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**CAROLINAS VIRGINIA
NUCLEAR POWER ASSOCIATES, INC.**

CVNA-287

PARR, S. C.

CVTR FUEL MANAGEMENT

Work Performed By
WESTINGHOUSE ELECTRIC CORPORATION
Atomic Power Divisions

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CVTR FUEL MANAGEMENT

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ABSTRACT

Based upon the performance of the core, it is apparent that the analytical methods and models developed for use throughout the fuel management program were quite adequate. Due to an extensive Research and Development Program, most of the analytical predictions were later measured during operation. The good agreement demonstrated that the analytical techniques applied to this core are applicable to the larger heavy water cooled and moderated pressure tube power reactors.

The calculated lifetimes of the two refueling operations were in excellent agreement with plant performance. In addition, the predicted power distributions as a function of both core depletion and control rod withdrawal were in good agreement with the measured values.

In conjunction with the fuel management program, four special high power density test assemblies were inserted into the core. Operation of these assemblies provided additional knowledge of the performance of high power density fuel. During the tests, these assemblies developed leaks which led to their ultimate withdrawal significantly prior to their planned burnup exposure. Subsequent to the removal of all the test assemblies the decision was made to terminate CVTR operation.

1.0 INTRODUCTION

The post-construction research and development program* associated with the fuel management of the Carolinas Virginia Tube Reactor (CVTR) has yielded a wealth of information concerning various aspects of long term operation of a heavy water moderated and cooled, pressure tube reactor. It is the purpose of this report to record that part of the experimental information pertaining to the nuclear evaluation of the fuel management program following initial core burndown and, where possible, to compare the experimental observation to the analytical predictions. Accordingly, Section 2.0 summarizes the nuclear design and analytical methods employed throughout the fuel management program. Section 3.0 records the experimental program results and Section 4.0 compares these values with the design predictions. A discussion of the special high power density test assemblies is given in Section 5.0. The measurement problem areas encountered during this phase of the program are discussed in Section 6.0 and finally a history of the failed fuel occurrences is presented in Section 7.0.

Throughout the fuel management program, all subsequent refueling operations consisted of replacing certain spent fuel assemblies with standard fuel enriched to 2.0 w/o in U-235. The out-in cycling scheme was used with Cycle 1 consisting of eight fresh fuel assemblies and Cycle 2 consisting of 12 fresh fuel assemblies. In conjunction with the first cycle, four of the eight assemblies were special high power density test assemblies. Depleted fuel from the central region of the core, which was initially enriched to 1.5 w/o, was withdrawn with each cycle. During Cycle 2 operation three of the special high power density test assemblies developed leaks and a detailed account of the test assembly operation is presented in Section 5.6.

1.1 Description of the Core

The Carolinas Virginia Tube Reactor (CVTR) is a power demonstration prototype, heavy water moderated and cooled, UO_2 fueled, pressure tube reactor. The core consists of 36 cold wall Zircaloy-4 pressure tubes

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(U-tubes), where the pressure tube walls are immersed in the moderator and are thermally insulated internally from the higher temperature coolant. Each pressure tube contains 2 fuel assemblies forming a reactor core of 72 fuel assemblies. Provision was made in the original design of CVTR for possible future expansion of the core to a total of 84 fuel assemblies. Figure 1.1 shows a plan view of the CVTR final design of 72 pressure tubes and the six pressure tubes which were deleted. Each pressure tube provides a path for the D_2O coolant in addition to housing two fuel assemblies.

The core was designed to be controlled with 28 rectangular shaped control rods with possible expansion to 32 as depicted on Figure 1.1. The control rods are operated in seven groups of four rods per group. The neutron absorption characteristics of the control rods are separated into two categories having twelve black (borated steel) and sixteen grey (304 stainless steel) rods. The black rods were used to account for the excess reactivity installed in the initial core. After sufficient depletion, four of these rods were removed from the core (Group VI) leaving eight black and sixteen grey control rods for use throughout subsequent power operation. Initially, the black rods were used to account for the installed excess reactivity and during subsequent fuel cycles, when the installed excess reactivity was small, the black groups were used as shutdown rods. Throughout all power operation of CVTR, only the grey rods were used for reactor control (i.e., as moving groups).

The CVTR standard fuel assemblies contain nineteen individual fuel rods arranged in a hexagonal array with a 0.600 inch triangular pitch as illustrated on Figure 1.2. The fuel rods consist of UO_2 in the form of cylindrical pellets sintered to approximately 93% theoretical density and are clad with Zircaloy-4. The initial core consisted of fuel rods enriched to 1.5 w/o and 2.0 w/o in U-235. All replacement fuel was enriched to 2.0 w/o with the exception of four special high power density test fuel assemblies which were inserted into the core as part of the first refueling operation. Summaries of the physical dimensions of the initial CVTR core are given in Table 1.1. More detailed information of the reactor core and the original design of CVTR is presented in "The Nuclear Design of the CVTR Core."¹

CAROLINAS VIRGINIA TUBE REACTOR

PLAN VIEW

LEGEND

⊗ 1.5% ENRICHED FUEL	⊕ NEUTRON SOURCE
○ 2.0% ENRICHED FUEL	• FLUX MONITORING WIRE
⊙ FLOW INLET	■ BLACK CONTROL ROD
⊙ FLOW OUTLET	□ GREY CONTROL ROD

Figure 1.1 Plan View of CVTR Core

Figure 1.2 Cross Section of CVTR Pressure Tube and Fuel Assembly

TABLE 1.1
SUMMARY OF CVTR DESIGN CHARACTERISTICS
FOR 65 MW GROSS FISSION POWER

Core Description

Average core diameter (cold)	69.04 inches
Active core length (cold)	95.4 inches
Number of fuel assemblies	72
UO ₂ in the core	6852 lbs.
Uranium enrichment (24 assemblies)	1.5 w/o
(48 assemblies)	2.0 w/o

Fuel Design Data (Cold Dimensions)

Pellet diameter	0.430 inch
Pellet length (1.5 w/o)	0.530 inch
(2.0 w/o)	0.636 inch
Average UO ₂ density	10.28 gm/cm ³
Fuel tube material	Zircaloy-4
Fuel tube O.D.	0.487 inch
Fuel tube I.D.	0.436 inch
Fuel tube center-to-center pitch (triangular)	0.6 inch
Number of fuel rods per assembly	19
Total number of fuel rods	1368

Pressure Tube Data

Number of U-tubes	36
Outside diameter of pressure tubes, inches	3.94
Inside diameter of pressure tubes, inches	3.56
Tube pitch, rectangular, inches*	8 x 6.5
Number of thermal baffle regions (D ₂ O)	4
Thickness of circular thermal baffle gaps, inches, (avg.)	0.042

*The value 6.5 is an average of 7.5 and 5.5.

TABLE 1.1 (cont'd.)

Control Rod Data

Number of control rods	28	
Control rod types	Black	Grey
Number of each type	12	16
Control rod materials	Borated Steel	304 Stainless Steel
Control rod shape	Rectangular	Rectangular*
Control rod size, inches	1 x 4	1 x 4

Heat Output, MW Thermal

Heat output from fuel rods	61.7
Nominal heat loss to moderator with 4 baffles	2.2
Heat generated by neutrons and gammas in moderator (5% of gross fission power)	3.3
Gross heat release from fission	65.0

Coolant Flow, lb/hr**

Total coolant flow rate	3.3×10^6
Average flow per tube	9.17×10^4
Maximum flow per hot tube	10.5×10^4
Moderator flow rate	4.04×10^5

Pressure, Psia

System pressure, nominal (outlet of core)	1500
---	------

Heat Transfer

Active core surface area, ft ²	1387
Average flux, BTU/hr ft ²	0.152×10^6
Nominal steady state maximum flux, BTU/hr ft ²	0.495×10^6

*Perforated to reduce volume by 30%.

**Coolant flow rates shown are dependent on the origicing pattern.

TABLE 1.1 (cont'd.)

Design Hot Channel Factors

Radial, F_R^N	1.525
Axial maximum to average, F_Z^N	1.5
Maximum to average rod in cluster, G_o	1.25
Flux (F_q^E) engineering	1.045
Thermal flux skew across hot rod, F_{SKEW}	1.09
(F_q^T) total*	3.26

Temperature, °F

Moderator, average	155
Moderator, inlet	130
Moderator, average rise	50
Coolant, average in core	531
Coolant, inlet to pressure tubes	505
Coolant, discharge from pressure tubes (avg.)	556
Coolant, average rise in core	51

Nuclear Data

Effective multiplication (beginning of life)	
Hot, power, equilibrium Xe and Sm poison	1.201
Hot, power unpoisoned	1.244
Hot, no power unpoisoned	1.204
Cold, no power, unpoisoned	1.291
Estimated average control rod worth	
Grey	0.8%
Black	1.5%
Full power lifetime, hours, design minimum	8500
Core average flux levels, neutrons/cm ² -sec	
Thermal (0 ev to 0.625 ev)	4.3×10^{13}
Fast (0.625 ev to 10 Mev)	5.8×10^{13}

*Applicable for computing peak heat flux and DNB ratio. For fuel rod power density, use $F_q^T/F_{SKEW} = 2.99$.

TABLE 1.1 (cont'd.)

Reactivity coefficients (beginning of life)

Temperature coefficients of reactivity ($^{\circ}\text{F}$)⁻¹

Fuel	$(-0.2 \text{ to } -0.8) \times 10^{-5}$
Coolant	-2.1×10^{-5}
Moderator	$(-0.8 \text{ to } -1.3) \times 10^{-4}$
Pressure coefficient of reactivity	$(+1.9 \text{ to } +2.1) \times 10^{-7} \text{ psi}^{-1}$
Pressure tube void coefficient of reactivity	$(-1.9 \text{ to } -2.1) \% \text{k}/\% \text{ void}$
Delayed neutron fraction including photoneutrons	0.70%
Prompt neutron lifetime (hot core)	$5.6 \times 10^{-4} \text{ sec.}$

During operation, the average moderator temperature is 155°F and the average temperature of the primary coolant is 531°F at a nominal gauge pressure of 1500 psi. The reactor transferred 59.4 MW of heat to the secondary system at the steam generator at rated steam and feedwater conditions. This corresponds to a heat output from the reactor of 65.0 MWt.

1.2 Core 1 Operation

Throughout Core 1 operation, CVTR proved the heavy water moderated and cooled pressure tube reactor design concept for a power reactor. The reactor became critical on March 31, 1963. After criticality, an extensive period of startup and Research and Development testing followed which included initial power operation on December 16, 1963. Power production for commercial applications was initiated on May 1, 1964. Power capability was raised from 44.3 MW to 65 MW in September 1965. Core 1 operation was completed on March 4, 1966 at which time the refueling operations commenced.

It was apparent from the results obtained during the operation of Core 1 that the analytical methods and models applied to the investigation of this core, although cumbersome, were quite adequate. Therefore, these methods were carried over to the fuel management program. The analytical model represented by the TURBO*¹² computer program was employed throughout Core 1 operation and subsequent refueling operations. Predicted power distributions, as a function of both core depletion and control rod withdrawal, were in very good agreement with the measured distributions. The predicted total Core 1 lifetime, 9450 MWD/MTU, was in good agreement with the observed value, 10,100 MWD/MTU. A more detailed description of Core 1 operation is given in the "CVTR Core 1 Nuclear Operations Follow Summary."¹³

1.3 Description of Fuel Movement Operation

The CVTR fuel and pressure tube handling facilities inside the vapour container consisted of the refueling machine, the rotating secondary shield, the fuel transfer mechanism, and the pressure tube drying facility.

The refueling machine consists of a lead-shielded cask, equipped with cooling and hoisting apparatus, mounted on a crane-type vehicle. Cooling is provided by a self-contained system which circulates heavy water through a heat exchanger cooled with light water. Reactor components are handled with pneumatically operated grippers and lifted with an electrical hoist. The grippers are guided through the header cavity by two guide tracks which are retracted into the cask cavity by the grippers. The machine is operated from a control station located on the crane trolley. A closed circuit TV system was employed to aid in positioning the refueling machine.

The rotating secondary shield over the header cavity provides radiation protection for operating personnel during the transfer of spent fuel between the core and the refueling machine. A portion of that shield consists of a rotating disc designed to provide an access hole above any portion of the pressure tube array. The disc, which is filled with lead shot and boron carbide to a thickness equivalent to 12 in. of lead, is supported by wheels on a circular track. It is rotated by a pneumatic drive motor recessed in the operating floor or by a removable hand crank. The access hole is provided by an arrangement of movable shielding blocks along a radial slot in the disc. The blocks are moved by a remotely operated air cylinder. During reactor operation and other periods when access to the equipment in the header cavity is not required, the rotating shield is sealed to the operating deck with an inflatable rubber gasket, and a gasketed cover is placed over the radial slot, to stop the leakage of helium to the container atmosphere. The peripheral seal is released during refueling operations, but a secondary sliding lip seal limits the inleakage of air to the header cavity to ensure a positive downward flow of air through the access hole.

The fuel transfer mechanism is an underwater conveyor which was designed to transport irradiated fuel assemblies, pressure tubes, neutron streaming plugs and control rods between the fuel transfer canal to the vapor container and the spent fuel basin. The mechanism works through a penetration sleeve in the vapor container wall which is closed by a gate valve when the reactor is in operation. The apparatus includes a carrier car and a conveyor car which operate on tracks through the penetration

sleeve; a drive winch, which moves the conveyor car by means of a system of wire ropes; and two lifting frames, one in the fuel transfer canal and one in the spent fuel basin. The lifting frames are winch-operated, and are used to tip the carrier car to a vertical position for loading and unloading. The carrier car and load are pulled through the penetration sleeve by a drawbar attached to the conveyor car. In their extreme position of travel, only the carrier car and drawbar extend through the penetration to the transfer canal side. The water in which the mechanism is submerged provides for in-transit shielding and cooling of spent fuel assemblies. Concrete and steel floor slabs covering the fuel transfer canal have an equivalent shielding capability to 12 in. of concrete plus 12 in. of steel in the vicinity of the fuel discharge point for protection of personnel during lowering of a spent fuel assembly from the refueling machine.

The pressure tube drying facility consists of a shielded drying station; radiant electric heaters to heat the pressure tubes; ducting to convey air with light water from the external surfaces of pressure tubes to the vapor container purge system; a vacuum pump to remove heavy water from internal surfaces; and flexible hoses for vacuum and air connection to the pressure tubes. The drying station is located in the southwest corner of the reactor structure below the operating floor. It consists of an insulated thimble suspended within the shielded chamber. The thimble can accommodate four pressure tubes. Radiant strip heaters are provided within the insulated thimble, and air in-drawn through the thimble to evaporate moisture from the external surfaces. Internal surfaces are dried by the application of a vacuum and by external heat. The drying station is provided with two shielding plates. The lower plate surrounds the pressure tubes below the connector body and provides protection to the operators when access is required to the hose connections. Port plugs are inserted in this shield with the refueling machine when a pressure tube is removed, to prevent radiation streaming from adjacent tubes. The upper shield plate protects against oblique radiation during transfer of a pressure tube to or from the refueling machine.

The actual fuel movement operations conducted in the CVTR consist of removing spent fuel assemblies from the core, shuffling irradiated fuel assemblies from outside to inside core positions, and the insertion of fresh fuel assemblies into the core.

The initial step in the refueling operation is the removal of the two fuel port caps from one of the U-tubes containing fuel to be removed from the core. These fuel port caps are stored in the vapor container to be reinstalled at the completion of refueling. The neutron streaming plugs are then removed from the U-tube and stored in the spent fuel pit. The fuel port caps are removed from the second U-tube to be defueled and installed in the first U-tube and the neutron streaming plugs removed.

This process is continued until the neutron streaming plugs have been removed from each of the U-tubes to be defueled. The refueling machine is then used to remove the fuel assemblies, in sequence, from the U-tubes and transfer them to the fuel transfer mechanism to be transferred to the spent fuel pit. Cooling of the spent assemblies is provided by the refueling machine during movement to the transfer mechanism, and by the water in the mechanism canal and spent fuel pit. The defueled U-tubes are then removed to the drying station where the heavy water primary coolant is removed. When dry, the U-tubes are transferred to the spent fuel pit through the fuel transfer mechanism.

Shuffling of irradiated fuel assemblies from outside core positions to inside core positions, is accomplished by transferring the U-tubes containing the assemblies to be transferred. The jumper port caps are removed from the U-tube to allow for the attachment of the refueling machine cooling attachment. Neither the fuel port caps nor the neutron streaming plugs are removed during the transfer. The U-tube is then removed from the reactor by the refueling machine and moved to the specified reactor position.

The insertion of fresh fuel into the reactor is accomplished by loading the fresh fuel into U-tubes and inserting them in the proper core position. Fresh fuel is loaded into previously irradiated pressure tubes in the spent fuel pit. The previously removed neutron streaming plugs are

also placed in the U-tube in the spent fuel pit. The U-tube is then moved by the fuel transfer mechanism into the vapor container and then to the drying station where all light water from the spent fuel pit is removed. When dry, the U-tube is vacuum back-filled with heavy water to prevent the formation of pockets of air in the baffles. The U-tube is then moved to the specified location in the reactor, bolted down and the fuel port caps installed.

1.4 Description of Fuel Management Program

The fuel management program for the CVTR consisted of two complete fuel cycles which entailed eight fresh fuel assemblies for Cycle 1 and twelve fresh fuel assemblies for Cycle 2. The cycling scheme used the out-in approach where the most depleted fuel was removed from the central region and the lesser depleted fuel was sequentially relocated toward the center of the core. Fresh fuel was then placed in the vacant locations on the core periphery. This approach resulted in a flatter radial power distribution, thus reducing the core hot channel factor. In addition, economical fuel usage was maintained by maximizing the power production in those assemblies scheduled for removal during the next refueling operation.

Cycle 1 power operation commenced on May 12, 1966 and was terminated on August 31, 1966. During this time an accumulated burnup of approximately 2111 MWD/MTU was produced. Cycle 2 power operation commenced on September 15, 1966 and operated through January 24, 1967. An additional 1600 MWD/MTU was accumulated during Cycle 2 operation.

Prior to selecting the program used for CVTR refueling, a detailed study was conducted to establish the refueling enrichment and the refueling scheme which would best meet the program objectives.

Initially, the expected total energy generation in CVTR for a five year period was 31,600 EFPH*, which was to be achieved from a total loading of 6400 kg of uranium in 168 assemblies. Therefore, an average fuel burnup

*1 EFPH = 0.988 MWD/MTU.

of 13,400 MWD/MTU was required. The batch loaded core was not applicable due to a large installed excess reactivity which would be prohibitive on the basis of fuel enrichment and control requirements. In order to achieve this burnup with lower enrichments, the initial core loading was to be followed by fuel cycling with the remaining assemblies. The number of fuel assemblies involved was tentatively scheduled to be equivalent to two full core loadings. If the second core was inserted in one refueling, the burnup which would have been obtained would be 17,000 EFPD from the two cores. By adding the second core in small steps, the major portion of the first core is pushed to much higher fuel burnups increasing the power production obtained from the two cores.

Initially, the fuel cycling studies were conducted using a one-dimensional model having a cylindrical geometric representation. This model investigated the merits of the "out-in" cycling consisting of three equivalent annular fuel regions. Using the out-in scheme the center portion of the core was removed and the remaining two regions were relocated towards the central regions with fresh fuel occupying the vacant peripheral region. Calculations were conducted having feed fuel enriched to 2.0, 3.0 and 4.0 w/o U-235 in order to establish the enrichment required to give the desired core life. The results of the calculations indicated that the radial hot channel factor increased with the U-235 feed enrichment. The gross radial factors computed were considered to be greater than allowable for the 3.0 and 4.0 w/o U-235 feed enrichments. Therefore, the feed enrichment of 2.0 w/o in U-235 was selected for use during fuel cycling in CVTR. A more detailed description of these preliminary fuel cycling calculations is available in CVNA-181⁴.

Upon selection of the feed enrichment of 2.0 w/o U-235, another series of CANDIE⁵ calculations were performed in order to evaluate the merits of various "out-in" fuel cycling schemes. A total of five schemes were investigated using fuel cycles of four, eight, twelve, sixteen and twenty-four assemblies per cycle, starting at the end of life of the initial core. Each model was extended through at least three cycles with a maximum of six

cycles for the model having eight assemblies per cycle. These calculations were performed to establish the effect of the various cycling schemes on the average additional equivalent full power hours (EFPH) produced by the reactor for each new fuel assembly added to the core. In addition, the effect on the calculated radial hot channel factor was also obtained. All calculations were performed at 65 MWt.

Based on the results of the above calculations, it was evident that the average EFPH per assembly was inversely related to the number of fresh assemblies added per cycle. However, the gain in the EFPH per assembly was obtained at the expense of increasing radial hot channel factors. Since the CANDLE code uses a one-dimensional model, only trends in power peaking were considered and absolute values were later evaluated using a discrete 2 dimensional x-y model. The CANDLE results were used to select the particular schemes for further investigation with the TURBO*¹² code. Evaluation of the CANDLE calculations and subsequent TURBO* calculations resulted in a decision to choose an eight assembly out-in fuel cycling scheme as the basic cycle for operation.

