements introduced by manufacturing and handling operations. Measurements showed a release fraction (R/B) of 5×10^{-6} for ^{85m}Kr and 4×10^{-6} for ^{138}Xe. As had been anticipated, release rates showed little dependence on half-life or chemical element type. The release fractions observed at 2% power were not significantly different from those predicted.

Fission product release rates from the fuel had been limited during production by acceptance standards on quality control test results called out in the fuel specifications of heavy metal contamination in fuel rods and fractional release of ^{85m}Kr at 1100°C from fuel rods. Table I shows that the initial core as-manufactured satisfied all the specifications. Tests on historical samples indicate that storage of fuel in shipping containers at the reactor site and simulated exposure to moisture-duplicating conditions in the core during the major moisture ingress incidents discussed previously increased the fission gas release over the as-manufactured fuel by about 60%. Estimates of initial fission gas release were made using noble gas release data taken periodically on production rod samples preserved for this purpose.

Calculations which project the initial measurements to full-power operation indicate that fission product release will be well below the level for which the plant was designed.

5. CURRENT STATUS

In early May 1975, while performing the first step in the rise-to-power test program at a nominal 2% power level, it was discovered that the top head liner penetration cooling system was unable to maintain the design temperature for the 37 refueling penetrations. Subsequent investigations showed that a small amount of helium was flowing up into the control device. The flow of hot helium caused a local volume of concrete at the junction of the top head liner and the penetration liner to exceed the conservatively low maximum concrete temperature. This high temperature in the bordering concrete presented no danger to the integrity of the vessel, because the liner by itself has adequate strength and does not require the concrete backing in this particular area. After the flow path of the helium through the control rod drive was determined, a combination of seals was designed to reduce the helium flow through the control rod drive.

In May 1975, concurrent with the top head cooling problem and while waiting for NRC permission to proceed above 2% power, a concern about electrical cable segregation developed. An investigation revealed that the routing of some essential and nonessential cables used at Fort St. Vrain was not consistent with the Final Safety Analysis Report criteria, and a full audit of the cabling was initiated. The audit of the cabling has been accomplished, and completion of rerouting of cables is expected in late 1975.

6. CONCLUSIONS

From the time the construction permit was issued to the end of construction, the Fort St. Vrain project progressed on or ahead of schedule. Subsequent schedule slippages have resulted from problems that developed during the startup and test program.

The thorough test program, though progressing more slowly than planned, has been quite successful and has provided important feedback to the ongoing design of large HTGR systems. Major component systems and design concepts characteristic of the HTGR have been demonstrated to be sound and reliable, despite the minor setbacks encountered.

Reactor core tests to date have demonstrated again the thorough understanding of the physics of the HTGR and confirmed predictions in fuel performance and design parameters. Fort St. Vrain will reach full power in the near future, and it will contribute an important milestone to the commercialization of the HTGR.

ACKNOWLEDGMENT

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TABLE I
INITIAL CORE FABRICATION

	Specification	Actual
U Loading	773.2 kg	774.1 kg
Th Loading	15,971 kg	15,905 kg
Th Contamination	6×10^{-4}	1.9×10^{-4}
Fission Gas Release from Fuel Fuel Rods		
^{85m}Kr R/B at 1100°C	3×10^{-5}	2.8×10^{-5}

TABLE II
CONTROL SYSTEM WORTHS, ΔK

	Predicted	Measured
Full Shutdown Margin	0.098	0.096
Max. Worth, 1 Rod Pair	0.042	0.036
Max. Worth, 2 Rod Pairs	0.081	0.077
Rod Group 3C (3 rod pairs)	0.036	0.034
2A (3 rod pairs)	0.035	0.037
4B (3 rod pairs)	0.008	0.008
4F (3 rod pairs)	0.010	0.009
Critical Core Configuration, k_{eff}	1.001	1.000