TURBO* calculations were then employed to establish the specific fuel moves and control rod configurations during the fuel management program. In addition, cycle lifetimes, control rod worths, and power distributions were also established.

Cycle 1 resulted in eight fresh fuel assemblies inserted into the core. Cycle 2 consisted of inserting twelve fresh fuel assemblies in order to extend the second cycle lifetime. During Cycle 2 operation, it was necessary to replace two of the special high power density test assemblies which had developed leaks. Near completion of Cycle 2 operation, CVTR power operation was terminated. Calculations were performed for the third cycle, however, and consisted of the basic eight assembly replacement similar to Cycle 1.

Throughout operation of the fuel management program, the special high power density test assemblies caused considerable perturbations upon overall plant operation. During Cycle 1 it was necessary to limit the plant

power level due to temperature measurements incorporating the U-bend thermocouple which indicated that the test assemblies would exceed their rated power densities at plant power levels of 100 percent. It was later established that gamma heating was causing the U-bend thermocouple to give temperature readings which were considerably higher than the actual values. Upon establishing the correct power levels of the test assemblies, full power operation was resumed during Cycle 2 operation. During Cycle 2, two of the four high power special test assemblies developed leaks while operating at peak power levels of approximate 22 kw/ft. These assemblies were replaced with standard fuel and power operation was continued. One of these assemblies failed during a special pushed power test which was performed in order to increase the peak power levels of the test assemblies. During this test two standard fuel assemblies developed leaks in addition to the test assembly. A subsequent leak in a third test assembly resulted in the removal of the remaining two special test assemblies from the core. Upon removal of these assemblies, it was decided to terminate CVTR power operation several months earlier than originally scheduled.

The failed fuel occurrences in CVTR are presently being evaluated as part of another AEC contract, AT(30-1)-3850.

In summary, the fuel management program on the CVTR provided a wealth of information relating the physics characteristics of a D_2O reactor over extended periods of operation. The results of this program are presented in this report and in the "CVTR Summary Report."⁶ In addition, information relating to the performance of Zircaloy-4 clad fuel rods was established both at normal power levels and at high power levels of operation.

1.5 Outline of Failed Fuel Occurrences

During operation of the second cycle, a total of five fuel leaks were identified. Two were associated with the high power density test assemblies which were rated at peak power densities of 25.1 kw/ft and were located at core positions 3C and 8Q. The remaining fuel leaks were detected in pressure tubes 6J, 7J and 3R. Pressure tubes 6J and 7J contained the

standard Zr-4 clad 2.0 w/o enriched fuel assemblies which were inserted into the core in conjunction with the initial fuel loading of Core 1. Pressure tube 3R contained one of the 20.9 kw/ft peak test assemblies and a standard 2.0 w/o fuel assembly. It was established upon withdrawal of the pressure tube that the test assembly contained the leaking fuel.

2.0 NUCLEAR DESIGN AND ANALYTICAL METHODS

The purpose of this section is to present the analytical methods and models used throughout the nuclear design of the fuel management program on the CVTR. The basic design tool was the TURBO* computer program, which was used to specify fuel moves, control rod withdrawal programs, and power distributions. It also played a dominant role in the success of the flux wire measurements through the generation of nuclear factors⁷. During operation, discrepancies in power sharing measurements arose and the TURBO* calculations were ultimately used to establish the fractional power split in the instrumented pressure tube for the special high power density test assemblies.

Three specific fuel cycles were established on an analytical basis from TURBO* calculations. Actual plant operation consisted of only the first and second cycles; the third cycle did not materialize due to an early termination of plant operation. The design of the third cycle is presented for completeness of this report.

2.1 Objectives of the Fuel Management Program

The fuel management program on CVTR consisted of two separate considerations; that pertaining to core performance based upon standard fuel, and the operation of the special test assemblies. In some cases these objectives were contradictory in nature resulting in compromise to meet the overall program objectives.

Basically, the following items list the fuel management objectives in their relative order of importance for CVTR.

- (1) Maximize the power production per fuel assembly inserted into the core.
- (2) Maintain an acceptable radial hot channel factor for operation of the standard fuel at 65 MWt operation.
- (3) Maintain a maximum power output in the special high power density test assemblies such that a peak power production of 25.1 kw/ft can be achieved in the 4.7 w/o enriched test fuel rods.
- (4) Withdraw the most depleted fuel assemblies during cycling.

(5) Minimize the fuel movements.

(6) Control the reactor by insertion of individual grey control rod groups.

These objectives were for the most part accomplished by cycling eight fresh fuel assemblies into the core and establishing a control rod program which satisfied the power distributions desired. Due to the complexity of the fuel management program and the applied restrictions resulting from the special high power density test fuel assemblies, no attempt was made to establish an equilibrium cycle. Instead, each fuel cycle was determined based on the current objectives of the program. This allowed flexibility in the fuel moves and permitted modification when desired.

The objective of the test assemblies was to operate at as near the peak power densities for as long a period as possible. This would require operating such that the peak power density in the 4.7 w/o enriched fuel rods was 25.1 kw/ft for an accumulated maximum burnup of 40,000 MWD/MTU. Therefore, in addition to the usual high power limit which is placed upon all fuel rods, the test assemblies also had a lower limit. This limit was chosen as a compromise between convenient plant operation and test objectives. The design was based on nominal values such that the calculated peak conditions were the same as the assumed peak conditions for thermal design. Thus, there existed an equal probability that the actual peak power density within the test assembly would be above the calculated nominal value as there was for it to be below.

The design limit was based upon the peak conditions expected during any one cycle of operation. This resulted in lower test conditions during a substantial fraction of the time due to the typical change in flux distributions which occur during a cycle of operation (~ 3 months). Thus, the integrated power of the test assemblies was further reduced from the design objective. In CVTR these changes are largely due to power shift with control rod motion. An alternate approach was to design above the desired limit and derate the plant such that the peak power density may be achieved for longer time periods. During Cycle 1 operation, the first method was used and during Cycle 2 the second approach was used.

2.1.1 Plant Physical Limitations

In addition to the operating constraints listed previously, additional problems were encountered during the fuel management program due to physical limitations within the reactor core and the fuel handling equipment. In particular, all special test assemblies were designed to operate in the inlet position of the pressure tubes in order to reduce the average operating temperature of the fuel rods. Since CVTR did not maintain quarter core symmetry with respect to inlet locations, quarter core symmetry was not maintained after Core I. It was necessary to place the higher enriched fuel (4.7 w/o) at peripheral locations and the lower enrichment (3.8 w/o) nearer the center of the core. The particular locations chosen are illustrated in Figure 2.1.

The refueling procedure at CVTR required fuel moves based on pressure tubes and not individual fuel assemblies. This placed further restraints upon the fuel management program. Therefore, all analytical fuel cycling was performed having fuel moves consisting of two assemblies per move in order to maintain compatibility between the calculational mode and actual plant operation.

Consideration of the cost of heavy water required that heavy water losses and degradation be maintained at a minimum during refueling operations. For this reason, pressure tubes were dried in the drying stations to recover residual heavy water when they were removed from the reactor. Similarly, pressure tubes containing fresh fuel were also dried after coming from the spent fuel pit and before insertion into the reactor.

Irradiated fuel could not be dried when removed from the reactor due to the decay heat of fission fragments which would melt the assembly if the coolant were removed. Consequently, the heavy water in the fuel assembly was lost when the assembly went to the spent fuel pit. For similar reasons, irradiated fuel in the spent fuel pit could not be reinserted into the reactor. The assembly could not be dried to remove the light water from the spent fuel pit due to decay heat

constraints. Direct reinsertion from the spent fuel pit into the reactor core would have resulted in degradation of both the moderator and primary coolant heavy water which was highly undesirable.

A similar situation existed in the event of a fuel assembly failure. Location of the pressure tube containing the defective assembly was accomplished without removing U-tubes from the reactor. However, locating the defective assembly within the U-tube was not possible without removing the assemblies from the U-tubes, encapsulating them in failed element tubes in the spent fuel pit, and testing samples of water from the thimbles for radioactivity. The non-defective fuel assembly could not be returned to the reactor. Therefore, it was necessary to remove both elements from the pressure tube if one of them became defective.

The black control rod groups had to be the first groups withdrawn from the core to avoid bowing problems. This, of course, limited the control rod withdrawal sequence to some extent. In addition, Group VI rods have no drive mechanism and could not be withdrawn from the core during operation which further limited the control rod withdrawal

sequence. Group VI control rods were permanently withdrawn during the June 1965 shutdown with a core average exposure of 5100 EFPH. Subsequent examination revealed that in most cases the rods were bowed in the direction anticipated, based upon flux gradient considerations.

Certain limitations on core reactivity existed from both a plant physical limitation and a technical specifications limitation standpoint. The removal of Group VI control rods limited the amount of fresh fuel that could be added without exceeding a specified shutdown margin. In addition, technical specifications required that the reactor be shutdown by at least 3% $\Delta\rho$ with all rods in during refueling. Therefore, fresh fuel could not be added if the addition would exceed these reactivity limitations.

2.1.2 Enrichment Specification

The enrichment specification of the standard fuel for successive refueling operations was chosen based upon the expected termination of CVTR power operation. In addition, prior to specifying the enrichment of additional fuel, a number of spares were on hand and could be used for the first refueling operation. The spares consisted of 2 assemblies enriched to 1.5 w/o U-235 and fourteen assemblies enriched to 2.0 w/o U-235.

Prior to specifying the enrichment of the charging fuel, exploratory calculations were performed to investigate the feasibility of using natural uranium in conjunction with a higher enriched equilibrium cycle feed enrichment of 2.0 w/o U-235. Upon pursuing this approach, it was determined that the use of natural uranium yielded undesirable power distributions. Subsequent calculations were then performed to investigate fuels enriched to 2.0 w/o, 3.0 w/o and 4.0 w/o U-235. During this time it was estimated that sufficient fuel would be required to operate the reactor from June 1, 1964 through November 30, 1967. The required burnup throughout the CVTR life for operation through November 30, 1967 would then be 24,500 EFPH. This estimate was based

upon full power operation of 65 MWt and an assumed load factor of 0.80. The accumulated burnup listed above includes that obtained from the initial core loading of 72 fuel assemblies.

The results of the study are presented in detail in Reference 4. The calculations were performed using the CANDLE code where the out-in cycling scheme was assumed and fresh fuel was placed in the outer core region and sequentially moved toward the central region in successive cycles. Each cycle consisted of replacing one-third of the 72 depleted fuel assemblies with fresh fuel. The results of the calculations are presented in Table 2.1, which illustrates the total number of fuel assemblies required to meet the desired burnup objective for the various enrichments investigated. It is emphasized that these calculations were performed with a one-dimensional cylindrical model and did not include any provision for non-uniform axial burnup or degradation of the D₂O moderator and coolant systems. The primary purpose of the calculations was to determine trends.

TABLE 2.1

TABULATION OF ASSEMBLY BURNUPS FOR VARIOUS FEED ENRICHMENTS

<u>Feed Enrichment</u> <u>(w/o U-235)</u>	<u>Full Power Hours</u> <u>(Required 24,500)</u>	<u>Total Number of Assemblies</u> <u>Required to Obtain Listed Lifetime</u>	
		<u>EFPH/Assembly</u>	
2.0 w/o	23,600	144	} 164
2.0 w/o	28,400	168	
3.0 w/o	24,400	120	203
4.0 w/o	19,200	96	} 224
4.0 w/o	24,000	107	
4.0 w/o	30,000	120	

The maximum gross radial factors calculated from the CANDLE code were 1.28 for 2.0 w/o, 1.78 for 3.0 w/o and 2.28 for 4.0 w/o fuel. A

comparable value of 1.13 was obtained for the first core. Thus, although the absolute values given for these calculations did not agree with results obtained from two-dimensional calculations, the trend was considered to be a good estimate. As shown in Table 2.1, the equivalent full power hours of operation increase with enrichment. It was desirable to incorporate the spare fuel presently on hand, consisting mainly of 2.0 w/o enriched fuel, into the first cycle. In addition, upon consideration of the fabrication costs, cycle lifetime, and power sharing for the enrichments investigated, it was decided that the best feed enrichment for use during refueling was 2.0 w/o. The reasons for choosing this enrichment were as follows:

- (a) Spares presently on hand consisted of 14 assemblies enriched to 2.0 w/o U-235 and 2 assemblies enriched to 1.5 w/o U-235.
- (b) Favorable radial power distributions.
- (c) Calculational model used for Core 1 could be applied directly to subsequent refueling operations.

2.1.3 Assembly Burnup Optimization

A series of CANDIE calculations were performed to investigate various out-in fuel cycling schemes. In the calculational model, the central fuel assemblies were removed first and the remaining assemblies were moved sequentially, one region toward the core center and the fresh fuel placed on the core periphery. In all cases examined, the specified enrichment for use during fuel cycling consisting of 2.0 w/o, was used. A total of five different cycling schemes were investigated consisting of four, eight, twelve, sixteen, and twenty-four assemblies per cycle. All calculations were initiated from the end of life condition of the first core. The purpose of the calculations was to determine an optimum fuel usage in terms of the average additional equivalent full power hours (EFPH) produced for each new assembly added to subsequent cores. In addition, the perturbation on the core radial hot channel factor was investigated on a preliminary basis.

Prior to performing the fuel cycling studies, CANDIE calculations were performed for the initial batch loaded core to establish a basis for all additional work. The results of the base case gave an accumulated lifetime of 10,200 EFPH as given on Figure 2.2. Upon completion of the initial core burndown, a series of branch calculations were performed where 4, 8, 12, 16 and 24 fuel assemblies were removed from the central depleted 1.5 w/o region and the remainder of the depleted fuel was relocated toward the core center. New fuel was added on the core periphery and subsequent additional calculations were performed until at least three cycles were accumulated. This constituted the out-in cycling scheme which as described above.

The results of the various cycling schemes are given on Figure 2.3 which demonstrated that the average EFPH per assembly was inversely proportional to the total number of assemblies added per cycle. The corresponding variation in the radial hot channel factor is depicted on Figure 2.4. It is emphasized that the CANDIE calculations, as presented, serve only as a relative evaluation of the expected change in the power distribution. As demonstrated in the figures, the advantage of additional burnup with the lesser number of assemblies is accomplished at the expense of an increase in the radial peaking factor. Based on the CANDIE cycling studies, it was decided to cycle eight fresh fuel assemblies per cycle in order to increase the total power production per assembly, thus reducing the number of fresh fuel assemblies required to achieve the total accumulated burnup of 24,500 EFPH desired. More precise calculations were then performed using the TURBO* computer code which utilized explicit representation of each fuel assembly in a two-dimensional x-y representation. Unlike the CANDIE calculations which used a homogenized radial fuel region, TURBO* calculations represented each fuel assembly and associated regions explicitly and thus provided a good measure of the relative radial power distribution.

Additional CANDIE calculations were performed to determine the additional power production which would be available by shuffling the

Figure 2.2 Calculated Lifetime for the Batch Loaded First Core From CANDLE

Figure 2.3 Plot of Average (EFPH/Assembly Added) Versus Number of Assemblies Added Per Cycle

Figure 2.4 Calculated Maximum Radial Hot Channel Factor Versus Fresh Assemblies Added

existent depleted fuel assemblies in the core instead of adding fresh fuel. Two specific calculations were performed where after 48 new assemblies were cycled into the core, the existent fuel was then reshuffled instead of adding fresh fuel. These calculations were performed for the eight and twenty-four assembly per cycle refueling schemes. The procedure used was to interchange the peripheral 24 assemblies with the central 24 assemblies. An additional core life of approximately 2400 EFPH was gained in both cases. A substantial increase in the radial peaking factor resulted, however, increasing to 1.93 as compared to the original value of 1.32 for the normal 24 assemblies per cycle scheme. It is anticipated that excessive hot channel factors may be avoided by the proper selection of core locations for the relocated fuel or by using fewer assembly moves. Thus, although the reshuffling scheme was not used in CVTR, it shows merit for future investigation.

The importance of the results obtained by shuffling depleted fuel is based on the substantial increase in power which can be achieved by relocating depleted fuel. More detailed calculations in the form of TURBO* would be required in order to determine the particular fuel moves required and power sharing obtained. It was anticipated that a substantial gain in terms of EFPH could be shown with an acceptable power distribution making this scheme very attractive when the objective is to increase the lifetime with existent fuel and not reach an equilibrium fuel cycle.

2.1.4 Refueling Downtime

Prior to actual fuel cycling on the CVTR, it was postulated that the refueling operation might be performed on a week-end basis when the total number of fuel moves was small. The actual downtime experienced during those refueling operations which had taken place exceeded the week-end operation by a considerable amount. Based on the two refueling operations which were performed, it is estimated that the

downtime under ideal conditions would be of the order of five days. Thus, the original postulation of week-end refueling had to be re-evaluated and incorporated into future cycling calculations.

Originally, the refueling downtime was not considered to be a deciding factor in choosing the number of fuel assemblies to move per cycle. Therefore, the minimum number of eight assemblies per cycle was chosen in order to gain the additional burnup per assembly for the fewer number of assemblies added. Since experience with the refueling operation demonstrated that the downtime was significant, twelve fuel assemblies were chosen for the modified second cycle in order to gain the additional lifetime thus increasing the time between cycling operations and extending Cycle 2 operation through an undesirable refueling period.

Calculations for the third cycle were performed using eight fresh fuel assemblies. Due to known fuel leaks existent in the core, more consideration was given to maintaining core symmetry and replacing the leaking fuel than in the burnup expected for Cycle 3.

2.1.5 Special Test Assembly Objectives

The intent of the special high power density test assemblies was to evaluate present conservatism in nuclear fuel performance limitations. Four special test assemblies were fabricated which were designed to operate at peak power levels of 25.1 kw/ft for extended burnup periods reaching 40,000 MWD/MTU. Enrichments of 3.8 w/o and 4.7 w/o fuel rods were specified for the test assembly fuel in order to insure that the higher power densities could be obtained during normal plant operation.

Due to the higher temperatures at which the test assembly fuel rods would be operating, it was necessary to limit their positions to the inlet sides of the pressure tubes. This restriction resulted in an asymmetry in the core since the inlet legs of the pressure tubes do not maintain quarter core symmetry. An objective of this test was

to operate as near the designed peak power densities for as long a period as possible. Thus, in addition to the usual high power limit which is required for safe operation of normal fuel, the test assemblies also had a lower limit of operation. The lower limit was chosen as a compromise between convenient plant operation and test objectives.

Since quarter core calculations were performed, it was necessary to design for operation of the test assembly which was limiting. The limiting assemblies were the 25.1 kw/ft peak assemblies which were alternately placed at core locations 3C and 8Q. The remaining two 20.9 kw/ft peak assemblies were placed at core positions 3S and 8A. At these locations the calculations indicated that when the assemblies in positions 3C and 8Q are operating at 25.1 kw/ft, the assemblies at positions 3S and 8A were operating at a level which was typically 10% less than their design limit of 20.9 kw/ft.

In addition to the stringent requirements placed upon the radial power distribution of the test assemblies, it was equally important that an accurate axial peaking factor be known for these assemblies since the peak level of operation was the primary objective of the test program. The expected axial peaking was estimated based upon previous values measured with flux wires during Core 1 operation. Control rod withdrawal programs were then chosen which would result in the least perturbation on the axial factor during control rod withdrawal. Normally, this required operating with control rods which were removed from the positions containing the test assemblies.

2.2 Fuel Management Design Model

The calculational model used throughout the design of the fuel management program consisted of the techniques which were previously developed as part of the Research and Development Program during Core 1 operation. These methods are presented in Reference 3 and proved to give an accurate representation when used for core follow calculations of the initial core. Since

a discrete tube three dimensional representation was not practical due to the large amount of computer time required, a two dimensional (x-y) representation was used. This model agreed very well with Core 1 operation and was, therefore, extended to the fuel management program. It was developed for use with the TURBO* computer program but may easily be adapted for use on any computer code which accepts a two dimensional representation as input.

The TURBO* calculations were performed to determine the values of the nuclear gross radial hot channel factor, gross power distribution, spatial depletion and cycle lifetimes. It was used to establish the required fuel moves and control rod withdrawal sequences to use during each fuel cycle. The TURBO* computations played a dominant role in the successful prediction of power levels of the special high power density test assemblies. Without the accurate time-dependent power distributions from the TURBO* calculations, a more conservative approach would have been necessary for the operation of the test fuel which would have defeated the test program. Therefore, it is worthwhile to present the model used in its entirety due to the important role that this model played throughout power operation of the CVTR.

2.2.1 TURBO* Geometry

The TURBO* calculational model was set up to simulate the actual CVTR core using a two-dimensional (x-y) representation similar to that used in the PDQ calculations for the initial core design.¹ A departure in the post construction work consisted of representing the fuel, pressure tubes, and moderator as distinct separate regions. Figure 2.5 illustrates the geometric representation used throughout all TURBO* studies. In the figure the circled numbers represent the regional breakdown of the reflector, moderator, pressure tube, control rods, fuel, clad, and coolant for neutron cross section input. The specifics of the models used to obtain the neutron cross section input are described in Section 2.3. Only the fast group cross sections were generated using two different models, slab and cylindrical. Table 2.2 describes the TURBO* compositions and the geometries defined in the

CVTR-72 TUBE TURBO *

Figure 2.5 Geometry Used in TURBO* Calculations

TABLE 2.2

DESCRIPTION OF TURBO* COMPOSITIONS

<u>Composition</u>	<u>Description</u>
1	Black Region
2	Outer Reflector 1 (Slab Geometry)
3	Outer Reflector 1 (Cylindrical Geometry)
4	Outer Reflector 2 (Slab Geometry)
5	Outer Reflector 2 (Cylindrical Geometry)
6	Reflector 3 (Slab Geometry)
7	Reflector 3 (Cylindrical Geometry)
8	Reflector 4 (Slab Geometry)
9	Reflector 4 (Cylindrical Geometry)
10	Reflector 5 (Slab Geometry)
11	Reflector 5 (Cylindrical Geometry)
12	Outer Moderator (Slab, 2.0 w/o)
13	Outer Moderator (Cylindrical, 2.0 w/o)
14	Inner Moderator (Slab, 2.0 w/o)
15	Inner Moderator (Cylindrical, 2.0 w/o)
16	Standard Moderator (2.0 w/o)
17	Standard Moderator (1.5 w/o)
18	Pressure Tube Near Reflector (Slab, 2.0 w/o)
19	Pressure Tube Near Reflector (Cylindrical, 2.0 w/o)
20	Standard Pressure Tube (2.0 w/o Cell)
21	Standard Pressure Tube (1.5 w/o Cell)
22	Grey Control Rod
23	Black Control Rod
24	Fuel-Clad-Coolant with 50% Reflector (2.0 w/o)
25	Fuel-Clad-Coolant with 75% Reflector (2.0 w/o)
26	Fuel-Clad-Coolant with 25% Reflector (2.0 w/o)
27	Standard Fuel-Clad-Coolant (2.0 w/o)
28	Standard Fuel-Clad-Coolant (1.5 w/o)
29	25.1 kw/ft Special High Power Density Test Assembly

parentheses describe the model used for generation of the fast group cross sections. Note that only the unit cells representing the peripheral regions had differences in the associated reflector. Also, the reflector region was subdivided into five different regions so that spectrum changes could be adequately represented with one set of fast group cross sections.

2.2.2 TURBO* Simulation of Continuous Control Rod Withdrawal

Axial control rod movement could not be performed in the TURBO* calculations due to the x-y representation used. Therefore, in order to approximate the effect of continuous control rod motion during plant operation, the control rod configurations were represented with the controlling rod group either fully inserted or fully withdrawn. Since discrete fuel assembly depletion is strongly affected by control rod motion, a scheme was devised which would approximately account for axial control rod motion. The calculations were performed such that the controlling rod group was fully inserted for approximately one-half of the lifetime corresponding to the worth of the particular controlling rod group. The control group was then withdrawn and the calculation was continued for the remaining reactivity life of the particular controlling rod group in question. In this manner the overall depletion was assumed to represent actual core performance but the calculations performed in the middle of the rod motion did not. They were performed primarily to simulate the depletion of the fuel on a discrete fuel assembly basis. This method was previously applied to the core follow calculations for Core 1 and proved to be very successful in predicting core performance. It had the same success throughout the fuel management program.

A reactivity history of the depletion calculations performed by TURBO* was maintained throughout the TURBO* studies such that actual plant operation would be approximated as closely as possible. A plot of the calculated eigenvalues from each problem performed was maintained to assist in predicting the amount of depletion associated with

the withdrawal of each control rod group. This reactivity history is shown on Figure 2.15 and is presented to illustrate the specific details of the study. An approximate 3% bias existed in the calculated reactivity obtained from TURBO* calculations as opposed to the actual plant performance. Part of this discrepancy was attributed to not accounting for the (n, 2n) reaction in the deuterium (1.7%), and partly to the zirconium cross sections used. Therefore, an eigenvalue of 0.97 from the TURBO* computations was interpreted as the just critical condition in the CVTR.

Due to the nature of the TURBO* model used, the calculations gave the radial power distributions and the gross nuclear radial hot channel factor with the control rod group of interest either fully inserted or fully withdrawn. Therefore, the radial hot channel factor and the radial power sharing expected during rod motion was determined by weighting the calculated values for the rods in and rods out configurations, using factors obtained from calculated rod worth and power sharing curves. The relationship given in equation 2.1 was used for these estimates.

$$F_R^i = F_{R_{in}}^i (f_1) + F_{R_{out}}^i (f_2) \quad (2.1)$$

The quantities used in equation 2.1 are defined below.

- F_R^i = Predicted radial power sharing of U-tube leg (i) for the particular banked position on the controlling rod group.
- $F_{R_{in}}^i$ = Radial power sharing of U-tube leg (i) calculated by TURBO* with the control group fully inserted.
- $F_{R_{out}}^i$ = Radial power sharing of U-tube leg (i) calculated by TURBO* with the control group fully withdrawn.
- f_1 = Fraction of power produced in the core region below the tip of the banked control rod group.
- f_2 = Fraction of power produced in the core region above the tip of the banked control rod group.

The weighting factors (f_1 and f_2) were obtained by first determining the height of the controlling rod group for various percentages of the control rod worths using Figure 2.6 which was originally presented in the CVTR nuclear design report.^{/1} The fraction of fission power generated below the controlling rod group was then determined for each control rod position using Figure 2.7 to obtain f_1 . The weighting factor f_2 then is the difference between 1.0 and f_1 . The resultant nuclear gross radial hot channel factor and radial power sharing for each U-tube leg was then determined for continuous motion of the controlling rod group. This scheme was used to predict the gross radial hot channel factor as a function of both control rod motion and core life. This procedure was also used to determine the calculated core radial power sharing at the prevailing control rod positions at which flux wire data were obtained, thus providing a direct comparison between the calculated and measured radial power sharing. In addition, it was very successful in predicting expected power levels of the special high power density test assemblies throughout operation of Cycle 1 and Cycle 2.

2.3 Computer Codes

Several computer programs were used in conjunction with the physics effort of the CVTR fuel management program. These codes consisted of the NDC^{/8}, THERMOS^{/9}, PLMG^{/10}, CANDLE^{/5}, TURBO^{/2}, and CAVIAR^{/7} computer programs. The NDC and CAVIAR codes were written specifically for CVTR during the original design and were also used during fuel management. The remainder of the programs listed above are standard design tools used for pressurized water reactor design at Westinghouse Electric Corporation. Basically, the above computer programs fall into one of three categories consisting of the following:

- (1) Computations performed to determine physics parameters required as input to subsequent codes which solve the diffusion equation for power and reactivity. In this category are the NDC, THERMOS and PLMG programs.

Figure 2.6 Percentage of Control Rod Worth as a Function of Withdrawal

Figure 2.7 Fraction of Fission Power Generated Below Rod Group Versus Withdrawal

- (2) Computations which represent the geometric characteristics explicitly for a particular reactor core and solve the diffusion equation for power sharing, reactivity and burnup. In this category are the CANDLE and TURBO* programs.
- (3) Computations which reduce data. In this category is the CAVIAR program.

2.3.1 NDC

The NDC computer code was developed for the explicit purpose of performing basic nuclear calculations for the CVTR. These calculations consisted of generating number densities and physical properties of pressure tube lattices with clustered fuel rods by applying the unit cell method. Input to the code consists of the fuel assembly and the tube lattice geometry, material distributions and some physical data. The unit cell analysis is applied to a piecewise homogenous, cylindrical, equivalent model of the actual reactor structure.

Basically, the NDC code is a unit cell method program written specifically for clustered fuel assemblies which form pressure tube reactors. It was programmed in order to introduce both the internal thermal baffles and the external pressure tube. The program output includes the equivalent cylindrical geometry, the various number densities, the neutron flux ratios, the various cross sections, the criticality factors, the initial conversion ratio and the burnup results. The use of NDC calculations for fuel management was carried over from the basic calculational methods used previously for Core I.

2.3.2 THERMOS

The THERMOS computer program was used to generate thermal cross sections for use as input to the CANDLE and TURBO* computer programs. The code is written in Fortran for use on the IBM-704/709/7090 digital computers. It computes the scalar thermal neutron spectrum as a function of position in a lattice by solving the integral transport equation with isotropic scattering. Geometric models representing a one

dimensional slab or cylinder may be used. The purpose of the code was to accurately predict effective thermal cross sections.

A cylindrical model was used for application to the CVTR having an area equal to that of a 6.5" x 8.0" cell. In order to account for differences in reflector thickness for those pressure tubes adjacent to the reflector, additional THERMOS calculations were performed which included an 18" annulus of D₂O reflector surrounding the basic model. All nuclear thermal constants for the peripheral pressure tubes were then based in part on both the basic model and that which contained an extra 18" D₂O annulus. Each peripheral fuel region was then established by interpolation between the two cases to account for the amount of reflector seen by that fuel region (i.e., 25%, 50%, 75%).

The mixed number density scheme (MND)¹¹ was applied to the thermal constants for CVTR applications. In addition, Maxwellian diffusion constants were used instead of the diffusion constants generated by the THERMOS code. The MND scheme was used based upon the good agreement achieved with this scheme during analysis of the initial critical experiments.

2.3.3 PLMG

The PLMG is a multigroup computer code which was written to determine the accuracy of the few-group diffusion scheme with various imposed conditions and also to provide an alternate computational method when this scheme fails to be sufficiently accurate. The program solves for the spatially dependent multigroup flux; it accounts for the slowing down and removal of neutrons resulting from epithermal capture and fission resonances. It also includes the regeneration of fast neutrons resulting from fissioning. As many as 80 fast multigroups may be represented. The code will accept as input a one-dimensional physical description of the reactor using slab, cylindrical or spherical geometry, group dependent boundary conditions, transverse buckling and a prepared library of nuclear properties of all the isotopes in each composition.

Output from the program consists of multigroup fluxes, currents and isotropic slowing down densities as well as pointwise and regionwise few-group macroscopic cross sections.

The PLMG computer code was used to generate the fast group micro and macroscopic cross sections reduced to three fast groups. Two geometric representations were used to generate the fast group constants for the peripheral unit cells, slab and cylindrical geometry, depending upon the location and the amount of associated moderator in the reflector. The reflector was subdivided into five different regions so that spectrum changes would be adequately represented with fast few group cross sections generated by the PLMG computer code.

2.3.4 CANDLE

The CANDLE computer code carries out the solution of the multigroup, one-dimensional diffusion equation in combination with a pointwise depletion analysis for the determination of reactivity lifetime relationships. Either slab, cylindrical or spherical geometries may be used. The code uses a maximum of four groups in solving the one-dimensional diffusion equation.

CANDLE calculations using a cylindrical core model were performed to establish fuel depletion rates for the CVTR fuel management program. They were also used for enrichment specifications and in the establishment of the required number of fuel assemblies to meet program objectives. Trends in the relative power sharing were investigated in order to provide a basis for x-y calculations which were later performed to determine power sharing and burnup information on an individual fuel assembly basis.

2.3.5 TURBO*

TURBO* solves the few group, 2 dimensional (x-y geometry) neutron diffusion equations in combination with a pointwise burnup calculation for the determination of reactivity lifetime relationships. It

contains a calculation of the effect of power density on resonance capture in uranium 238. The code utilizes single line relaxation. The number of lethargy groups is restricted to 2 or 4 with the number of points of solution limited to 7500. Two group g-factors were incorporated into the version used throughout the CVTR fuel management program to permit simulation of a regionwise cross section library in the TURBO* code.

The TURBO* calculations were performed to determine the value of the nuclear gross radial hot channel factor, gross power distribution, spatial depletion and the depletion rate throughout the fuel management program. In addition, the calculations yielded values for U-tube leg power sharing, installed reactivities, and cycle lifetimes. The calculations also played an integral part in the evaluation of the operating power densities of the special high power density test assemblies.

2.3.6 CAVIAR

The CAVIAR computer program is a data reduction code written in the standard FORTRAN language for use on the IBM-7094 digital computer. The code was written explicitly to perform the laborious task of reducing flux wire activation data from CVTR. The code consists of three separate links which perform interrelated calculations on the flux wire data. Input consists of standard data acquisition sheets and punched paper tape which are converted into card form.

CAVIAR output consists of three dimensional flux distributions, power distributions, axial maximum to average ratios, local to average tube power densities and various other parameters useful in appraising core performance. These quantities are calculated based on data obtained from induced Mn activation of wires which are inserted and withdrawn from the active core region. The code permits repeated use of the incore instrumentation by processing data obtained from flux wires which were used to measure more than one thimble location on any given flux wire irradiation sequence.

2.4 Fuel Movements

Throughout the fuel management program on CVTR all fuel movements were chosen to be compatible with the current objectives as described previously in Section 2.1. The fuel movements used in the calculational procedure were slightly different from those which actually took place at the site. This was necessary due to limitations in the computational model which assumed quarter core symmetry. The difference between the calculated fuel moves and actual plant movements were investigated and found to be conservative based upon power sharing considerations.

2.4.1 Calculated Fuel Moves

Due to limitations of the TURBO* computer code which restricted the total number of allowable mesh points to be no greater than 7500 in the x-y model used, quarter core symmetry was assumed applicable throughout the fuel management program. Based on this assumption all fuel movements were performed on a discrete fuel assembly basis. Figures 2.8, 2.9, and 2.10 show the particular fuel assembly movements used in the calculational scheme for Cycles 1 through 3.

Cycle 1 as depicted on Figure 2.8 consisted of inserting eight fresh fuel assemblies into the core at positions 3A, 3C, 4J, 5J, and their symmetric locations.

Figure 2.9 illustrates the fuel movements chosen for Cycle 2 operation. In this case twelve fuel assemblies were inserted into the core in order to increase the cycle lifetime. Cycle 3 is shown on Figure 2.10 and demonstrates the calculational procedure chosen for operation of the third cycle. In this case the objectives were modified such that the assemblies suspected to have fuel leaks were removed from the core. It is emphasized that although the third cycle of operation on the CVTR did not materialize, the calculations were completed and a third cycle would have appeared as given on Figure 2.10.

TUBE AND THIMBLE REPRESENTATION OF CVTR QUARTER CORE

North ↑

Figure 2.8 Fuel Assembly Movements Used in TURBO* Calculations of the First Cycle

TUBE AND THIMBLE REPRESENTATION OF CVTR QUARTER CORE

Figure 2.9 Fuel Assembly Movements Used in TURBO* Calculations of the Modified Second Cycle

TUBE AND THIMBLE REPRESENTATION OF CVTR QUARTER CORE

NORTH
↑

Figure 2.10

Fuel Assembly Movements Used in TURBO* Calculations of the Third Cycle

2.4.2 Plant Fuel Moves

The fuel movements performed at the plant were slightly different from those represented in the calculational model. Plant limitations necessitated moving two fuel assemblies at a time, thus all plant fuel movements were performed on a pressure tube basis. The specific fuel moves performed for Cycles 1 and 2 are presented on Figures 2.11 and 2.12, respectively. The anticipated plant fuel movements for the third cycle are given on Figure 2.13. Discrepancies existed between the fuel moves used in the calculations as compared to the actual plant movements made. The calculational model assumed a core consisting of four identical quadrants as given previously on Figures 2.8 through 2.10, respectively. The plant moves were performed on a pressure tube basis and the two legs of the pressure tube did not have the same exposure from previous power production. Also, since the 25.1 kw/ft test assembly was inserted into position 3C in the calculational model, all symmetric locations contained the same fuel. The actual plant refueling consisted of inserting fuel rated at peak power densities 25.1 kw/ft in locations 3C and 8R and 20.9 kw/ft fuel in positions 8A and 3S. These discrepancies were investigated and their effect upon the hot tube was determined to be small. The overall effect resulted in the calculational model generating a power distribution which was conservative based upon the core hot tube power operation levels.

2.5 Nuclear Design of Cycle 1

The nuclear design of Cycle 1 was performed primarily with the TURBO* computer code which solves the diffusion equation for reactivity and power sharing as a function of core life. Upon judicious selection of the particular fuel movements desired, the calculations were then carried out to establish feasibility in terms of core life and acceptable power sharing. Initial fuel moves were based primarily on direct observation of the fissionable nuclide number densities for the existent fuel assemblies prior to fuel cycling. Using the objectives of the fuel management program

Figure 2.11 Pressure Tube Moves During Cycle 1 Refueling

Figure 2.12 Pressure Tube Moves During Cycle 2 Refueling

/1 - Special Test Assy.
20.9 kw/ft Removed	— - ROD OUT	— - BLACK ROD BANKED AT _____
/2 - Leaking Standard
2.0 w/o Fuel Removed	— - BLACK ROD IN	— - GRAY ROD BANKED AT _____
/3 - Contains Depleted
1.5 w/o Fuel Assy.
from Core 1	— - GRAY ROD IN	• - FLUX MONITORING WIRE

Figure 2.13 Proposed Pressure Tube Moves During Cycle 3 Refueling

previously described, all fuel movements were selected on the basis of meeting these objectives. The sequence of events was as follows:

- (1) Select the appropriate value of the axial peaking factor based on the most recent values established from measurements of the existent core.
- (2) Establish the required radial peaking factor for the special test assemblies and select core positions which meet these requirements based on anticipated power sharing results established from the current number densities.
- (3) Perform calculations with the TURBO* computer code with the selected core positions for refueling.
- (4) Compare TURBO* results with design requirements and, if satisfactory, continue the TURBO* calculations to establish the expected cycle life-time and control rod withdrawal program. If the original fuel moves are unsatisfactory, make appropriate adjustments to establish an acceptable pattern based upon the results from TURBO*.

The above approach, although crude, was used successfully due to the large amount of information from past experience of typical plant operation. It is recognized that no equilibrium cycling scheme was achieved but the particular objectives of the CVTR fuel management program were obtained with a minimum amount of effort.

Prior to performing the fuel cycling calculations, it was necessary to establish the effect upon core symmetry due to inserting the high power density test assemblies at different core symmetric positions. Since the test assemblies must be placed in the inlet side of the pressure tubes, quarter core symmetry could no longer be maintained because quarter core symmetry does not exist on CVTR for the inlet legs of the symmetric pressure tubes. Therefore, prior to the actual cycling calculations, two calculations were performed to determine the effect on the power distribution due to the difference of enrichments in the special test assemblies. The resultant power distributions are given on Figure 2.14 in conjunction with the end of Core I life distribution.

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NORTH

Eigenvalues

/1 - λ = 0.9884

□ - Rod Out

■ - Black Rod Bank Out

/2 - λ = 1.0254

■ - Black Rod In

■ - Gray Rod Bank Out

/3 - λ = 1.0242

■ - Gray Rod In

- /1 Radial power distribution from end of Core I life (before cycling).
- /2 Cycle 1 22 kw/ft assembly in 8A; 2.0 w/o fuel in 8C (after cycling)
- /3 Cycle 1 20 kw/ft assembly in 8A; 2.0 w/o fuel in 8C (after cycling)

Figure 2.14 Comparison of Local-to-Average Radial Power Distributions With and Without Test Assemblies Inserted

A few comments are appropriate to understand the calculational model used in the distributions on Figure 2.14. These calculations were performed using the end of Core I life history tape. Therefore, xenon equilibrium exists for all assemblies except the standard 2.0 w/o assembly and the test assembly which were inserted in the core after end of life of Core I. Observation of Figure 2.14 illustrates that the effect of the difference in the power distribution due to enrichment between test assemblies is a localized effect. That is, only location 8A has a substantial difference in power sharing in the two calculations performed after cycling. This difference is of the order of 6% due to enrichment alone. An additional effect due to differences in the relative positions of the two different test assemblies is of the order of 11%. The 3.8 w/o enriched assemblies contained six fuel rods whose lengths are typically 81.1 inches as opposed to the higher enriched assembly which contained three of the 4.7 w/o fuel rods which were 77.8 inches long resulting in the lower power test assemblies having a 2% increase in the overall power production. Therefore, the net difference in power production due to geometric effects alone is of the order of 15%. When the 25.1 kw/ft peak assemblies are placed in positions symmetric to 8C and the 20.9 kw/ft peak test assemblies are placed in location 8A, the higher power density test assemblies are limiting. Typically, based on conservative estimates, when the 25.1 kw/ft assembly is operating at its peak value, the 20.9 kw/ft test assembly would be approximately 3% less than its design limit. Based on these results, all quarter core calculations were performed using the higher power density test assembly which was the limiting case.

Due to computation time and computer code limitations, all subsequent calculations were performed with the 25.1 kw/ft peak test assembly located at position 8C and standard 2.0 w/o enriched fuel in position 8A.

The results of the design calculations for Cycle 1 resulted in an acceptable cycling scheme after the xenon equilibrium condition was reached. During the xenon transient period, excessive power sharing existed in both the core hot tube and the limiting special test assembly (25.1 kw/ft) requiring the plant power level to be derated until the xenon equilibrium

condition was reached. During this period, full power operation was not anticipated due to additional materials requirements resulting in power sharing measurements and subsequent incremental power increases in order to conform to the stress requirements of the clad in the high power density test assemblies. Thus, the cycling scheme used was compatible with prevailing planned operation.