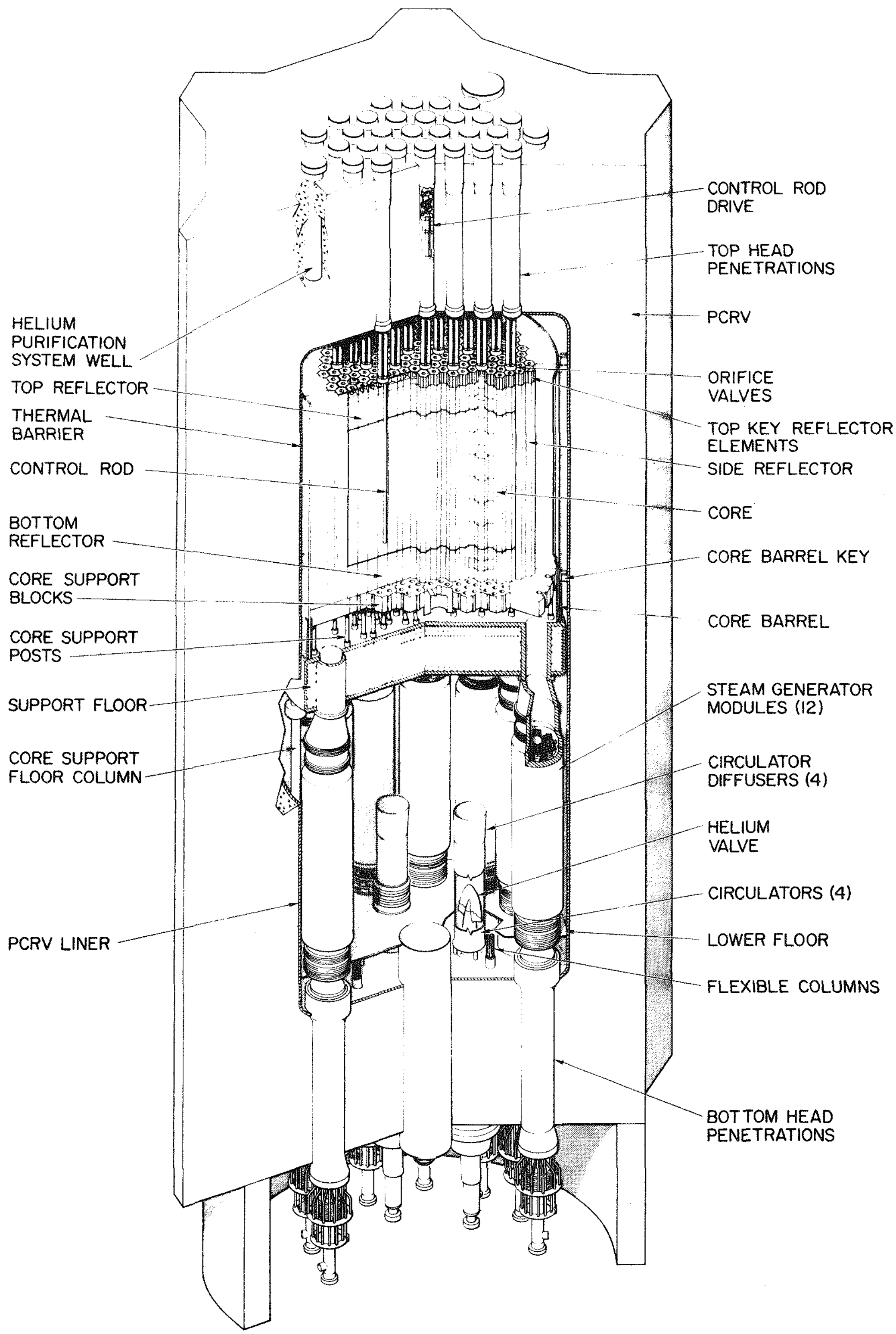

Fig. 1. Primary reactor system of Fort St. Vrain HTGR

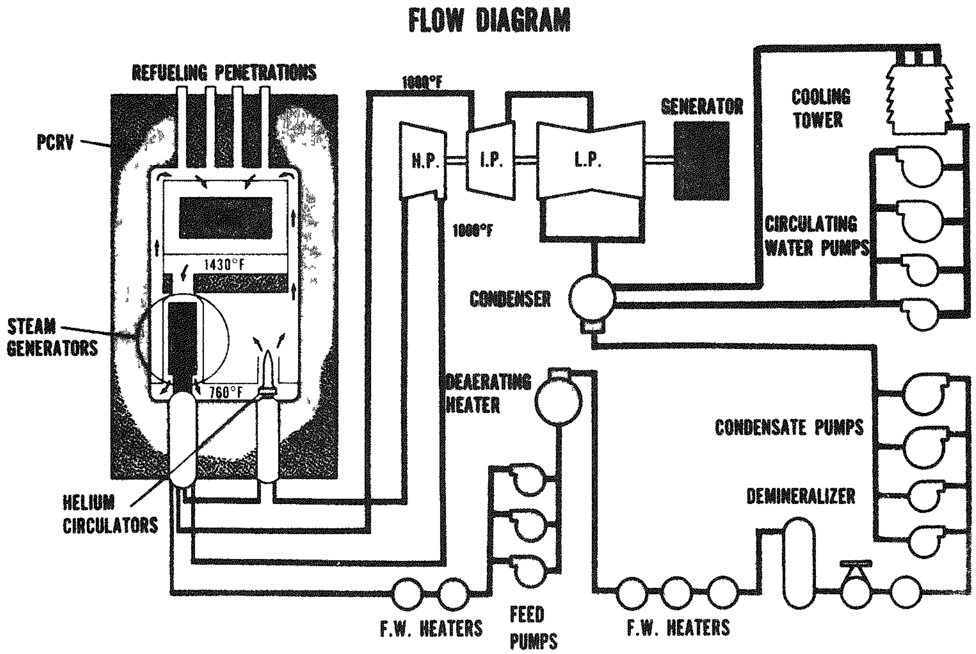

Fig. 2. Simplified Fort St. Vrain Nuclear Generating Station cycle flow diagram for turbine generator throttle conditions of 2400 psig/1000°F/1000°F