2.5.1 Lifetime

The total life expected from TURBO* calculations of the first cycle was 1540 EFPH. This value was based upon the results obtained from a series of TURBO* calculations as presented on Figure 2.15. Typical plant operation was simulated in the TURBO* calculations using explicit representation in a two dimensional (x-y) model having quarter core symmetry. Continuous control rod withdrawal was accounted for by performing calculations with the control rod group either fully inserted or fully withdrawn. During the calculations, a detailed history of the calculated reactivity at specific burnup steps was maintained as illustrated on Figure 2.15. All subsequent time steps were then chosen based upon previous performance in order to simulate actual core performance as closely as possible with the existent model.

The control rod withdrawal program given in Figure 2.15 was to withdraw control rod Group IV, then Group V. It was anticipated that xenon equilibrium would occur (i.e., 100 EFPH) when control rod Group IV was banked at 64 inches above the bottom of the fuel and control rod Group V fully inserted. At 305 EFPH control rod Group IV would be fully withdrawn and control Group V fully inserted. Actual core performance later verified the above results.

2.5.2 Power Distributions

A total of six calculations were performed to determine the relative power sharing expected and the life of the first cycle. The power distributions obtained from these calculations are given in

Figure 2.15 Plot of Eigenvalue Versus Lifetime From TURBO* Calculations of the First Cycle on CVTR

Appendix I. Since only two dimensions were used in the calculational model, partial withdrawal of the controlling rod group could not be performed explicitly. Thus, only those calculations performed at the burnup (EFPH) expected to have the controlling rod group either fully withdrawn or fully inserted represent the actual plant operating condition. The intermediate calculations do not represent actual core operation but were performed in order to simulate continuous control rod withdrawal in the burnup calculations as described above.

Observation of the power distributions given in Appendix I show that during the xenon transient the radial peaking factor in the hot tube (6I) exceeds the design limit of 1.525. Also, the power sharing in the test assembly (8C) is considered to be excessive during this period. In order to show the expected variation of the radial power sharing in both the hot tube and the test assembly as a function of lifetime and rod position, hand calculations were performed for various positions of the banked controlling rod group. The results of these calculations are given in Figures 2.16 and 2.17. Observation of Figures 2.16 and 2.17 illustrates that during the xenon transient period excessive power sharing may be experienced in both the hot tube and the test assembly. Therefore, during this period the plant was derated until an acceptable condition was reached (i.e., 100 EFPH). The plant was derated to approximately 80% of full power based on the hot tube. Allowance was made for a 10% uncertainty in the calculations to arrive at the 80% level.

In addition to the radial peaking factor, the axial peaking factor was very important in analyzing the power density expected for the test assembly (8C). Figure 2.18 shows the maximum allowed value of the axial peaking factor for the test assembly throughout the first cycle. Flux wire measurements at that time indicated a typical value of F_z in 8C of the order of 1.25. Since this value was expected to increase slightly due to the insertion of the new fuel, it was reasonable to assume that the expected value of F_z during the first cycle will be

Figure 2.18

Variation of F_Z^N max in 3C (25.1 kw/ft Test Assembly)
as a Function of Burnup for the First Cycle

of the order of 1.3. Thus, based on Figure 2.18 the power density in the test assembly was calculated to exceed its design value of 25.1 kw/ft (4.7 w/o enriched fuel rods) if the plant was operated at full power during the xenon transient period. Therefore, both the hot tube and the test assembly would exceed the design limits during this period. The usual 10% allowance for calculational uncertainty was not incorporated in these calculations of the expected power density in the test assembly.

2.5.3 Control Rod Withdrawal Program

In addition to those calculations performed to determine the control rod withdrawal program during the first cycle, one calculation was performed to investigate the control rod withdrawal sequence to use during rise to power. Control rod groups II and VII were inserted in addition to groups IV and V. The resultant power distribution is shown on Figure 2.19. Based on the results of this calculation, the following control rod withdrawal program was recommended and used for operation throughout the first cycle.

- (a) Withdraw control rod group I
- (b) Withdraw control rod group III
- (c) Withdraw control rod groups II and VII simultaneously until the critical condition is reached. Continue to withdraw II and VII in increments of four inches on each control rod group in order to maintain no more than a two-inch height variation between the two groups during withdrawal.
- (d) Follow the xenon transient until control rod groups II and VII are fully withdrawn using the procedure given above in (c).
- (e) After withdrawal of groups II and VII, continue to follow the xenon transient by withdrawal of control rod group IV. At approximately 64 inches on group IV, the reactor was predicted to be at equilibrium xenon for the particular power level

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CVTR Fuel Management Calculations
Cycle-1

NORTH
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Figure 2.19 Local-to-Average Power Distribution From TURBO* Calculation Before Xenon Equilibrium in the Fresh Fuel

specified. At this time an increase in power level beyond 80% may be performed.

- (f) After control rod group IV is fully withdrawn, initiate withdrawal of control rod group V.
- (g) Continue to withdraw control rod group V until the end of life condition is reached for the first cycle.

This control rod withdrawal program was intended as an outline and required modification due to normal plant limitations during power operation. For instance, normal plant procedure was to initiate withdrawal of the succeeding control rod group when the controlling rod group presently in use had reached the position such that the minimum insertion shutdown rate could not be maintained (i.e., usually in the top third of the core).

2.6 Nuclear Design of Cycle 2

The calculations performed for Cycle 2 were based upon both an eight and a twelve assembly replacement scheme. The original eight assembly cycle was modified due to a desire to obtain additional life during Cycle 2 operation. The design procedures previously established for Cycle 1 were extended to Cycle 2 and subsequent fuel cycling calculations. In the original study with an eight fuel assembly replacement, a total life of 1710 EFPH was expected. The modified version consisting of a twelve fuel assembly replacement extended the eight assembly lifetime an additional 720 EFPH resulting in a total life for the modified second cycle of 2430 EFPH.

The modified version of the second cycle was actually used during refueling. Both versions are presented, however, for completeness of this report. The primary difference between the eight and twelve assembly cycle schemes for the second cycle was that the twelve assembly scheme inserted fresh fuel in positions 3J and 8J in addition to the fuel moves planned for the eight assembly cycle scheme.

Figure 2.20 Eigenvalue Versus Lifetime From TURBO* Calculations of the Second Cycle

Results from the calculations performed indicated that an acceptable cycling scheme was achieved for both the eight and twelve assembly cycling schemes. The twelve assembly version was more desirable in that it had the advantage of additional lifetime in conjunction with higher power levels in the special test assemblies.

2.6.1 Lifetime

Both Cycle 2 refueling schemes were initiated from the end of life condition of the first cycle. The special high power density test assemblies were maintained in the positions previously established for Cycle 1. As in the previous calculational scheme, a detailed history of the calculated reactivity for specific lifetimes was maintained and is presented on Figure 2.20. The results given in the figure represent the two cycling schemes used. The point represented by a solid circle was performed to investigate the effect of withdrawing control rod group II then group V in the modified version of Cycle 2. This control rod withdrawal sequence was not pursued due to the relatively low power sharing in the test assembly. The total life expected for the modified version of the second cycle is the crossover point between the all rods out line and the just critical line resulting in a value of 2430 EFPH. In the figure the solid line connects actual calculations performed and the dotted lines were drawn parallel to the depletion established directly from calculations.

2.6.2 Power Distributions

A total of six TURBO* calculations were performed for the second cycle utilizing an eight assembly replacement scheme, and an additional six TURBO* computations were performed for the modified version of Cycle 2. The resultant power distributions are presented in Appendix I. In addition to the radial distributions given in the Appendix, more detailed hand calculations were performed to investigate the radial power sharing expected in the hot tube and the 25.1 kw/ft peak test

assembly. The results of the calculations are presented on Figure 2.21 and Figure 2.22. It is emphasized that a detailed calculation of the xenon transient period was not performed.

Since the axial peaking factor is very important in analyzing the power density expected for the test assembly, the maximum allowed value for the axial peaking factor for the 25.1 kw/ft test assembly was also determined based upon the calculated radial power sharing in that assembly for Cycle 2. The resultant values are given on Figure 2.23 and are less than the design limit of 1.5 which is applicable to the standard fuel. Current flux wire irradiation measurements obtained at that time from Core I, and from the test assembly locations, gave values of the order of 1.25 for F_z . Thus, it was anticipated that the design limit of 25.1 kw/ft peak would not be achieved during Cycle 2 operation. Actual operation resulted in a peak power density level of approximately 23 kw/ft in the high power density test assembly.

As in the previous calculations for Cycle 1, xenon equilibrium existed in all fuel assemblies with the exception of the fresh fuel inserted as part of Cycle 2. Therefore, the power distributions obtained do not represent the actual mode of operation during the xenon transient period. During rise to power in the plant, xenon was building up in all fuel assemblies, therefore, the power distributions actually looked more like the distributions obtained at xenon equilibrium for those power distributions presented in Appendix I. The results in the figures were considered to be a conservative estimate of the power sharing during this period. During rise to power, after refueling, a detailed history of the spatial power sharing was established from measurements with incore instrumentation throughout the xenon transient period and are presented in Section 4.0.

2.6.3 Control Rod Withdrawal Sequence

One calculation was performed for each version of Cycle 2 to investigate the control rod withdrawal sequence to use during rise

Figure 2.21 Variation of the Radial Peaking Factor as a Function of Burnup on the CVTR

Figure 2.22 Variation of F_R^N in 3C (25.1 kw/ft Test Assembly) as a Function of Burnup for the Second Cycle

Figure 2.23 Variation of the Maximum Allowed Value of the Axial Peaking Factor in the 25.1 kw/ft Test Assembly Versus Burnup

to power. The control rod withdrawal programs as outlined on the following pages assume that the reactor core was initially shutdown with all control rod groups fully inserted.

A. Withdrawal Program for the Second Cycle (8 Fresh Fuel Assemblies)

1) Rise to Power

- a) Withdraw control rod group I
- b) Withdraw control rod group III
- c) Withdraw control rod groups II and VII simultaneously in increments of four inches on each group until the critical condition is reached. The four-inch increments should be spaced such that no more than a two inch height variation exists between groups II and VII at any time.

2) Power Operation

- a) Continue to withdraw control rod groups II and VII in increments of four inches on each group. During this time xenon will be building up.
- b) After withdrawal of control rod groups II and VII continue to follow the xenon buildup by withdrawal of control rod group IV.
- c) Withdraw control rod group IV until xenon equilibrium is reached at the specified power level.
- d) After control rod group IV is fully withdrawn, initiate withdrawal of control rod group V until the end of life condition is reached.

B. Control Rod Withdrawal Program for the Modified Second Cycle (12 Fresh Fuel Assemblies)

1) Rise to Power

- a) Withdraw control rod group I.
- b) Withdraw control rod group III.

- c) Withdraw control rod groups IV and VII simultaneously in increments of four inches on each group until the critical condition is reached. The four-inch increments should be spaced such that no more than a two inch height variation exists between groups IV and VII at any time.

2) Power Operation

- a) Continue to withdraw control rod groups IV and VII in increments of four inches on each group. During this time xenon will be building up.
- b) After withdrawal of control rod groups IV and VII continue to follow the xenon buildup by withdrawal of control rod group V.
- c) Withdraw control rod group V until xenon equilibrium is reached at the specified power level.
- d) After control rod group V is fully withdrawn, initiate withdrawal of control rod group II until the end of life condition is reached.

As in the previous work, it was recognized that the specified control rod withdrawal program was intended only as an outline. Modifications were required due to normal plant limitations as outlined previously for Cycle 1.

2.7 Nuclear Design of Cycle 3

The third cycle of the fuel management program for CVTR was initiated on an analytical basis due to fuel failures which occurred during operation of the second cycle. Power operation during Cycle 2 was prematurely terminated on January 24, 1967 after an accumulated burnup on Cycle 2 of 1600 MWD/MTU in contrast with the original anticipated life of 2400 MWD/MTU. Power operation was terminated in order to remove the leaking fuel in assembly locations 3S, 6I and 7K. Replacement fuel was incorporated with the third cycle refueling scheme in order to minimize reactor downtime and fuel

handling. Based on the calculational results obtained, a total life of 1433 MWD/MTU was expected during Cycle 3 power operation, if continued operation would have occurred.

Two previous fuel failures occurred prior to Cycle 2 shutdown which were replaced with standard fuel having enrichments of 1.5 and 2.0 w/o in U-235. Specific details are given in Section 5.6. Thus, at the time of Cycle 2 shutdown, only two special high power density test assemblies remained in the core at locations 8A and 3S. Since the test assemblies in 3C and 8Q were previously replaced and 3S also developed a leak, only one undefective test assembly remained in the core. Therefore, the test fuel in location 8A was also scheduled for withdrawal during Cycle 3 since it was postulated to have a high probability for failure during additional power operation. Thus, Cycle 3 core would have contained standard fuel assemblies at all 72 locations. During design of the third cycle, future plant operation was questionable, therefore, the calculations were terminated when an acceptable scheme was achieved. Refinements such as the peak assembly power sharing as a function of both burnup and control rod withdrawal were not determined. These calculations would have been performed prior to Cycle 3 power operation.

2.7.1 Lifetime

The results as given in this section came directly from TURBO* calculations as did previous refueling design calculations. As in the results presented previously, only those power distributions presented with the control rods either fully inserted or fully withdrawn at the appropriate lifetimes represent actual plant operation. All other distributions were obtained in order to simulate continuous control rod withdrawal in the calculational scheme and did not represent the expected mode of plant operation.

A detailed history of the calculations performed is given on Figure 2.24 showing the calculated eigenvalues as a function of core lifetime. Specific control rod withdrawal is also depicted in the

Figure 2.24

Plot of Eigenvalue versus
Burnup From TURBO* Calculations of the Third Cycle.

figure. One calculation was performed to investigate the total excess reactivity without xenon in the fresh fuel at the beginning of Cycle 3 life.

2.7.2 Power Distributions

Cycle 3 design calculations were initiated at the time power operation was terminated on the second cycle representing 1600 MWD/MTU burnup on the Cycle 2 core. At this time, 8 fresh 2.0 w/o fuel assemblies were inserted into the calculational model constituting Cycle 3. The fresh fuel was inserted into pressure tube locations 3B, 3R, 8B and 8R as illustrated on Figure 2.13 in Section 2.4. Previously these locations contained the high power special test assemblies and replacement fuel of 1.5 and 2.0 w/o.

During early operation of Cycle 2, two special test assemblies developed leaks and were replaced with fresh fuel. The replacements consisted of 1.5 w/o enriched fuel in location 3B and 2.0 w/o enriched fuel in location 8R which were used to replace the 25.1 peak kw/ft test assemblies which developed leaks at 3C and 8Q, respectively. The replacement fuel would have remained in the core during Cycle 3 causing an asymmetry in the core. It was expected that this asymmetry would have no adverse effect on the overall core power distribution. Since Cycle 2 contained an asymmetry of the order of 3% with the south half high, the 1.5 w/o and 2.0 w/o replacement fuel was positioned to counterbalance the known existent asymmetry. Therefore, Cycle 3 fuel movements were designed to compensate for existent asymmetry by placing the 2.0 w/o fuel in the north half and the 1.5 w/o fuel in the south half of the core.

A total of six calculations were performed to determine the anticipated power sharing for discrete control rod configurations throughout Cycle 3 life. The resultant radial power distributions obtained are presented in Appendix I. As in the previous cycling studies, the initial power distributions for the third cycle did not contain xenon in

the fresh fuel cycled into the core. The power distributions presented for 100 EFPH represent the expected radial power sharing during xenon buildup. Throughout the major portion of Cycle 3 operation, assembly 8I and its symmetric locations would have operated as the hot tube. This fuel was previously inserted into the core during refueling for Cycle 2 and had an existent average burnup of approximately 1900 MWD/MTU.

2.7.3 Control Rod Withdrawal Sequence

One calculation was performed to investigate the control rod withdrawal sequence to use during rise to power. Control rod group II was inserted in addition to group V at the beginning of life of the third cycle. The resultant power distributions are presented on Table I.5. Based on the results of the calculations given in the table, the following control rod withdrawal program was recommended for use during Cycle 3. As in the previous case, this program assumes that the core is completely shutdown and all control rod groups are initially fully inserted (i.e., groups I, II, III, IV, V, VII in).

A. Cycle 3 Control Rod Withdrawal Sequence

1. Rise to Power

- a) Withdraw control rod group I
- b) Withdraw control rod group III
- c) Withdraw control rod group IV
- d) Withdraw control rod group VII at which time the reactor will become critical, the specified power level will be obtained, and xenon will begin to build up.

2. Power Operation

- a) Continue to withdraw control rod group VII; during this time xenon will be building up in the fuel and reactor power level should be limited to 70%.
- b) Withdraw control rod group II and continue to follow the xenon buildup. When xenon equilibrium is reached, obtain

measurements of the relative radial power sharing and increase the power level to 100%.

- c) When control rod group II is fully withdrawn to the permissible limit, initiate withdrawal of control rod group V and continue until the end of life conditions is reached.

Note that no calculations were performed with control rod groups VII, II and V fully inserted. It was expected that upon reaching the 70% level, group VII would be nearly fully withdrawn and most of the xenon buildup period would have been controlled with group II.

3.0 FUEL MANAGEMENT PHYSICS TESTS AND MEASUREMENT RESULTS

The Research and Development Program associated with the development of the CVTR core provided a detailed experimental program in order to measure various physics quantities of interest. Within the framework of this experimental program were the measurements of the temperature coefficients, differential and integral control rod worths and core depletion.

The measurement techniques used during the fuel management program differed slightly from those used previously during Core I. All modifications were made to either improve measurement accuracy or reduce the time required to perform the measurement. The measurement techniques used and the methods of reducing the data are presented in detail in this section along with the reduced results.

3.1 Review of Core I Physics Program

The objective of the Core I physics program was to measure the nuclear characteristics of the core as a function of plant life, and to compare the measured results to the design predictions. Experiments were conducted with the purpose of evaluating burnup and obtaining nuclear data. It was necessary to develop techniques to be employed while executing these experiments. Some of the techniques developed during Core I were later utilized during the fuel management physics tests. Others were modified slightly for use during these tests.

The experimental results obtained during the operation of Core I indicated that the analytical methods and models applied to the investigation of the CVTR core were quite adequate. Predicted power distributions, as a function of both core depletion and control rod withdrawal, were in good agreement with the measured distributions. The predicted total core lifetime was also in good agreement with the observed value. Fair agreement was obtained between calculated and measured control rod worths.

Measurements of reactivity coefficients fell generally within the predicted range of values. The reactivity coefficients which were measured

consisted of the moderator temperature coefficient, the primary temperature coefficient, and the power coefficient. The classical equations for xenon number density as a function of time and flux level (a point model) were found to reproduce the shape of measured xenon transients. Measured and predicted values of the nuclear heating of the moderator were in good agreement.

Decay heat measurements were also conducted and several approaches to the determination of the depletion rate were investigated. The effect of xenon on the differential control rod group worth was investigated and the effect of core depletion on the solution of the inhour equation was considered.

The various instrumentation systems on which the experimental program relied (flux wire system, datalogger, thermocouples, etc.) were found to perform consistently following an initial period of problem area location and solution. The performance of this equipment demonstrated that reliable data could be obtained from the reactor core. These data were valuable in monitoring the performance of the reactor throughout the fuel management program.

A complete description of the Core I physics tests and results is presented in Reference 3.

3.2 Physics Measurement Program and Objectives During Fuel Management

The fuel management physics tests included the periodic measurement of temperature coefficients, differential moderator worth, differential control rod worths, and integral control rod worths. These measurements were conducted at the beginning of operation of each fuel cycle. The differential moderator worth and integral and differential control rod worth measurements were conducted in basically the same manner as that prescribed during Core I operation. The technique employed while measuring the temperature coefficients was significantly different and is described in Section 3.5 of this report. This technique was developed during Core I operation and compared favorably to the conventional technique. Although the method of measurement of the integral and differential control rod worth was the same

as that used during Core I, the method of data analysis differed to some extent. These differences are noted in Section 3.4.

Flux wire measurements were conducted during the fuel management program to compare measured and predicted power distributions. In addition, calculations were performed to determine the fuel cycling depletion rates. The shape of the integral control rod worth curve at equilibrium xenon was investigated. Several gamma scan measurements were performed (See Section 6.1.3) to investigate reflector peaking of the special test assemblies, if existent. Decay heat measurements in the instrumented U-tube were performed to determine the fractional power split between the high power density test assembly and the standard fuel contained within that pressure tube. A correlation between the flux at the U-Bend thermocouple and the power sharing indicated by the thermocouple was derived (See Section 6.1.2).

The objective of the fuel management physics measurement program was to provide a series of tests which would establish the necessary data on core characteristics with a minimum time spent in the shutdown condition. In order to accomplish this objective, the method of measuring temperature coefficients was modified to perform the measurement during the approach to power and initial power operation. The control rod worth measurements were also altered to require less time in the shutdown condition.

A secondary objective in the fuel management physics program was to operate the special high power density test assemblies at or near their rated power densities. A program was then established to monitor these assemblies which included flux wire irradiation and thermal-hydraulic measurements.