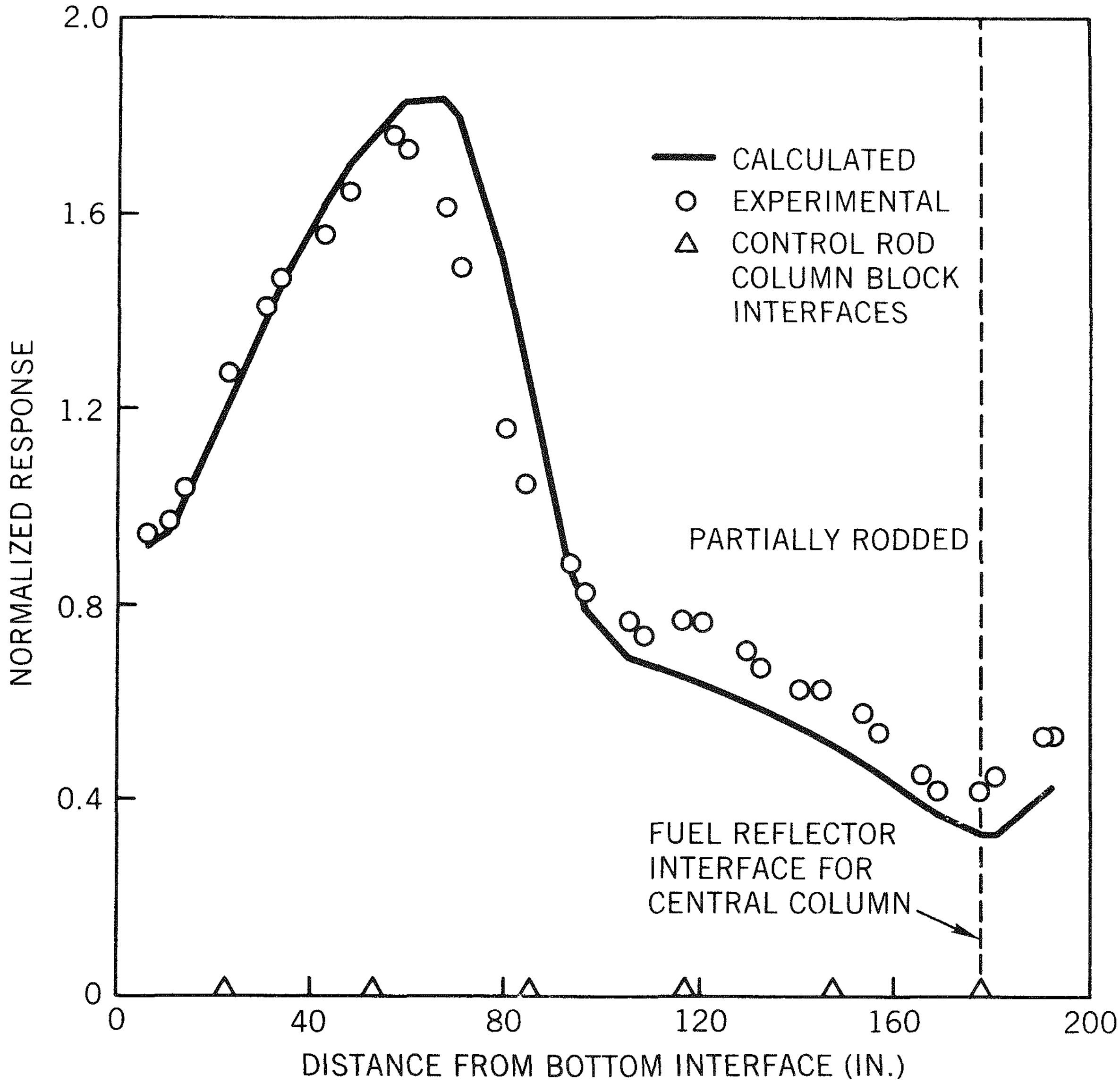

Fig. 3. Comparison of measured and calculated axial distributions, Configuration 1, Region 3, partially rodded region