3.3 Test Procedures

The physics test and experiments were performed according to procedures containing sufficient detail such that the experiments could be carried out on a step by step basis directly from the procedures. Data sheets were also prepared along with the procedures to insure that all information required would be obtained at the proper time. Copies of the procedures were distributed

for review to CVNPA, Westinghouse, AEC personnel, and the CVNPA Safety Committee. This permitted multiple judgments on each procedure from the standpoint of safety and operations and helped to insure that the recommended procedure would result in the desired measurement. Advantages of using procedures in this manner are that the particular experiment and its analysis must be well thought out in advance and the procedures result in well organized experiments. Use of the procedures also allowed the operations personnel to make valuable contributions to the conduct of experiments in the areas of plant and operator capabilities.

3.4 Experimental Techniques

The experimental techniques employed during the fuel management physics tests were, in some cases, identical to those employed during Core I physics tests. The basic method employed during Core I tests was to determine core reactivity as a function of the particular parameter of interest. This was accomplished by measuring the asymptotic period or startup rate (SUR) which was then related to reactivity through the inhour equation. The relatively rapid control rod speeds in the CVTR (25 inches/minute withdrawal and 100 inches/minute insertion for experiments) contributed to the feasibility of these measurements. In practice, the period measurements were accomplished in the following manner. The control rod configuration was such that with a specified control rod group completely withdrawn, the reactor would be on a positive period of an acceptable magnitude. Then the control rod group was completely withdrawn, and the asymptotic periods were measured. The reactor was shutdown by inserting the previously withdrawn control rod group, and the parameter of interest was varied in anticipation of the next startup measurement.

Data obtained in this manner were usually fit to a straight line by the method of least squares. Typically, the quantity of interest (e.g., temperature coefficient, differential moderator worth, etc.) was equal to the slope of the straight line. Reference 3 provides a complete description of the experimental techniques which were employed during the Core I physics tests.

In order to shorten the shutdown time required to complete physics testing and thereby improve reactor availability for power production, two alternate experimental techniques were developed for use during the fuel management physics experimental program.

The first of these was an alternate approach to the measurement of the moderator and primary temperature coefficients. This technique was employed during the heatup and initial operation of the reactor following a shutdown. A reactivity balance relating the several contributors to core reactivity (e.g., moderator and primary systems temperature, control rod withdrawal, xenon buildup, power level, and core depletion) was maintained such that a temperature coefficient could be inferred, i.e.,

$$\Delta\rho_{\text{Temperature}} = \sum_i \Delta\rho_i \quad (3.1)$$

where the summation over the index (i) includes all the mechanisms contributing to core reactivity except that on the left hand side of the equation. A basic feature of this technique is the ability to calculate the reactivity effect of the xenon transient based on previous xenon transient measurements.

The second alternate technique employed was a method to determine the integral control rod group worth through a technique of "inverse normalization". From repeated measurements of the differential control rod worth curve, it was apparent that the normalized curve (to a 1% $\Delta\rho$ group worth) was relatively insensitive to group worth and core depletion. Therefore, a normalized differential control rod worth curve was assumed as valid. Several $(\Delta\rho/\Delta h_{\text{rod}})$ points were measured at intervals along the control rod group of interest and ratioed to the 1% $\Delta\rho$ normalized values, at the same rod height, to determine the group worth.

These methods of measuring the integral control rod group worths and the moderator and primary system temperature coefficients are considered less precise than the techniques previously employed; however, they do provide a satisfactory measure of these coefficients without adding significantly to the shutdown time.

3.5 Test Results

The fuel management physics experiments conducted at the CVTR included the periodic measurement of temperature coefficients, differential moderator worth, integral and differential control rod worths, and reactivity depletion rates. All of the measurements undertaken under this program were reported in the appropriate Research and Development Program Quarterly Progress Report. Where convenient, experimental results are reported in terms of MWD/MTU in this report. However, most of the data and analysis were compiled in terms of effective full power hours (EFPH) based on a full power level of 65 MW, and portions of the results are reported in this manner. Fortunately, it is of little consequence whether EFPH or MWD/MTU is used because there is nearly a one to one conversion: (1 EFPH = 0.988 MWD/MTU).

3.5.1 Temperature Coefficients

Measurements of the moderator temperature coefficient and the primary temperature coefficients were made at the beginning of Cycle 1 life and again at the beginning of Cycle 2 life. The moderator temperature coefficient measurement at the beginning of Cycle 1 life was conducted twice and employed each of the two techniques described in this section under Experimental Techniques. The measurement at the beginning of Cycle 2 life employed only the reactivity balance technique. The measurement employing the reactivity balance technique was accomplished during power operation prior to establishing equilibrium xenon conditions. With the reactor power and primary coolant temperature stabilized and the control rod positions fixed, the average moderator temperature was lowered gradually to compensate for the build-in of xenon and reactivity depletion through core burndown. A reactivity balance was established to determine the reactivity defect due to changes in moderator temperature.

$$\Delta\rho_{\text{Moderator Temperature}} = \Delta\rho_{\text{Xenon}} + \Delta\rho_{\text{Burndown}} + \Delta\rho_{\text{Primary Temperature}} \quad (3.2)$$

The moderator temperature reactivity defect was then plotted versus the applicable moderator temperature. An example of this is shown in Figure 3.1. The slope of the straight line fit through these data is the moderator temperature coefficient. The moderator temperature coefficients measured during the fuel management physics tests are tabulated in Table 3.1.

The measurement of the primary coolant temperature coefficient was performed at the beginning of Cycle 1 life and again at the beginning of Cycle 2 life by the reactivity balance method. The measurement was made during the primary coolant heatup and approach to power. The reactor power was stabilized at approximately 5% power and the moderator temperature was maintained as constant as possible. Control rods were moved as necessary to compensate for xenon and other reactivity defects. A reactivity balance was established to determine the reactivity defect due to changes in the primary temperature.

$$\Delta\rho_{\text{Primary Temperature}} = \Delta\rho_{\text{Control Rods}} + \Delta\rho_{\text{Xenon}} + \Delta\rho_{\text{Power Defect}} + \Delta\rho_{\text{Moderator Temperature}} \quad (3.3)$$

The primary coolant temperature reactivity defect was then plotted versus the applicable primary temperature, similar to Figure 3.1. The slope of a straight line through these data is the primary coolant temperature coefficient. The coefficients measured as part of the fuel management physics tests are tabulated in Table 3.1.

3.5.2 Differential Moderator Worth

The differential moderator worth was determined at an average moderator height by making a series of successive asymptotic period measurements at slightly different moderator heights, while other core parameters remained unchanged. During these period measurements, individual control rod groups were either completely withdrawn or fully inserted to compensate for moderator height changes. The measured values of core reactivity were then corrected to a reference set

Figure 3.1

Change in Core Reactivity
vs
Moderator Temperature

TABLE 3.1

MEASURED TEMPERATURE COEFFICIENTS OF REACTIVITY

T_{Pri}	T_{Mod}	$\frac{\partial \rho}{\partial T_{\text{Pri}}}$	$\frac{\partial \rho}{\partial T_{\text{Mod}}}$	Time of Measurement	Method
530°F	140°F		$-.35 \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$	Cycle 1	Reactivity Balance
217°F	107°F		$-(.22 \pm .02) \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$	Cycle 1	Period Measurement
216°F	106°F		$-(.25 \pm .03) \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$	Cycle 1	Period Measurement
525°F	145°F		$-.37 \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$	Cycle 2 Run 1	Reactivity Balance
522°F	147°F		$-.36 \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$	Cycle 2 Run 2	Reactivity Balance
230°F	125°F	$-.24 \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$		Cycle 1	Reactivity Balance
250°F	120°F	$-.20 \times 10^{-4} \frac{\Delta \rho}{\text{°F}}$		Cycle 2	Reactivity Balance

of primary and moderator temperatures through the appropriate temperature coefficients to compensate for small variations in system temperatures between individual reactivity measurements.

One measurement of the differential moderator worth was conducted at the beginning of Cycle 1 life and three were conducted during Cycle 2 life at various burnups. All of the data obtained are plotted in Figure 3.2. The data obtained at the 9420 MWD/MTU shutdown (End of Core I) are also included on this figure.

3.5.3 Integral Control Rod Worth

The integral worth of the control rod groups employed in the experiments previously discussed can be determined by integrating the differential moderator worth curve over the applicable range. Control rod withdrawal was used to compensate for moderator level reduction in going from one critical moderator height to another. Therefore, the reactivity worth of the change in moderator height between critical levels with the particular control rod group fully inserted and completely withdrawn should be the reactivity worth of the rod group. Since these measurements were performed in a core of less than full height, the effect of the uncovered portion of the core on integral rod worth was not incorporated in the measurement. Presumably, in a core of uniform axial composition, this effect is negligible. Some error may exist in applying these results to the full core integral rod worths which were measured in a core significantly shorter than full height and with a pronounced axial variation in composition.

The values of integral control rod group worth obtained in this manner are tabulated in Table 3.2. Additionally, integral group worths computed by the "inverse normalization" of differential rod worth data (see next section) are given in the table.

In addition to this work, an investigation was conducted to determine the effect of power operation at xenon equilibrium on the integral control rod worth curve. The results of this work were previously

Figure 3.2 Differential Moderator Worth vs Moderator Height

TABLE 3.2

MEASURED INTEGRAL CONTROL ROD GROUP WORTHS

<u>Measurement</u>	<u>Rod Group</u>	<u>Other Groups Inserted</u>	<u>Group Worth (% $\Delta\rho$)</u>	<u>Method of Measurement</u>
Cycle 1 (BOL)	IV	V	2.56	Differential Moderator Worth
Cycle 1 (BOL)	V	None	2.60	Differential Moderator Worth
Cycle 1 (BOL)	II and VII	IV, V	5.79	Inverse Normalization
Cycle 1 (BOL)	IV	V	2.21	Inverse Normalization
Cycle 1 (BOL)	V	None	2.46	Inverse Normalization
Cycle 2 (BOL)	V	II	3.66	Differential Moderator Worth
Cycle 2 (BOL)	II	None	3.76	Differential Moderator Worth
Cycle 2 (37 ⁴ EFPH)	II	None	3.95	Differential Moderator Worth
Cycle 2 (647 EFPH)	II	None	3.95	Differential Moderator Worth
Cycle 2 (BOL)	IV and VII	II, V	6.52	Inverse Normalization
Cycle 2 (BOL)	V	II	2.85	Inverse Normalization
Cycle 2 (BOL)	II	None	3.55	Inverse Normalization

presented in a Quarterly Progress Report^{/12} which described the effect of xenon on the differential, and consequently the integral, control rod worth at zero power.

Control rod group V was selected as the group for investigation since most of the group worth was withdrawn during the life of Cycle 1. Group V worth was 2.60% $\Delta\rho$ at the beginning of Cycle 1 life as determined by differential moderator height analysis.

The following analysis was undertaken to determine the shape of the Integral Control Rod Worth Curve. At the end of each day, the average position of Group V rods was obtained from plant operating records and plotted versus the cumulative Cycle 1 exposure in EFPH. Group V position versus Cycle 1 exposure is plotted in Figure 3.3. The break in the data at Group V rod position of approximately 45 inches is due to a reactor shutdown and subsequent xenon transient after the following startup. The shift of the data to the right in the interval from Group V rod position of approximately 48 inches through 96 inches is due to operation at a higher power level than achieved prior to the reactor shutdown at a rod position of 42 inches. The increase in power results in rod movement at 48" withdrawn other than that necessary for burnup compensation.

The break in the data over the interval from 0 to 17 inches withdrawn occurs since the preceding control rod group in the withdrawal sequence was not fully withdrawn during this interval. Since the preceding control rod group was not fully withdrawn at the initial withdrawal of Group V rods, the Cycle 1 exposure at which the initial withdrawal of Group V rods occurred was not precisely known. Therefore, the Cycle 1 exposure at the beginning of Group V withdrawal was estimated based on the previous group withdrawal.

Assuming a constant depletion rate, Figure 3.3 represents the integral control rod worth curve for Group V. The Cycle 1 exposure at which Group V withdrawal began was estimated to be 555 EFPH. The exposure at which Group V rods were fully withdrawn (100 inches) was

Figure 3.3 Control Rod Group V Position vs Cycle 1 Exposure (EFPH)

estimated to be 2110 EFPH. Using these parameters and a smooth curve drawn through the data in Figure 3.3, the Integral Control Rod Worth Curve shown on Figure 3.4 can be constructed.

The integral control rod worth curve constructed in Figure 3.4 is compared to the xenon free integral curve on Figure 3.5. Also shown on Figure 3.5 is an integral control rod worth curve constructed from differential control rod worth data obtained during a xenon transient and reported in a previous quarterly progress report¹².

3.5.4 Differential Control Rod Worth

Several differential control rod worth measurements were performed in order to establish the integral control rod worth through a technique of "inverse normalization". From repeated measurements of the differential control rod worth curve, it was apparent that the normalized (to a 1% $\Delta\rho$ group worth) differential control rod worth curve was relatively insensitive to group worth and core depletion. Therefore, a normalized differential rod worth curve (Figure 3.6) was assumed valid before the measurement and a technique of "inverse normalization" was employed as follows: Several differential rod worth points were measured at an average rod height by performing a series of startup rate measurements at slightly different positions of the control rod group of interest. Other core parameters were unchanged during the series of measurements. The moderator level was changed to permit successive series of measurements at different average control rod heights. Each series of measurements, when corrected to full core height, defines a differential rod worth point. Points for a normalized differential rod worth curve were computed as follows:

$$\left. \frac{\partial \rho}{\partial h} \right|_{\text{rod Normalized}} = \left. \frac{\partial \rho}{\partial h} \right|_{\text{rod Full height}} \div \frac{\rho}{100} \quad (3.4)$$

where 100" $\approx$ effective length of control rod travel

ρ = the integral group worth.

Figure 3.4 Integral Control Rod Worth Curve

Figure 3.5 Integral Control Rod Worth Curves

Figure 3.6 Normalized Differential Control Rod Worth

Therefore, the ratio of the full core height differential rod worth to the corresponding value on the assumed normalized differential rod worth curve was defined as the integral group worth (in % $\Delta\rho$). The results using this method are shown in Table 3.2

3.5.5 Depletion Rate

Two methods were employed to determine the reactivity depletion rate for the fuel management cycles. The first method consisted of determining the amount of reactivity depleted over an interval of control rod movement. The second method employed the use of the installed reactivity measurement made at the beginning of each cycle and the implied installed reactivity at the end of each cycle. The installed reactivity was obtained from integration of the differential moderator worth curve from the all rods out critical height to full effective core height.

The control rod movement method utilized to determine the Core I depletion rate was considered to be less accurate than other methods due to the dependence of control rod worth on core depletion. However, this method is considered to be more valid for the fuel management cycles since the cycling depletion was considerably less than the Core I depletion. To determine the depletion rate using this method, a set of parameters bracketing a significant control rod movement was chosen. The parameters were control rod position and cycle depletion. Using the control rod worths previously determined and the integral control rod worth curve, the amount of reactivity depleted between the two sets of conditions was determined. The depletion rate is the cycle depletion between the two sets of conditions divided by the reactivity depleted between these two sets of conditions. Table 3.3 tabulates the parameters chosen and the depletion rates obtained from fuel management cycling.

The second method of determining the depletion rate employed the use of installed reactivity measurements. The difference between the

TABLE 3.3

FUEL MANAGEMENT DEPLETION RATES BY CONTROL ROD MOVEMENT METHOD

<u>Fuel Cycle</u>	<u>Initial Parameter</u>		<u>Final Parameter</u>		<u>Reactivity Depleted by Rod Movement (% $\Delta\rho$)</u>	<u>Interval Depletion (MWD/MTU)</u>	<u>Depletion Rate $\frac{\text{MWD/MTU}}{\% \Delta\rho}$</u>
	<u>Rod Position</u>	<u>Cycle Burnup (MWD/MTU)</u>	<u>Rod Position</u>	<u>Cycle Burnup (MWD/MTU)</u>			
Cycle 1	IV at 60"	123	All Out	2087	3.53	1964	556
Cycle 2	II at 35.1"	1089	II at 51.3"	1550	.82	461	562

TABLE 3.4

FUEL MANAGEMENT DEPLETION RATES BY INSTALLED REACTIVITY METHOD

	<u>Beginning of Life Cycle 1</u>	<u>End of Life Cycle 1</u>
Core Average Burnup	8110 MWD/MTU	
Initial Installed ρ Measurement	7.22% $\Delta\rho$	.02% $\Delta\rho$ (Est.)*
Correction to:		
a. Xe Free	0	3.07
b. Sm Free	1.41	.86
c. 0 Power	0	.52
d. 100°F Core	0	1.08
Change in effective top of core	0.61	0
TOTAL	9.24	5.55

*Estimated reactivity available for depletion. Estimate made at power, equilibrium conditions.

$$\text{Cycle 1 Depletion Rate} = \frac{2087 \text{ MWD/MTU}}{9.24 - 5.55\% \Delta\rho} = 566 \frac{\text{MWD/MTU}}{\% \Delta\rho}$$

installed reactivity at the beginning of the cycle and the installed reactivity at the end of the cycle is the reactivity depleted during the cycle. No physics tests were conducted at the end of the fuel cycles, so a reactivity balance was necessary to determine the installed reactivity under the same conditions that the beginning of life measurement was conducted. It was also necessary to correct the initial beginning of life measurement for the change in effective top of core and the effect of Sm-149 in a similar manner to that described in previous reports /3,13,14. The depletion rate determined by this method for Cycle 1 is shown in Table 3.4. Attempts to determine the Cycle 2 depletion rate using this method were unsuccessful due to the large amount of installed reactivity remaining in the core when reactor operations was terminated.

4.0 CORE PERFORMANCE COMPARED WITH DESIGN

4.1 Cycle Burnup

The reactor was shutdown in late August at the end of Cycle 1 operation. The Cycle 1 burnup at that time was 2077 MWD/MTU (2100 EFPH). Based on the movement of Group V control rods, an estimated 2087 MWD/MTU (2110 EFPH) could have been achieved during Cycle 1 operation.

The TURBO* calculations predicted the Cycle 1 burnup to be 1523 MWD/MTU (1540 EFPH) and assumed no Core I reactivity available for depletion. At the termination of Core I operation, approximately 1.15% $\Delta\rho$ was available for depletion. The eight assemblies removed during the refueling operation prior to Cycle 1 operation had an average gross radial factor (F_R) of approximately 1.25. The reactivity available for depletion carried over from Core I to Cycle 1 was estimated to be 1.05% $\Delta\rho$.

Assuming a depletion rate of 560 MWD/MTU for 1.05% $\Delta\rho$, the additional capability of Cycle 1 would be 588 MWD/MTU. The corrected TURBO* prediction would be 1523 MWD/MTU, plus 588 MWD/MTU or 2111 MWD/MTU. The difference between the TURBO* prediction of 2111 MWD/MTU and the estimated capability of 2087 MWD/MTU is about 1%.

Cycle 2 operation was terminated prematurely having an accumulated burnup of only 1600 MWD/MTU. A calculated lifetime of 2425 MWD/MTU was expected for Cycle 2 operation. Prior to Cycle 2 termination, an accumulated life of 1550 MWD/MTU was experienced with control rod group II banked at 51.3 inches. By extrapolating the remaining worth of group II a total expected lifetime for Cycle 2 may be derived.

The total measured worth of control rod group II for the Cycle 2 core was 3.55% $\Delta\rho$ as given in Table 3.2. Since group II was banked at 51.3 inches, approximately 47% of the worth of II remained in the core as observed from Figure 3.4. Therefore, the total life of Cycle 2 is 1550 MWD/MTU plus the remaining worth of group II which was 938 MWD/MTU using a depletion rate of 562 MWD/MTU / % $\Delta\rho$, as presented in Table 3.3. Thus, the resultant 2488 MWD/MTU derived from operation and measurements can be compared to the

2425 MWD/MTU expected from calculations resulting in a 2.6% difference which is excellent agreement. Table 4.1 summarizes the calculated and measured lifetimes and the percent differences.

TABLE 4.1
COMPARISON BETWEEN OBSERVED AND PREDICTED CYCLE LIFETIMES

<u>Cycle</u>	<u>Observed Lifetime (MWD/MTU)</u>	<u>Predicted Lifetime (MWD/MTU)</u>	<u>Difference</u>
1	2111	2087	1.0%
2	~2488*	~2425**	2.6%

*Inferred from integral control rod worth measurements.

**Does not account for changes due to fuel leak replacements.

4.2 Control Rod Worths

TURBO* calculations were performed to provide calculated control rod worths for direct comparison with measurements at specific lifetimes. The results of these calculations are tabulated in Table 4.2 for Cycle 1 and Cycle 2 of the fuel management cycling program. A comparison between calculated and measured values is presented in Table 4.3. The measured values were determined through integration of the differential moderator height and through use of the "inverse normalization" technique described previously in Sections 3.5.2 and 3.5.4, respectively.

4.3 Radial Power Sharing

The CVTR is instrumented such that a measure of the relative radial power sharing for preselected fuel assemblies may be made in either of two ways. By indirect measurements in terms of Mn-56 activation induced from neutron bombardment on steel wires containing manganese or by a direct measurement of the temperature rise and the rate of coolant flow through

TABLE 4.2

TURBO* PREDICTED CONTROL ROD WORTHS

<u>Cycle</u>	<u>Burnup</u>	<u>Rod Group</u>	<u>Control Rod Group Worth</u>	<u>Eigenvalue</u>	<u>Other Groups Inserted</u>	<u>Burnup* Correction (% $\Delta\rho$)</u>
1	0	V IV II and VII	2.43 2.59 8.11	1.0250	All Out	
1	0			1.00006	V	
1	0			0.9748	IV, V	
1	0			0.9034	II, IV, V, VII	
2	100	IV and VII V II	5.93 3.23 3.65	0.9141	II, IV, V, VII	
2	100			0.9665	II, V	
2	400			0.9918	II	0.59
2	1400			1.0087	All Out	1.96
2	500	II	3.81	0.9841	II	1.76
2	1400			1.0043	All Out	

*Assumes a depletion rate of 510 EFPH/% $\Delta\rho$.

TABLE 4.3

FUEL MANAGEMENT INTEGRAL CONTROL ROD WORTHS

<u>Cycle Depletion</u>	<u>Rod Group</u>	<u>Other Groups Inserted</u>	<u>Method of Measurement</u>		<u>Calculated (% $\Delta\rho$)</u>
			<u>Integration of Differential Mod. (% $\Delta\rho$)</u>	<u>Inverse Normalization (% $\Delta\rho$)</u>	
Cycle 1 (BOL)	IV	V	2.56	2.21	2.59
Cycle 1 (BOL)	V	None	2.60	2.46	2.43
Cycle 1 (BOL)	II and VII	IV, V	-	5.79	8.11
Cycle 2 (BOL)	V	II	3.66	2.85	3.23
Cycle 2 (BOL)	II	None	3.76	3.55	3.65
Cycle 2 (BOL)	IV and VII	II, V	-	6.52	5.93
Cycle 2 (374 EFPH)	II	None	3.95	-	3.81
Cycle 2 (647 EFPH)	II	None	3.95	-	-

specific pressure tubes. Throughout operation of Cycles 1 and 2 both measuring techniques were used in order to maintain a complete history of the relative power sharing within the reactor core. The measurement results were then compared with design predictions in order to establish the validity of the analytical techniques used. Throughout operation, very good agreement existed between the calculated and measured power distributions. A rather large discrepancy existed when the thermocouple installed in the U-bend of the special instrumented pressure tube (8R) was used to establish the fractional power split for the special test assembly as presented in Section 6.0. In addition, several problems were encountered while using the flux wire system which limited the amount of data available from the system. Specific details of the problem areas encountered are also presented in Section 6.0.

4.3.1 Flux Wire Measurements Versus Design

Several flux wire irradiation measurements were performed throughout operation of Cycles 1 and 2. These measurements were compared to design predictions and also used to complement the thermal-hydraulic measuring devices. The specific details of each flux wire measurement are presented in Table 4.4. The results in the table were obtained upon reduction of the flux wire data through the CAVIAR code. All radial power distributions and axial peaking factors obtained from the flux wire irradiation measurements are presented in detail in Table 4.5.

In general, good agreement existed between the predicted values and those measured with flux wires. A typical example is given in Figure 4.1. A gross comparison showing the overall statistical evaluation of the calculated values compared to flux wire results is given on Figure 4.2. In this case the total number of samples was 84 which resulted in a probable error of $\sim 5\%$ for a 1σ variation as illustrated in the figure. In the figure the integral of the frequency distribution, which gives the probability (P) that a measurement (M) is less than or equal to the percent deviation (D) is presented. It is observed that 67% of the measurements agree to within 5% corresponding to the 1σ confidence level. At the 95% confidence level (2σ) the error is 11%.

TABLE 4.4

POWER SHARING MEASUREMENTS PERFORMED WITH FLUX WIRES

Identification Number	Date Performed	Fuel Cycle	Cycle Life (EFPH)	Power (MWt)	No. of Passes	No. of Thimbles Measured	Control Rod Configuration					
							I	II	III	IV	V	VII
Experiment #24	5/18/66	1	85	51.5	3	13				57"	In	
Experiment #25	6/08/66	1	474	52.1	3	10				87.1"	In	
Experiment #26	7/13-15/66	1	1154	49.0	3	10					40.0"	
Experiment #27	8/1-3/66	1	1284	55.9	3	10					61.5"	
Flux 2A*	10/16/66	2	162	45.6	1	2		In			75"	
Flux 2B	10/21/66	2	262	55.6	1	2		In			89.3"	
Flux 2C	10/24/66	2	312	55.6	1	2		13.5"			83.9"	
SUCV-109 Flux 1	11/13/66	2	407	32.4	1	2		19.3"		68.6"	43.1"	
SUCV-109 Flux 2	11/13/66	2	407	31.9	1	3		34"		34"	34"	
SUCV-109** Flux 3	11/21/66	2	512	52.0	1	4		45"		45"	45"	

*Control rod configuration similar to the configuration existent when high power density test assembly (25.1 kw/ft peak; group 3) located at 3C failed.

**Measurement was conducted during pushed power banked rod tests. It is postulated that during this time three failures occurred consisting of: 1) high power density test assembly (25.1 kw/ft peak; group 1) located at 8Q; 2) standard 2.0 w/o assembly in pressure tube 6J; 3) standard 2.0 w/o assembly in pressure tube 7J.

TABLE 4.5

SUMMARY OF RADIAL LOCAL-TO-AVERAGE POWER SHARING AND AXIAL
MAXIMUM-TO-AVERAGE RATIOS FROM FLUX WIRE MEASUREMENTS

U-Tube Leg Location	Expt. #24		Expt. #25		Expt. #26		Expt. #27		Flux 2A	Flux 2B	Flux 2C	SUCV-109 Flux 1	SUCV-109 Flux 2	SUCV-109 Flux 3
	F _R	F _Z	F _R	F _Z	F _R	F _Z	F _R	F _Z	F _Z	F _Z	F _Z	F _Z	F _Z	F _Z
6A	0.835	1.21	0.778	1.25	0.822	1.29	0.868	1.30						1.63
6C	0.758	1.21	0.709	1.25	0.838	1.44	0.937	1.38						1.63
6E	0.746	1.20	0.743	1.25	0.861	1.36	0.946	1.33						
6G	0.977	1.20	0.973	1.25	0.994	1.25	1.026	1.24						
6I	1.488	1.20	1.474	1.23	1.421	1.22	1.383	1.21						
7A	0.835	1.19	0.817	1.23	0.828	1.27	0.840	1.27						
7C	0.952	1.19	0.944	1.23	0.962	1.27	0.979	1.27						
7E	0.902	1.22	0.911	1.23	0.906	1.23	0.907	1.25				1.37		
7G	0.975	1.21	0.993	1.23	0.951	1.24	0.931	1.22				1.37		
7I	1.427	1.22	1.438	1.22	1.373	1.24	1.327	1.20						
8A	0.697	1.21	0.673	1.18	0.700	1.17	0.696	1.17					1.25	1.34
8C	1.169	1.24	1.132	1.23	1.121	1.28	1.102	1.23					1.40	1.55
8E	1.096	1.25	1.058	1.23	1.055	1.22	1.009	1.25						
8G	1.125	1.23	1.222	1.24	1.104	1.22	1.050	1.24						
8I	1.154	1.25	1.211	1.24	1.147	1.21	1.112	1.22						1.73
9E	1.005	1.32	0.973	1.24	0.973	1.22	0.922	1.24						
9G	0.915	1.27	1.024	1.26	1.029	1.24	0.967	1.23						
9I	0.979	1.30	1.047	1.25	1.040	1.23	0.988	1.21						
3A	1.048	1.22	0.970	1.22					1.25	1.15	1.08			
3C	0.782	1.14	0.723	1.17					1.17	1.20	1.12			
3I													1.52	
3Q	1.163	1.21												
3S	0.688	1.20												
8Q	0.779	1.16	0.718	0.718	0.720	0.720	0.698	1.16	1.19	1.16	1.09	1.20	1.24	1.35
8S	1.044	1.24	0.961	1.25	0.963	0.963	0.955	1.27	1.27	1.21	1.14	1.32	1.38	1.55
2E			0.965	0.965	0.923	0.923	0.899	1.23						
2G			1.018	1.27	0.976	0.976	0.942	1.22

Figure 4.1 Example of Comparison Between Flux Wire Measurements and Calculated Radial Power Sharing

Figure 4.2 Cumulative Distribution Curve for TURBO* Calculated Power Sharing Versus Flux Wire Measurements

Figure 4.3 Comparison Between Flux Wire and Thermocouple Power Sharing Measurements

4.3.2 Thermocouple Measurements Versus Flux Wires

Power distribution measurements were also performed using the thermal-hydraulic measuring devices at the site consisting of thermocouples and the individual coolant flow rates. These measurements were usually performed in conjunction with the flux wire instrumentation on a complementary basis. The resultant power distributions were then compared to those obtained from flux wire measurements. These were treated the same as described in the preceding section resulting in a direct comparison between the two measuring techniques used. In general, the agreement is good as demonstrated on Figure 4.3 which gives the statistical variation between the two measurement techniques. Two separate temperature sensing devices were available, plant thermocouples and R & D thermocouples. In the results presented, both the plant and R & D thermocouples were compared to flux wire results. Figure 4.4 gives the pressure tube identification of the available in-core instrumentation which were used for power sharing measurements on the CVTR.

4.4 Radial Peaking Versus Control Rod Withdrawal

A detailed history of the radial peaking factor was maintained throughout Cycle 1 operation as determined from the in-core instrumentation. Figure 4.5 gives the results obtained and illustrates good agreement between measured and design values as a function of both core burndown and control rod withdrawal. In the figure the dotted lines represent a 5% deviation from design. The measurement results illustrate that both flux wires and the R & D thermocouples show excellent agreement with design. The results as given in the figure are consistent with similar results for Core I operation. No quarter core flux wire maps were obtained during Cycle 2 operation but thermal-hydraulic measurements also agreed with design.

4.5 Axial Peaking Measurements

No satisfactory calculational techniques were developed which could accurately predict the axial peaking factor anticipated at particular radial

LEGEND

PLANT - Plant Thermocouple Instrumentation

● - Flux Wire Instrumentation

R&D - R&D Thermocouple Instrumentation

Figure 4.4 Plan View of In-Core Instrumentation Used For Power Sharing Measurements

Figure 4.5 Radial Peaking Factor Versus Core Burndown During Cycle 1 Operation

positions on CVTR. Although the gross core value could be calculated, past experience from Core 1¹⁹ had demonstrated that the axial model was inadequate for selecting axial factors at specific radial positions. Since a thorough knowledge of the anticipated axial peaking for the special test assembly location was required, a series of physics measurements were performed on the reactor in order to determine reliability of calculational technique and variation of F_z with control rod motion.

The measurements consisted of a series of hot, low-power experiments which were carried out in CVTR Core 1 after approximately 7000 EFPH operation. Measurements of the axial peaking factor for four different control rod configurations were performed using flux wires to measure the axial power profile at preselected radial positions. The experimental results are summarized in Table 4.6. One-dimensional calculations were then performed using an axial model in the CANDLE code in order to determine the feasibility of duplicating the measured values analytically. This analysis was carried out for two of the four configurations given in Table 4.6; Configurations #1 and #2.

TABLE 4.6

TABULATION OF AXIAL PEAKING FACTORS ON CORE I AT 7000 EFPH OPERATION

<u>Rod Configuration</u>	<u>U-Tube</u>	<u>F_z</u>	<u>Height of F_z</u>
1. IV, V, VII at 39.3"	6J	1.54	~23"
I, III-Out; II-In	4N	1.65	~22"
2. II, VII at 55.2"	4N	1.48	~30"
I, III-Out; IV, V-In	6J	1.46	~30"
3. V, VII at 53.0"	7F	1.46	~30"
I, III-Out; II, IV-In	6J	1.40	~28"
	9F	1.44	~30"
4. IV, V at 63.3"	7F	1.36	~35"
I, III-Out; II, VII-In	6J	1.32	~34"
	9F	1.40	~34"

Previous measurements had shown that the worth of a control rod group is strongly dependent upon the other rod groups in the core. Consequently, a basic problem in comparing the analytical model with experiment was the necessity of assuming how to distribute the total reactivity worth between banked and fully-inserted rods. Attempts to match experiments 1 and 2 with the CANDLE code were unsuccessful. In the calculations, the control rods were represented by an increase in absorption cross section in those regions simulating the banked and fully inserted rod configurations. The results of the analysis demonstrated that the analytical model was satisfactory in determining core average effects but could not determine the local effects near banked control rods.

It is postulated that an elaborate three-dimensional calculation could adequately predict the localized effects but would be expensive and require extensive engineering efforts. Therefore, it was decided to measure these effects during Cycle 1 operation and extrapolate the measured values to future fuel cycling.

4.5.1 Cycle 1 Axial Peaking Factors

Due to the unsuccessful attempts to provide a simple analytical model to predict the axial peaking factor with two banked rod groups, it was necessary to perform additional measurements to establish by direct observation, the various effects which contribute to the axial peaking factors. Therefore, during Cycle 1 operation, several single wire irradiations were conducted in order to establish the relation of control rod position with the axial profile. These irradiations were performed in position 8R to determine the trend of the axial peak-to-average factor (F_z) with control rod withdrawal. The axial factors obtained from these experiments are tabulated in Table 4.7 for the 25.1 kw/ft high power density test assembly (78.4 inch fuel) and the standard 2.0 w/o U-235 fuel (95.4 inch fuel) in tube 8R. The results showing the variation of the axial factor with control rod withdrawal are presented in Figure 4.6.

TABLE 4.7

AXIAL MAXIMUM TO AVERAGE FACTOR IN U-TUBE 8R

<u>Date</u>	<u>Control Rod Position (inches)</u>	$\frac{8Q}{F_z}$ <u>(78.4")</u>	$\frac{8S}{F_z}$ <u>(95.4")</u>
Group V			
June 24	21.7	1.152	1.211
June 28	26.2	1.134	1.200
July 15	40.7	1.151	1.225
July 20	46.1	1.162	1.242
July 25	52.9	1.172	1.251
August 3	62.4	1.174	1.251
August 10	70.0	1.160	1.221
August 17	78.2	1.157	1.202
August 24	90.1	1.176	1.215

Figure 4.6 Variation of Axial Maximum to Average Factor With Control Rod Group V Movement at U-Tube 8R

Only the positions containing the special high power density test assemblies were investigated since in these cases it was necessary to have a thorough knowledge of the peak power densities anticipated during any cycle of operation.

5.0 HIGH POWER DENSITY SPECIAL TEST ASSEMBLIES

Throughout operation of the fuel management program for CVTR the special high power density test assemblies became a primary consideration in reactor operations. It was originally intended that the test assemblies would operate at significant power levels consistent with overall plant objectives. Due to discrepancies in the radial power sharing of the test assemblies, as inferred from the special U-bend thermocouple, the test assemblies became limiting. This resulted in reducing the plant power operation level. Additional tests were then devised to substantiate or invalidate the U-bend thermocouple measurements consisting of decay heat and gamma irradiation measurements. Based on the results of these tests the U-bend thermocouple results were ultimately discarded for absolute power determination and normal power operation was resumed.

The original program objectives of the test assemblies were not met due to fuel failures in these assemblies during Cycle 2 operation and subsequent withdrawal from the reactor core. A post-construction examination program is currently underway to determine the nature and cause of the fuel failures.

5.1 Program Objective

The original objective of the special high power density test assemblies was to establish a basis for improvement in conventional power reactors by extending the present conservative fuel performance limitations. This was to be accomplished by fabrication of four special test assemblies, which were then to be irradiated in the CVTR for approximately two years. Each assembly contained six UO_2 fuel rods enriched with U-235 to yield peak steady state power levels of 20.9 and 25.1 kw/ft to a maximum burnup approaching that of 50,000 MWD/MTU. Various combinations of fabrication procedures were used in an effort to determine long term irradiation and power operation effects and optimize future fuel clad fabrication design^{/20}.

To comply with the overall objectives as listed on the previous page, it was necessary to design the core power distribution such that the test assemblies would operate as near the design peak power densities for as long a period as possible. Thus, in addition to the usual high power limitation which is placed upon all fuel rods, the test assemblies also had a lower limit. The lower limit was chosen as a compromise between convenient plant operation and test objectives. In addition to maximum and minimum power level limitation requirements, the test assemblies also had to operate in the inlet side of the pressure tubes, further complicating the design efforts. This restriction resulted in asymmetry in the core since the CVTR core did not maintain quarter core symmetry for the inlet and outlet legs of symmetry pressure tubes. Therefore, the calculational model used throughout Core I operation, which assumed quarter core symmetry, was no longer valid for the core locations containing the high power density test assemblies. This necessitated additional calculations to determine the perturbation on the overall power distribution from the existent asymmetry. Results from these calculations demonstrated that the asymmetry was a localized effect in that only pressure tube 8B and its symmetric locations had a substantial difference in the relative radial power sharing.

5.2 Description of the Test Assemblies

Four special high power density test assemblies were fabricated each containing six fuel rods which were designed to operate at peak power densities of either 25.1 or 20.9 kw/ft. The remaining 13 rods of the conventional 19 rod cluster for CVTR were inert fuel rods containing Al_2O_3 . The four test assemblies may be conveniently categorized into two separate groups having similar nuclear characteristics.

Group 1 - Two assemblies each containing six fuel rods enriched to 3.8 w/o and designed to operate at 20.9 kw/ft at the peak.

Group 2 - Two assemblies each containing three fuel rods enriched to 3.8 w/o, designed to operate at 20.9 kw/ft and three fuel rods enriched to 4.7 w/o, designed to operate at 25.1 kw/ft at the peak.

The average design nuclear power density in the group 1 test assemblies is 20 kw/ft* and that of group 2 was 22 kw/ft. The 22 kw/ft value is the linear average of 20 and 24 kw/ft**, respectively. The specific physical characteristics of each fuel rod in the test assemblies are tabulated in Table 5.1. A cross sectional view is given in Figure 5.1. The mechanical design was based upon standard CVTR design with the exceptions being a removable center fuel section in each assembly and individual fuel rod design characteristics. The fuel column lengths were 78 and 81.25 inches which differed from the standard fuel length of 95.4 inches. The shorter length fuel column was used to provide additional space for release of fission gases.

Due to DNB consideration, the six UO_2 fuel rods in each test assembly were located in the center ring of the hexagonal cluster and inserted in the inlet legs of the pressure tubes selected.

5.3 Enrichment Specification

The enrichments specified for the special high power density test assemblies were obtained from combined PLMG-TURBO* calculations. These calculations assumed the UO_2 density as 93% theoretical. As in the original design objectives, it was assumed that three of the six high power density rods (4.7 w/o) would operate at 25.1 kw/ft and the other three (3.8 w/o) would be operated at 20.9 kw/ft at the peak. The six high power density fuel rods were located in the central ring of the 19 rod cluster. The remaining 13 rods of the cluster were assumed to be pure Al_2O_3 and clad in Zr in the calculations. The pellet diameter of 0.4286 inch was used and the density of the Al_2O_3 pellets taken to be 3.7 gm/cc. The pressure tube, thermal baffles and remainder of the fuel housing group were identical to the standard CVTR fuel design. PLMG-TURBO* calculations resulted in specifying the following enrichments in Table 5.2 for the peak power densities and conditions as outlined above.

*20 kw/ft becomes 20.9 kw/ft with inclusion of the engineering factor of 1.045.

**24 kw/ft becomes 25.1 kw/ft with inclusion of the engineering factor of 1.045.

TABLE 5.1

PHYSICAL DATA OF CVTR TEST RODS

	Assemblies #1, 2, 3		Assembly #4			Standard CVTR Assembly (65 MW)
No. Rods per Assembly (UO_2 , Al_2O_3)	19		19			19
No. UO_2 Rods per Assembly	6		6			19
Clad Material	Zr-4		(2) Zr-4)	(2) Zr-4)	(2) #304 SS	Zr-4
Clad Type	freestanding		thin-wall	freestanding	thin-wall	freestanding
Clad Outside Diameter, in. (± 0.0015)	0.4875		0.4875	0.4875	0.4875	0.4875
Clad Thickness (nominal), in.	0.024		0.017	0.024	0.0125	0.0258
Sp. Power Density, Kw/ft (peak)	20.90	25.08	20.90	20.90	20.90	17.00
Nominal Cold Pellet-Clad Gap (diam.), in.	0.0066	0.0078	0.0060	0.0066	0.0065	0.0060
Pellet Diameter, in. (± 0.0005)	0.4329	0.4317	0.4475	0.4329	0.4560	0.4300
Pellet Length, in. (± 0.0500)	0.650	0.650	0.650	0.650	0.650	0.530, 0.636
Pellet Dish Diameter, in. (± 0.0150)	0.350	0.350	0.360	0.360	0.360	0.359
Dish Depth, in. (± 0.0035)	0.0135	0.0135	0.0135	0.0135	0.0135	0.0125, 0.0145
Cold Fuel Column Length, in. (± 0.350)	81.250*	77.80*	81.250	81.250	81.250	95.400
Hot Void Volume (Min.), in. ³	2.121	2.666	2.268	2.121	2.360	
Cold Void Volume (Min.), in. ³	2.180	2.753	2.268	2.180	2.360	
UO_2 Enrichment, w/o U-235	3.80	4.70	3.80	3.80	3.80	1.5, 2.0
UO_2 Density, % T.D. ($\pm 1.5\%$)	93, 96	93, 96	93	93	93	93

*Nominal value; individual fuel rod lengths vary.