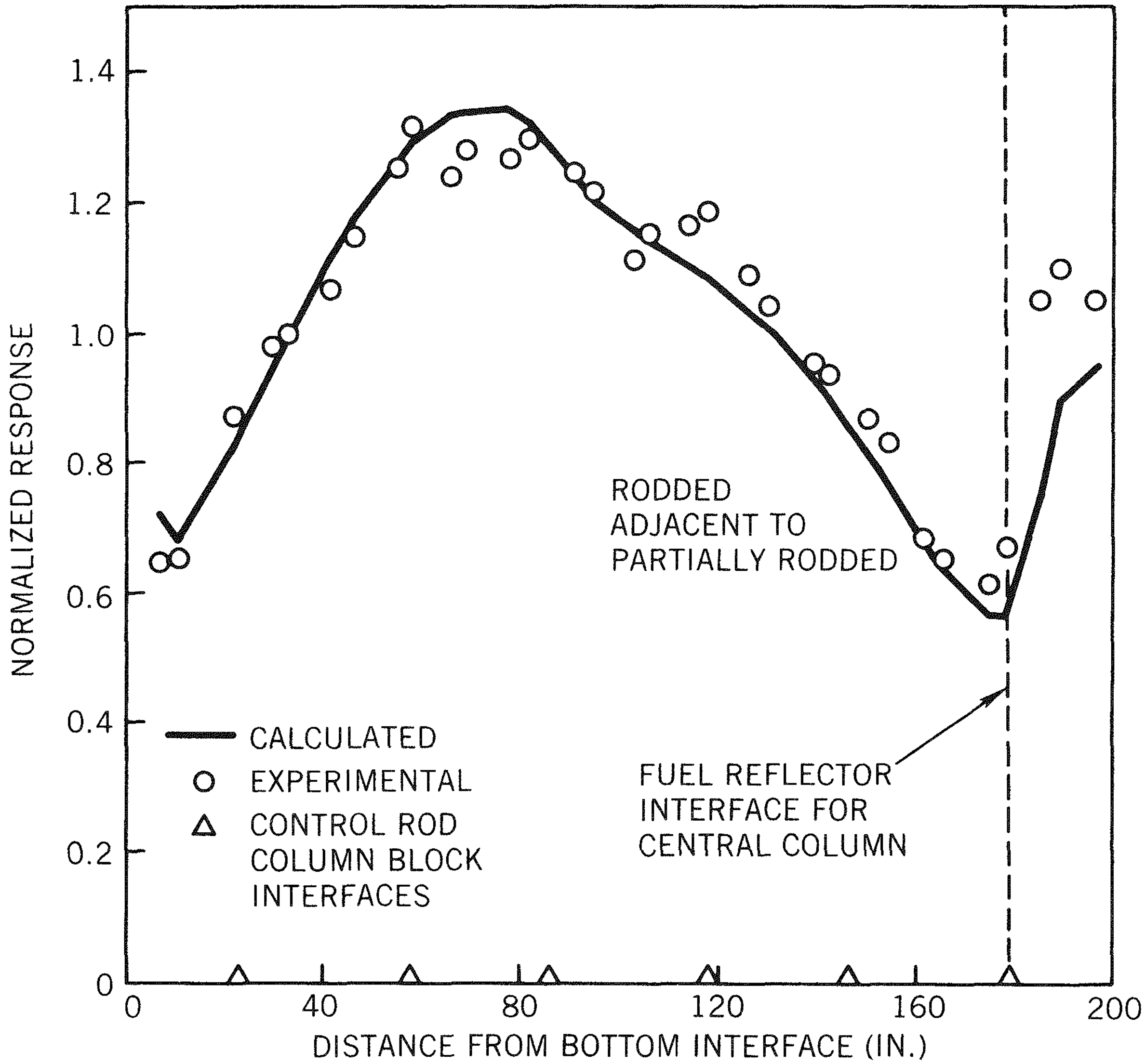

Fig. 4. Comparison of measured and calculated axial distributions, Configuration 1, Region 1, rodded region adjacent to a partially rodded region

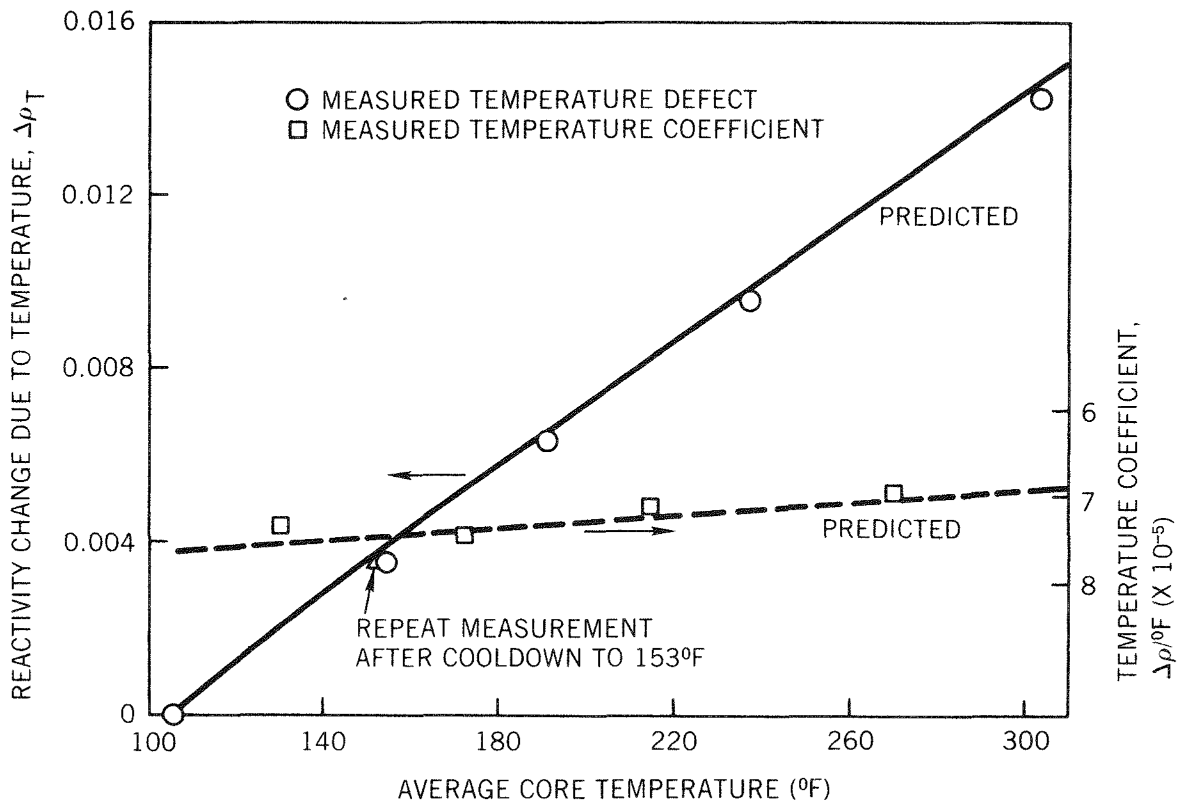

Fig. 5. Temperature defect and temperature coefficient (isothermal)