Figure 5.1 CVTR High Power Density Fuel Assemblies

TABLE 5.2

SPECIAL TEST ASSEMBLY FUEL ROD ENRICHMENTS

<u>Power Density</u>	<u>Enrichment (w/o U-235)</u>
25.1 kw/ft	4.70 w/o
20.9 kw/ft	3.80 w/o

A plot of the peak nuclear power in the high power density fuel rods is shown in Figure 5.2 as a function of the U-235 enrichment. The axial peaking factor in this case (F_z) was assumed as 1.4 and the power variation within the cluster (F_ϕ) was taken as 1.05. The results presented in the figure were generated from TURBO* calculations. The specified enrichments were based upon replacing an original 2.0 w/o fuel assembly which was operating at 92% power with the high power density test assembly. An average value of 22 kw/ft was assumed for the peak nuclear power output from each of the six high power density fuel rods and the required enrichments were extrapolated from the average value to obtain the required 24 kw/ft and 20 kw/ft nuclear ratings.

It is emphasized, however, that the specified enrichments were obtained using a TURBO* history tape having an accumulated burnup of approximately 5700 EFPH on Core I. Maintenance of the specified power levels with the existent enrichments were later accomplished by proper selection of the relative radial position during refueling.

A few points of specific interest which pertain to the test assemblies are given below.

- (1) Replacement of 13 standard fuel rods in the nineteen rod cluster with Al_2O_3 rods resulted in a total assembly power of 714 Kw as compared to the 856 Kw in an average assembly, assuming peak nuclear values of 24 and 20 Kw/ft in the 6 special fuel rods.
- (2) No uncertainty was used in the calculations relating the power level of the special test assemblies.

Figure 5.2 Calculated Peak Power Versus Enrichment in the High Power Density Test Fuel Rods

- (3) Operating conditions were based on a 50% probability that the fuel would operate at the design objective, therefore, a significant probability of exceeding the design objective was accepted.
- (4) Axial peaking factors were obtained from the most recent prevailing flux distribution in the core and the subsequent radial location was chosen to meet design objectives.

5.4 Critical Experiments

In view of the high probability of exceeding the peak operating power levels in the test fuel, critical experiments were performed to evaluate local peaking and flux asymmetries. Thus, the critical experiments helped to establish a high probability of meeting the design objectives while reducing the amount of overshoot, if any, by increasing the confidence of the design values.

A critical experiment program was carried out at the Westinghouse Reactor Evaluation Center to measure the relative fuel rod power sharing in a mock up test assembly containing fuel rods having different enrichments. The specific parameters of interest were the effect of enrichment differences on the fuel rod relative power, and the effect of a flux gradient across the assembly on the assembly power distribution. The intent of the program was to reduce the uncertainty in the operating peak power densities within the test assemblies in order to permit operation nearer the design values. The experiments were performed with fuel rods simulating the actual CVTR test assembly. The experimental results were later used for direct comparison with calculational techniques and also provided experimental factors to relate the average fuel rod power to the peak rod power. These factors were used specifically to account for power variation within the assembly due to differences in fuel enrichments and to account for flux gradients across the assembly.

5.4.1 Critical Experimental Results

Both power and flux measurements were made from a mock up of the CVTR test assemblies. The experiments were carried out with combinations of 3.7, 2.7 and 1.6 w/o UO_2 , stainless steel clad fuel rods. Figure 5.3 illustrates a plan view of the fuel assembly and the numbering scheme used. During the critical experiments this fuel assembly was contained within a 10 inch diameter cylindrical tank, filled with heavy water to simulate conditions in the CVTR core. The aluminum tank insert was surrounded by a light water moderated driving region containing stainless steel clad fuel enriched to 3.7 w/o in U-235. Figure 5.4 shows a cross section of the D_2O insert and driver region used.

The critical experiments were performed for two different lattice pitches in the fuel cluster simulating the CVTR test assembly, 0.6-inch and 0.4-inch triangular arrays. The 0.6-inch pitch was selected in order to maintain the same lattice spacing as in the standard CVTR assemblies. However, because of differences in fuel pellet radii and fuel enrichment, the water to uranium ratios differed from the CVTR values. Therefore, a second pitch was selected which maintained the same water to uranium ratio as in CVTR. The results of the experiments indicated no dependence on the lattice pitches used. Measurements were performed for both pitches with and without stainless steel rods inserted near the periphery of the aluminum tank. The steel rods were used to provide a flux gradient across the simulated fuel assembly.

In a typical measurement, fuel rods of two different enrichments were installed in alternate positions around the circle, irradiated, and counted. Short dysprosium aluminum wires (0.25" long) were placed on fuel rods at the lettered positions near the core midplane (Figure 5.3). Dysprosium wires of the same compositions were also threaded through the cluster in order to measure the flux distribution. Table 5.3 summarizes the program of experiments, outlining the specific measurements made during each experiment. The annular driver region

Figure 5.3 Plan View of Simulated Fuel Assembly With Numbering Scheme

Figure 5.4 Core Diagram of Critical Experiments

TABLE 5.3

CRITICAL EXPERIMENT PROGRAMComponents of Aluminum Can and Test Assembly Region

Run No.	No. of Test Rods Required				Pitch inches	Dysprosium Traverse Wires (Radial)	Fuel Rods With Dy Wire Segments	Dy Wire Around Can	No. of Fuel Rods To Scan
	3.7 w/o	2.7 w/o	1.6 w/o	304 SS					
1	6	-	-	-	0.6	3	6	1	6
2	3	3	-	-	0.6	0	2	0	6
3	3	-	3	-	0.6	1	2	0	6
4	-	3	3	-	0.6	0	2	0	6
5	6	-	-	9	0.6	3	6	1	6
6	3	3	-	9	0.6	0	4	0	6
7	3	-	3	9	0.6	1	4	0	6
8	-	3	3	9	0.6	0	4	0	6
9	6	-	-	0	0.4	0	4	0	6
10	3	3	-	0	0.4	0	0	0	6
11	3	-	3	0	0.4	0	0	0	6
12	-	3	3	0	0.4	0	0	0	6
13	3	3	-	9	0.4	3	2	1	6
14	-	3	3	9	0.4	0	0	0	6

consisted of 2072 fuel rods for the unrodded assembly and 2328 fuel rods when the stainless steel absorbers were inserted.

Tables IV.1 through IV.4 in Appendix IV presents all power production data, giving both fuel enrichments and the local-to-average power for each fuel rod. The power ratios are also given for each enrichment pair. The power ratios obtained from both pitches were averaged and plotted in Figure 5.5 as a function of the enrichment ratios. A power ratio of 1.19 was obtained for an enrichment ratio of 1.24 (4.7/3.8), thus indicating approximately 5% interaction due to mixed enrichment.

Based on the power sharing ratio obtained for the ratio of the 4.7 w/o and 3.8 w/o fuels, a maximum-to-average fuel rod power sharing ratio was determined as given in the following equations.

$$P_{4.7 \text{ w/o}} = 1.19 P_{3.8 \text{ w/o}} \quad (5.1)$$

$$F \text{ w/o} = \frac{P_{4.7 \text{ w/o}} (2)}{P_{4.7 \text{ w/o}} + P_{3.8 \text{ w/o}}} \quad (5.2)$$

Solving equations 5.1 and 5.2 results in $F \text{ w/o}$ equal to 1.09. Thus, based on these results, the maximum fuel rod power within the test assembly containing both 4.7 w/o and 3.8 w/o fuel is 1.09 times the average assembly power; for uniform azimuthal flow.

The results from the dysprosium-aluminum traverse wire scans were also reduced to determine the effect of flux gradients across the cluster on the relative fuel rod power sharing. A typical scan of the traverse wires is given in Figure 5.6. Additional scans gave similar results and were used to determine the flux gradient factor (F_{ϕ}) for direct application to the test assemblies. These data were reduced in order to determine the perturbation in the relative fuel rod power sharing within the test assemblies due to different flux gradients across the assembly. The results of these measurements shown in Figure 5.7 illustrate that an approximate 2.5% perturbation

Figure 5.5 Power Ratio versus Enrichment Ratio

DYSPROSIUM WIRE SCAN THROUGH FUEL CLUSTER AND STAINLESS STEEL ABSORBER RODS

Figure 5.6

Figure 5.7 Power Ratio as a Function of the Flux Ratio Across the 3.7 w/o Enriched Cluster
(Fluxes taken at 2.589 inches across the cell)

exists based on the flux ratios expected from TURBO* design calculations. Therefore, a value of $F_{\phi} = 1.025$ was applied to all power measurements of the test assemblies in order to account for flux gradients.

5.4.2 Calculational Procedure to Convert Power Sharing to Peak Power Densities in the Test Assemblies

The method used to relate the test assembly radial power sharing to the peak power density of the special test assembly fuel rods is presented in this section. This method is slightly different from that used for the standard fuel since the fuel rod lengths and number of fuel rods in the test assemblies differed. The various factors which are used to relate the radial power sharing to the peak power density in the test assemblies are:

F_R = Fuel Assembly Power/Core Average Fuel Assembly Power

$F_{Ru-tube}^*$ = U-tube Power with Test Assembly/Core Average U-tube Power

F_z = Axial Maximum-to-Average Ratio for Test Assembly

F_{split} = Power Produced in Test Assembly/Total U-tube Power

$F_{w/o}$ = Factor which accounts for difference in power sharing within the test assemblies due to fuel rods with different enrichments.

F_{ϕ} = Factor which accounts for difference in fuel rod power sharing within the test assembly due to flux gradient across the assembly. 14

F_q^e = Engineering Factor 21

An adjustment must also be made to account for the difference in fuel lengths of the test assemblies and the standard fuel. This is accomplished by dividing the total length of fuel into the power produced in the assembly. The relationship used to determine the test assembly peak power density is given in equation 5.3 and is used in conjunction with Table 5.4.

$$*F_R = (F_{Ru-tube}) F_{split}$$

TABLE 5.4

TABULATION OF PEAKING FACTORS TO USE WITH TEST ASSEMBLIES

Identification	F_r	$F_{\text{Ru-tube}}$	F_{split}	F_z	$F_{w/o}$	F_ϕ	F_q^e
25.1 kw/ft peak	measured	measured	measured	measured	1.09	1.025	1.045
20.9 kw/ft peak	measured	measured	measured	measured	-	1.025	1.045

$$\text{kw/ft (Peak)} = \left(\frac{\text{Avg. U-Tube Power}}{\text{Length of Test Fuel}} \right) (F_{\text{Ru-tube}}) (F_{\text{split}}) (F_z) (F_{w/o}) (F_\phi) (F_q^e) \quad (5.3)$$

for flux wires replace $(F_{\text{Ru-tube}})(F_{\text{split}})$ with F_R

The method summarized by equation 5.3 was used to determine the peak power densities in the test assemblies from flux wire data.

5.5 Operation History

5.5.1 Cycle 1

CVTR had accumulated 9520 EFPH through the life of Core I when operation was terminated to perform the first refueling, which consisted of refueling the depleted core with four fresh standard 2.0 w/o U-235 assemblies and four high power density test assemblies. Cycle 1 was the period of operation following the first refueling, and was initiated on May 12, 1966. The reactor operated until it was intentionally taken off the line on August 26, 1966, near the end of the available control rod movement. The burnup accumulated during Cycle 1 operation was 2087 EFPH. Reactor power was limited to 92% of full power (Full power is 65 MWth), or less, during Cycle 1 operation due to indications from the U-bend thermocouple that the power

sharing within the high power density test assemblies was excessive (See Section 4.0 of this report). The Cycle 1 power history is presented in Figure 5.8.

5.5.2 Cycle 2

Cycle 2 was the period of operation following the second refueling which consisted of refueling the depleted core with twelve fresh 2.0 w/o U-235 assemblies. Cycle 2 power operation was initiated on October 4, 1966. Two defective HPD test assemblies necessitated two reactor shutdowns to remove the defective assemblies during the cycle. The HPD test assembly irradiated at core position 3C was removed during the period October 27, 1966 through November 12, 1966 and replaced with 1.5 w/o U-235 assemblies. The HPD test assembly irradiated at core position 8Q was removed during the reactor shutdown extending from November 29, 1966 to December 9, 1966, and replaced with 2.0 w/o U-235 assemblies. Three other defective fuel assemblies were located during Cycle 2 operation. These defective fuel assemblies are discussed in detail in Section 5.6 and Section 7.0 of this report. The reactor was shutdown on January 24, 1967 to remove defective fuel assemblies and it was later decided to terminate power operation at that time. The burnup accumulated during Cycle 2 operation was 1600 EFPH. An estimated 938 EFPH additional burnup was available at the termination of Cycle 2. The Cycle 2 power history is presented in Figure 5.9.

5.5.3 Push Power Test

During operation of the second cycle, it was evident from design calculations that after a short period of operation the power density level of the 25.1 kw/ft test assembly would be substantially lower than the test objective. Therefore, a push power test was devised which required banking several control rod groups at the same height. The intent of the test was to cause a substantial increase in the axial flux factor resulting in an increase in the peak power density of the special test fuel.

Figure 5.8 Cycle 1 Power Operations History

Figure 5.9 Cycle 2 Power Operations History

Prior to performing the push power test, a TURBO* calculation was performed to determine the effect of banking control rod groups II, IV and V at the same position, upon the core radial power sharing. The results of the calculation are illustrated in Figure 5.10 and show a substantial increase in power in the 25.1 kw/ft peak test assembly, located at 8Q, when the rods are fully inserted. Based on the results of this calculation and the all rods out case, it was anticipated that the control rods would be banked at approximately 47" for this configuration at a burnup level of 400 EFPD on Cycle 2. The actual banked position was later established to be 45" during the test. The resultant power distribution expected with the control rods banked at 47" was obtained by weighting the TURBO* rods in and rods out distributions to establish a representative case when the rods are banked. This distribution is presented on Figure 5.11 and was derived from TURBO* calculations using the procedure previously presented in Section 2.2.2 to simulate partially withdrawn control rods.

5.6 Test Assembly Failures

Of the four HPD test assemblies inserted in the core during the first refueling, three developed leaks during Cycle 2 operation. Two of the defective assemblies were rated at peak powers of 25.1 kw/ft and were located in core positions 3C and 8Q. The third defective test assembly was rated at 20.9 kw/ft peak and was located in core position 3S.

The defect in the 25.1 kw/ft peak assembly at core position 3C was detected on 10/5/66 soon after the initiation of Cycle 2 operation. The reactor was subsequently shutdown and the test assembly and standard 2.0 w/o assembly in U-tube 3B replaced with fresh 1.5 w/o assemblies. The test assembly was operating at a peak power density of 20.6 kw/ft when the leak was detected.

The defect in the 25.1 kw/ft peak test assembly at core position 8Q was detected on 11/21/66 during the push power test described in Section 7.0 of this report. The peak power density at the time of the failure was

TUBE AND THIMBLE REPRESENTATION OF CVTR FULL CORE

Figure 5.10 Calculated Power Distribution for Cycle 2 With Control Rod Groups II, IV and V Fully Inserted

TUBE AND THIMBLE REPRESENTATION OF CVTR QUARTER CORE

Figure 5.11 Calculated Power Distribution Expected
During Cycle 2 Push Power Test

SUCV-109

FLUX 3

North ↑

11/21/66

Figure 5.12 Measured Radial and Axial Power Distribution
During Push Power Test
When Fuel Failures Were Detected

North ↑

TUBE AND THIMBLE REPRESENTATION OF CVTR QUARTER CORE

$$\% \text{ Diff} = \frac{(TC - TURBO*)}{1/2(TC + TURBO*)}$$

(TURBO* Calculation at 47")

Figure 5.13 Comparison Between Calculated and Measured Radial Power Distributions During the Banked Rod Test

detected and determined to be 21.3 kw/ft. The reactor was subsequently shutdown and the defective test assembly and the standard 2.0 w/o assembly in U-tube 8R were replaced with fresh 2.0 w/o assemblies.

The defect in the 20.9 kw/ft peak test assembly at core position 3S was located on 1/24/67. The assembly was operating at 16.0 kw/ft peak at the time of detection on 1/24/67. It is postulated that this assembly failed during the push power test on 11/21/66. At that time the assembly was operating at 17.6 kw/ft peak.

The defective HPD test assemblies and two 2.0 w/o standard assemblies which failed during Cycle 2 are discussed in detail in Section 7.0 of this report.

During the push power test an increase in the fission product activity level was observed indicative of a fuel leak. Power sharing measurements were currently in progress, thus the temperature rise and primary coolant flow for each pressure tube was measured immediately. In addition, flux wires were inserted into the core at selective positions and reactor power operation was continued until all measurements were acquired. Thus, an accurate measure of the reactor power sharing at the time of failure was secured. The results of this measurement are presented in Figure 5.12 and show that no unusual power condition occurred during the test and no test objectives were exceeded at any time. The axial peaking factors were as expected and the peak values occurred at approximately 26" above the bottom of the fuel.

The measured values were later compared to the predicted power distribution on a pressure tube basis as given on Figure 5.13. This comparison illustrates good agreement considering the approximations made in the calculational procedure. In the calculation it was necessary to extrapolate the fractional power sharing to a rod group worth of 7% $\Delta\rho$. The typical fractional power sharing curve used for CVTR was presented previously on Figure 2.7 and has an upper range of 4%.

6.0 MEASUREMENT PROBLEM AREAS

Throughout operation of the fuel management program on CVTR, several measurement problems were encountered. Remedial action was taken in most cases to resume normal operation when possible. The particular measurement problems were concentrated in the measurement of the relative fuel assembly power sharing due to excessive heating of the U-bend thermocouple in the instrumented pressure tube (8R). In addition, the flux wire system was troublesome due to photomultiplier tube saturation and excessively high radiation levels on the flux wires after irradiation.

The solution to the U-bend thermocouple measurements was to perform auxiliary measurements consisting of decay heat experiments, gamma scans and analytical evaluation of flux wire bottom factors to establish the amount of gamma heating on the thermocouple. An outline of the problem areas encountered and the results of the auxiliary experiments are presented in this section.

6.1 U-Bend Thermocouples

The reactor power was limited, throughout the life of Cycle 1, to less than full power operation due to indicated power sharing of the HPD* Test Assembly in U-tube 8R. Thermal-hydraulic instrumentation involving a U-bend thermocouple indicated that 50-58% of the power produced in U-tube 8R was produced by the test assembly which was positioned in the inlet leg of the U-tube. TURBO* code calculations predicted that 41-42% of the power in the pressure tube would be produced from the test assembly. A number of possible causes for the large discrepancy between predicted and measured power sharing were considered. The most plausible explanation was that gamma heating of the U-bend thermocouple was significantly larger than anticipated. The principal unknown in this case was the intensity of the gamma field in the vicinity of the U-bend thermocouple. A number of experimental and analytical techniques were developed in an attempt to determine the actual power sharing as described on the following pages.

*HPD; abbreviation for High Power Density.

6.1.1 Decay Heat Measurements

The first such experiment was the measurement of the decay heat generation of the two assemblies following the reactor shutdown at the end of Cycle 1. The decay heat generated in each assembly was considered to be directly proportional to the power produced by the assembly. With the reactor shutdown the gamma heating effect on the U-bend thermocouple would be negligible. The thermocouple had previously demonstrated reliability at lower primary coolant temperatures.

The decay heat measurement was conducted by providing flow through the U-tube from the Refueling Machine (RFM) and recording the U-bend temperature and the inlet and outlet temperatures. The inlet and outlet temperatures were measured by inserting chromel-alumel thermocouples in a "Tee" arrangement between the cooling attachments and the hoses coming from the refueling machine. These thermocouples were calibrated and agreed within 0.1°F.

The measurement was conducted for RFM flows of 5 GPM, 3 GPM and 1.3 GPM. With the RFM flow constant, the Component Cooling Water (CCW) flow to the RFM heat exchanger was varied between 14 GPM, 11 GPM, 8 GPM and 5 GPM for each RFM flow mentioned, in order to provide different heat transfer situations between moderator and coolant. Average coolant and moderator temperatures were held nearly equal at 85-95°F to minimize heat exchange between the moderator and the coolant.

In the first series of tests, the RFM flow entered the inlet leg of the U-tube (containing the test assemblies), flowed past the U-bend thermocouple, and then exited from the outlet leg. Following these tests, the cooling attachments were reversed on the handling tool such that the flow through the U-tube was reversed, with the flow entering the outlet leg. A number of additional readings were obtained using the same technique employed for the normal flow test.

The temperature rise in each leg of the U-tube was corrected for heat loss (or gain) to the moderator, and the fraction of heat generated

in the inlet leg fuel assembly was then calculated from a simple ratio of coolant temperature rises.

$$\frac{Q_{\text{test assemblies}}}{Q_{\text{U-tube}}} = \frac{\Delta T_{\text{test assemblies}}}{\Delta T_{\text{U-tube}}} \quad (6.1)$$

In the first series of tests where flow entered the inlet leg of the pressure tube, the results were very consistent at all 3 coolant flow rates as shown in Table 6-1. These data showed a mean value of 41.4% of the heat generated by the test assemblies.