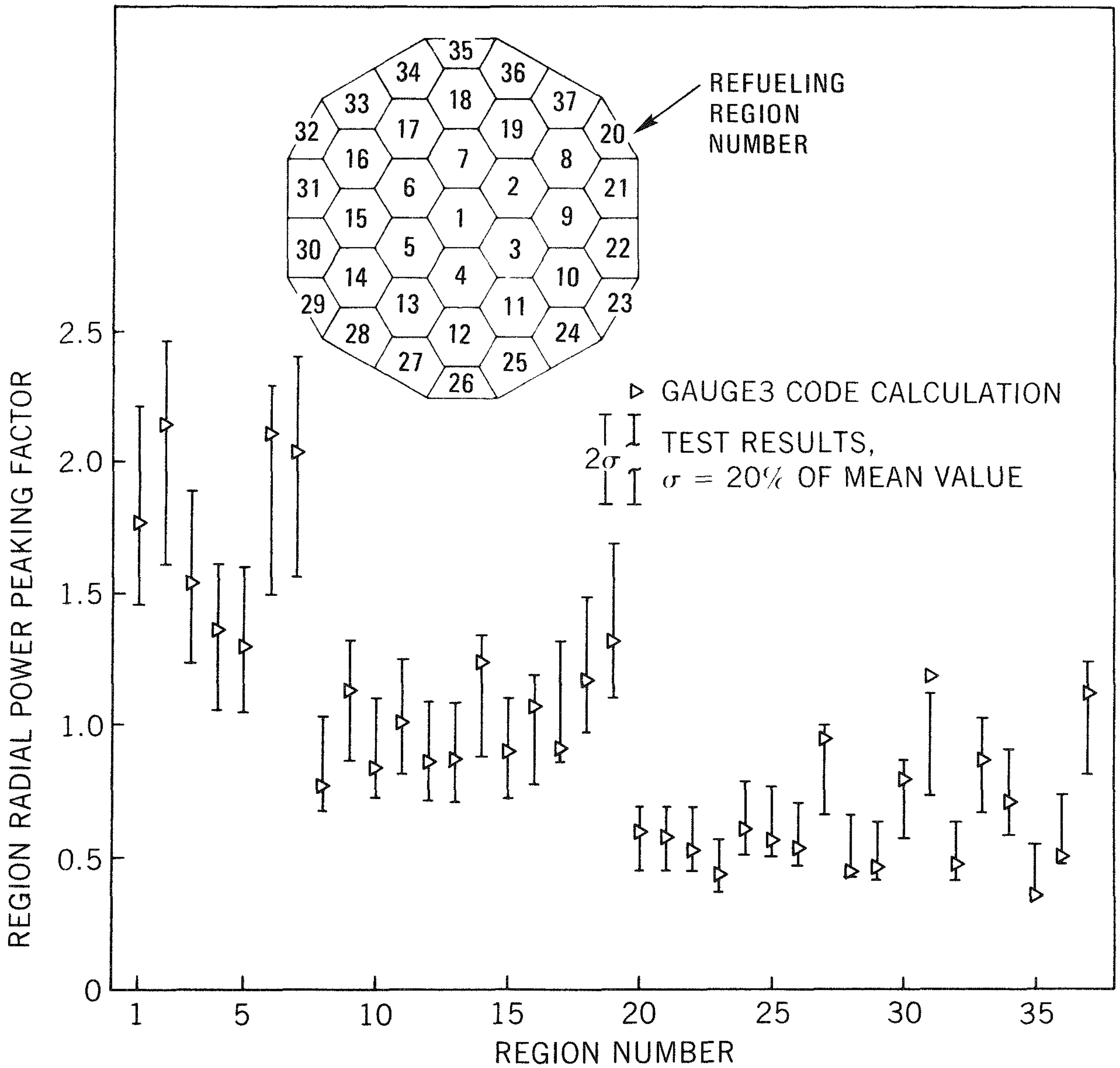

Fig. 6. Results of RPF measurement at 2% power for Fort St. Vrain HTGR