In the reverse flow tests, some difficulty was encountered with the datalogger which printed the coolant temperatures, and it was necessary to repeat these tests on the following day. Subsequent reduction of the data illustrated that steady state was attained in only 3 runs. The remainder of the data showed a definite drift with time, approaching the power sharing values previously obtained in the forward flow tests. Referring to Table 6-1, it can be seen that the second data point taken at a particular value of refueling machine flow is appreciably different than the initial data point under these conditions. Combining all the data, the mean value of percent power from the test assemblies is 41.8%.

The spread of data obtained during the test was in good agreement with the 41-42% power sharing for the test assembly predicted by the TURBO* code. Based on this test and evaluation of other operating data, it was concluded that the TURBO* code calculation gave the best estimate of the power sharing between the legs of the instrumented pressure tube.

6.1.2 Flux Wire Bottom Factor

The second technique developed in the effort to determine the actual power sharing of the test assembly was an analytical method which demonstrated the dependence of the U-bend thermocouple reading on the neutron flux (and hence the gamma flux) at the thermocouple.

TABLE 6.1

SUMMARY OF POWER SHARING EXPERIMENTAL RESULTS

<u>Date</u>	<u>Time</u>	<u>Refueling Machine Flow, gpm</u>	<u>Cooling Water Flow, gpm</u>	<u>Flow Direction</u>	<u>% of U-Tube Heat From Test Assembly</u>
8-29-66	15:21 - 15:25	5.0	14	Normal	40.2
"	15:38 - 15:41	"	11	"	40.5
"	16:21 - 16:25	"	9	"	43.5
"	16:34 - 16:38	3.0	9	"	41.8
"	16:48 - 16:52	"	11	"	42.4
"	17:09 - 17:13	"	14	"	41.3
"	17:30 - 17:34	1.3	14	"	41.1
"	17:49 - 17:53	"	11	"	41.8
"	18:36 - 18:40	"	9	"	40.3
"	21:02 - 21:06	3.0	8	Reverse	34.9*
"	21:14 - 21:18	"	11	"	43.4
8-30-64	11:10 - 11:14	"	14	"	47.0*
"	11:27 - 11:31	"	9	"	40.3
"	12:27 - 12:31	5.0	14	"	56.9*
"	12:44 - 12:48	"	8	"	45.1

*Not at steady state - these points not included in calculation of mean power sharing values

Few group diffusion theory calculations (TURBO* Code) indicated that the radial power sharing in the HPD test assemblies would be relatively insensitive to variations in control rod pattern. However, control rod movement had been observed to have a significant effect on the axial flux (or power) distribution. Generally, when a rod group starts withdrawing from the core, the axial flux distribution curve will be pushed toward the bottom of the core and the resultant distribution will become less pronounced as the rod group is withdrawn. With the control rod groups either fully inserted or fully withdrawn, the axial flux distribution approximates a chopped cosine.

The dependence of the thermocouple reading on the neutron (hence gamma) flux at the thermocouple is demonstrated in Figure 6.1. This figure is a plot of the radial hot channel factor (F_R) for HPD test assembly at 8Q and for the entire U-tube 8R versus time. The HPD test assembly at 8Q was in the flow inlet leg and the F_R was calculated based on the ΔT between an inlet thermocouple and the U-bend thermocouple. (The U-bend thermocouple was corrected for known calibration errors, flow mixing, and expected gamma heating based on Westinghouse calculations. The equation used to calculate the "corrected" thermocouple reading is presented in Reference 20. It is apparent from observation of Figure 6.1 that the F_R at 8Q changed radically on June 9, 1966. On this date, Group IV and Group V rods were interchanged in a standard procedure employed near the end of a control rod group's travel. Group IV rods were partially inserted and Group V rods were withdrawn to 18 inches above the bottom of the fuel. Banking of Group V rods resulted in pushing the axial flux distribution toward the bottom of the core. This, in turn, resulted in a higher gamma flux at the thermocouple which caused the thermocouple reading to increase. Therefore, the indicated radial power sharing (F_R) of the inlet leg (8Q) increased significantly. This indicated F_R continued to vary with the gamma flux at the thermocouple throughout the life of Cycle 1. In contrast to the trend noted above, the radial power sharing (F_R) for the entire U-tube (8R) did not change significantly with control rod movement.

RADIAL HOT CHANNEL FACTOR FOR TUBE 8R AND ASSEMBLY 8Q vs. TIME

Figure 6.1

This is evidenced by Figure 6.1. The radial power sharing (F_R) for the entire U-tube (8R) was calculated based on the ΔT measured by a U-tube inlet and an outlet thermocouple, neither of which was exposed to a significant gamma or neutron flux.

A direct measurement of the gamma flux at the U-bend thermocouple, under various control rod configurations, was not feasible. However, a technique was devised which was considered to give a close approximation of the relative change in the gamma flux at the thermocouple. An axial "Bottom Factor" was obtained from flux wire irradiations made at U-tube 8R during Cycle 1 and Cycle 2 operation. The Bottom Factor is defined as:

$$\text{Bottom Factor} = \frac{\text{Activity of wire at bottom of fuel}}{\text{Average activity over length of fuel}}$$

The Bottom Factor is considered to give an accurate indication of the variation of the neutron flux at the bottom of the fuel since the average activity over the length of the fuel (95.4 inches) did not vary significantly. This, in turn, gives an accurate indication of the gamma flux at the thermocouple which is several inches below the bottom of the fuel and directly below the flux wire guide tube. The Bottom Factors obtained from flux wire traces are tabulated in Table 6.2 with dates and core exposures. Note that factors were obtained on 11/21/66 and 11/28/66 during Cycle 2 operation at a banked control rod configuration. The control rod configuration on 11/21/66 was groups II, IV, and V banked at approximately 45.3 inches. The configuration on 11/28/66 was Group II, IV and V banked at approximately 48 inches. All other groups were out.

The large value of the Bottom Factor on these dates indicates a marked increase in the magnitude of the flux at the bottom of the fuel caused by pushing the power down in the core with the banked rod configuration. The Bottom Factors tabulated in Table 6.2 are plotted versus dates in Figures 6.2. A best visual fit curve is drawn through the data points.

TABLE 6.2

AXIAL BOTTOM FACTORS

<u>Date</u>	<u>Core EFPH</u>	<u>Magnitude of Flux Trace at Bottom of Fuel</u>	<u>Average Value of Flux Trace From 0 to 95.4"</u>	<u>Ratio: Bottom Flux To Average Flux</u>
6/6/66	436	2.66	5.194	.512
6/6/66		2.20	4.424	.497
6/6/66		2.06	4.120	.500
6/10/66	510	2.38	3.778	.630
6/10/66		2.55	4.049	.630
6/10/66		2.75	4.305	.639
6/13/66	572	3.43	5.589	.614
6/13/66		3.24	5.302	.611
6/13/66		2.94	4.668	.630
6/20/66	707	3.04	5.134	.592
6/20/66		2.84	4.863	.584
6/20/66		2.71	4.627	.586
6/24/66	786	3.29	5.514	.597
6/24/66		3.20	5.328	.601
6/24/66		3.01	5.028	.599
6/24/66		2.45	4.178	.586
6/24/66		2.37	3.965	.598
6/24/66		2.25	3.766	.597
6/28/66	867	3.04	4.859	.626
6/28/66		2.54	4.079	.623
6/29/66	884	2.63	4.192	.627
6/29/66		2.39	3.930	.608
6/29/66		2.17	3.559	.610
6/30/66	902	2.16	3.539	.610
7/15/66	1192	2.40	3.821	.628
7/15/66		2.26	3.626	.623
7/15/66		2.14	3.408	.628
7/20/66	1284	2.71	4.337	.625
7/20/66		2.58	4.138	.623
7/20/66		2.48	3.929	.631

TABLE 6.2 (cont'd.)

<u>Date</u>	<u>Core EFPH</u>	<u>Magnitude of Flux Trace at Bottom of Fuel</u>	<u>Average Value of Flux Trace From 0 to 95.4"</u>	<u>Ratio: Bottom Flux To Average Flux</u>
7/25/66	1393	2.46	4.008	.614
7/25/66		2.32	3.787	.613
7/25/66		2.23	3.604	.624
8/03/66	1590	2.60	4.192	.620
8/03/66		2.40	4.024	.596
8/03/66		2.27	3.813	.595
8/10/66	1744	1.88	3.242	.580
8/17/66	1898	1.78	3.132	.568
8/17/66		1.72	3.011	.571
8/17/66		1.63	2.894	.563
(Cycle 1 = 2100 EFPH)				
11/21/66	Cycle 2 = 535			.904
11/21/66				.908
11/28/66	Cycle 2 = 632			.881
11/28/66				.881

Figure 6.2

Comparison of Figure 6.2 and the 8Q trace in Figure 6.1 indicates that the radial power sharing (F_R) at 8Q, as measured by the U-bend thermocouple, follows the same trend as the Bottom Factor curve. This implies that the radial power sharing (F_R) (by the U-bend thermocouple method) is dependent on the gamma flux at the thermocouple. Since the radial power sharing (F_R) is dependent on the gamma flux at the thermocouple, it follows that the apparent power split between the two legs of the U-tube would also be dependent on the gamma flux. Thus, a correlation was drawn between the apparent power split for leg 8Q (as indicated by the U-bend thermocouple) and the Bottom Factor. This correlation is presented in Figure 6.3. The data required to construct Figure 6.3 is tabulated in Table 6.3. The apparent power splits were obtained from Figure 6.1 in the following manner:

$$\text{Power Split at 8Q} = \frac{F_R^{8Q}}{2 \times F_R^{8R}}$$

The points at the far right in Figure 6.3 were obtained during the banked rod configurations mentioned earlier. At that time the flux distribution was pushed toward the bottom of the core and the Bottom Factor was high which caused an increase in the apparent power split. The point on the far left of the figure was obtained near the end of Group IV rod movement when the axial flux distribution approximated a cosine fit. The remaining points were obtained during Group V rod movement.

Without a gamma heating effect on the U-bend thermocouple, the slope of the line in Figure 6.3 would be zero. Conversely, if the line is linearly extrapolated to the point where the Bottom Factor is zero (i.e., the point where there is no gamma heating effect on the thermocouple) the power split determined at that point is approximately 40%. The TURBO* code predicted the HPD test assembly (8Q) power split would be 41-42%.

CORRELATION BETWEEN APPARENT POWER SPLIT IN U-TUBE 8R AND BOTTOM FACTOR

Figure 6.3

TABLE 6.3
CORRELATION DATA

<u>Date</u>	<u>Average¹ Bottom Factor</u>	<u>Apparent² Power Split</u>
6/06/66	.5030	.503
6/10/66	.6330	.532
6/13/66	.6183	.523
6/20/66	.5873	.510
6/24/66	.5963	.512
6/28/66	.6245	.515
6/29/66	.6150	.520
6/30/66	.6100	.517
7/15/66	.6263	.526
7/20/66	.6263	.533
7/25/66	.6170	.519
8/03/66	.6037	.519
8/10/66	.5800	.507
8/17/66	.5673	.502
11/21/66	.9060	.570
11/28/66	.8810	.587

1. Extrapolated from Table 6.1.
2. Extrapolated from Figure 6.1.

6.1.3 Gamma Scans

On site gamma scans of two of the defective high power density (HPD) test assemblies and the standard 2.0 w/o assemblies in the same U-tubes were conducted in December 1966. The fuel assemblies scanned were irradiated in core locations 8R and 3B. In addition to the December 1966 scan, a second scan was conducted on the assemblies from U-tube 8R in January 1967. The primary purpose of these scans was to determine if excessive reflector power peaking had occurred at the top of the HPD test assemblies. (The HPD assemblies are shorter than the standard assemblies).

An apparatus was constructed to scan the fuel assemblies while in the failed fuel encapsulating tubes. The current output from a miniature ionization chamber shielded by approximately one and one half (1.5) inches of lead and located approximately 3.75 inches from the center of the fuel assembly, was recorded at one inch increments along the length of each fuel assembly. The encapsulating device was then rotated 180° and the fuel assembly was scanned a second time. The results of the December 1966 scans are depicted in Figures 6.4 and 6.5 for HPD test assembly group #3 from U-tube leg 3C and the standard 2.0 w/o fuel from U-tube leg 3A, respectively. Figures 6.6 and 6.7 depict the scans for HPD test assembly group #1 from U-tube leg 8Q and the standard 2.0 w/o fuel from U-tube leg 8S, respectively. The results of the January 1967 scans are depicted in Figures 6.8 and 6.9 for the HPD test assembly group #1 from U-tube leg 8Q and the standard 2.0 w/o fuel assembly from U-tube leg 8S, respectively.

Since the exact location of the bottom of the fuel assembly, relative to the ion chamber, was not known, a reference scale was used to indicate the location of the individual axial measurements. The location of the bottom of the fuel was then determined to be a point half way between "background" response on the ion chamber and a significant response on the chamber.

Figure 6.4 Gamma Profile of HPD Test Assembly Group #3 Irradiated in Core Location 3C

Figure 6.5 Gamma Profile of the Standard 2.0 w/o Assembly Irradiated in Core Location 3A

Figure 6.6 Gamma Profile of HPD Test Assembly Group #1 Irradiated in Core Location 8Q

Figure 6.7 Gamma Profile of the Standard 2.0 w/o Assembly Irradiated in Core Location 8S

Figure 6.8 Repeat Gamma Profile of HPD Test Assembly Group #1 Taken After One Month Decay

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Figure 6.9 Repeat Gamma Profile of Standard 2.0 w/o Assembly at 8S Taken After One Month Decay

TABLE 6.4

GAMMA SCAN MEASUREMENTS

<u>U-tube Leg</u>	<u>Assembly</u>	<u>Assembly Burnup (MWD/MTU)</u>	<u>Date of Reactor Shutdown Prior to Removal</u>	<u>Date of Removal</u>	<u>Date/Time of Scan</u>
					12/14/66
3A	Standard 2.0 w/o	2525	10/27/66	11/6/66	1430
3C	HPD Group 3	6360	10/27/66	11/6/66	0900
					12/13/66
8Q	HPD Group 1	7060	11/29/66	12/5/66	1130
8S	Standard 2.0 w/o	2804	11/29/66	12/5/66	1400
					1/25/67
8Q	HPD Group 1	7060	11/29/66	12/5/66	1600
8S	Standard 2.0 w/o	2804	11/29/66	12/5/66	1800

TABLE 6.5

EVALUATION OF GAMMA PROFILES

<u>U-tube</u>	<u>Assembly</u>	<u>Area (cm²)</u>	<u>Length of Fuel (in.)</u>	<u>Axial Maximum/Average</u>
3A	Standard 2.0 w/o	108.00	95.4 in.	1.276
3C	HPD Group 3	97.03	79.5 in.	1.180
8Q	HPD Group 1	122.52	79.3 in.	1.155
8S	Standard 2.0 w/o	151.42	95.4 in.	1.244
8Q	HPD Group 1	125.81	79.3 in.	1.162
8S	Standard 2.0 w/o	161.80	95.4 in.	1.245

} December
Data

} January
Data

The four assemblies scanned had been irradiated since the beginning of Cycle 1 life. Table 6.4 gives information concerning assembly burnup and the dates of shutdown, removal and scanning of the assemblies.

The scans conducted on the two HPD Test assemblies revealed that no reflector power peaking existed at the top of the fuel assembly as previously postulated. The gamma profiles obtained for the standard assemblies are similar to flux wire traces obtained at those positions and show no unusual axial power peaks.

Each of the averaged profiles was integrated by a compensating polar planimeter in order to obtain an axial maximum to average value for each HPD test assembly and each standard assembly. This information is tabulated in Table 6.5. The axial maximum to average factors obtained by this method are in generally good agreement with flux wire data obtained during Cycle 1 operation.

No attempt was made to determine the power sharing between the HPD test assemblies and standard 2.0 w/o assemblies in the same U-tube by comparison of the areas under the gamma profile. Since the geometries of the two assemblies are different (the test assembly has 6 UO_2 rods and 13 Al_2O_3 rods; the standard assembly has 19 UO_2 rods), it was felt that a comparison of the integrals to obtain assembly power sharing could not be accomplished without further calculations to determine the effects of the different geometries.

6.2 Flux Wire Measurements

Several problems were encountered which reduced the availability of the flux wire system for use on the CVTR during the fuel management program. These problems consisted of discrepancies in the manganese content of the flux wires during Cycle 1 and an increased manganese concentration in the replacement wires causing excessive wait periods between irradiation and counting during Cycle 2.

6.2.1 Photomultiplier Tube Saturation

The photomultiplier tube was replaced prior to Cycle 1 startup. Flux wire measurements obtained with the new tube had shown effects of saturation for those measurements performed during Cycle 1 startup. Therefore, several measurements were performed with the new photomultiplier tube in an effort to determine the effects of tube voltage on tube linearity. In addition, measurements were performed to determine the permissible activation level on the wires for counting purposes. The results of these measurements are illustrated in Figure 6.10 and demonstrate that the maximum tube voltage is of the order of 550 volts based upon the linearity characteristics shown in the figure. The results given in the figure show the integrated saturation activity plotted as a function of time for various tube voltages. Assuming the manganese wire decay constants are correct, this curve should be a straight line. The figure demonstrates that voltages greater than 600 volts resulted in deviation from the straight line, indicative of non-linearity. In addition, the figure indicates that saturation effects exist on the tube if the irradiated wires are counted within approximately 3 hours after a normal 10 minute irradiation. Based on the measurements performed, it was concluded that the best tube biasing voltage was of the order of 550 volts and a wait time of 3 hours after irradiation should occur before counting.

6.2.2 Flux Wire Manganese Concentration Discrepancies

During the interim period for the fuel replacement operation between Cycle 1 and Cycle 2, new flux wires were installed for use during Cycle 2 operation. These new wires were later calibrated to determine consistency of relative manganese concentration. The old wires were replaced due to discrepancies which required a 14% and 16% correction on two of the wires when used to measure relative power sharing.

(a) Photomultiplier Tube Saturation

(b) Photomultiplier Tube Linearity as a Function of Tube Voltage

Figure 6.10 Flux Wire Photomultiplier Tube Voltage and Saturation Effects

Calibration measurements were performed with the new set of wires by inserting each wire into thimble 6J while maintaining the reactor at a constant power level. The results of the measurements were subject to question, however, due to photomultiplier tube saturation effects requiring long wait periods before counting. The results did indicate that the relative manganese content in each wire was the same within the counting system accuracy. The results also indicated that the manganese concentration of the new set of wires although consistent on a relative basis, was significantly higher on an absolute basis than that of the previous wires. The effect of the higher concentration required excessively long wait periods between irradiation and counting, of the order of 9 hours. Efforts were made to decrease the wait period required by reducing the collimator size. This helped the situation but was not the solution. It appeared that a more concentrated effort was required to solve the problem and the most obvious solution would have been replacing the wires with another set having a lower manganese concentration content. A reduction in the irradiation time of 10 minutes was not feasible because a ten minute irradiation period was required in order to maintain the irradiation received during wire transit through the active core, negligible.

7.0 FAILED FUEL OCCURRENCES

The high power density test assemblies discussed in this section and in Section 5.0 of this report were developed under a separate Post Construction Research and Development task. However, the fuel management program was so closely coordinated with the HPD test assembly program that a description of the test assembly failures (and the standard assembly failures) is provided in this section in the interest of completeness. The material presented in this section is discussed in more detail in Reference 20.

7.1 History of Fuel Failures

A total of five leaking fuel assemblies was detected during Cycle 2 operation. No defective elements were experienced prior to Cycle 2. In addition to the three defective HPD test assemblies discussed in Section 5.6 of this report, two 2.0 w/o standard fuel assemblies were also found to be defective. These assemblies were located in pressure tube legs 6I and 7K. Thus, the assemblies which failed were located in pressure tube legs 3C, 8Q, 6I, 7K and 3S. A chronological log of the failures and events surrounding the failures is presented in this section.

The first failure occurred in the HPD test assembly located at core position 3C. The failure was detected on October 5, 1966, the day following the return to power subsequent to the second refueling operation. Reactor power at the time was 90% of full power (full power is 65 MWth). The test assembly in location 3C rated at 25.1 kw/ft peak, was operating at approximately 20.6 kw/ft at the time the defect was detected. The exact location of the defective assembly was not known until several weeks after the defect became apparent. For this reason, the reactor was not shutdown until October 27, 1966, for the removal of the test assembly at core position 3C. The defective test assembly and the associated 2.0 w/o standard assembly in pressure tube 3B were replaced by 1.5 w/o standard fuel assemblies. Following the removal of 3C, the reactor returned to power operation on November 12, 1966.

The second failure occurred in the HPD test assembly located at core position 8Q during a push power test (see Section 5.5.3) on November 21, 1966. This test assembly was also rated at 25.1 kw/ft peak and was operating at approximately 21.3 kw/ft during the push power test. After ascertaining that the defective assembly was located in U-tube 8R, the reactor was again shutdown on November 29, 1966, for removal of the HPD test assembly and the associated standard 2.0 w/o assembly in the same U-tube. These assemblies were replaced with fresh 2.0 w/o standard assemblies and the reactor returned to power on December 9, 1966.

The third failure was detected in the 2.0 w/o standard assembly at core position 6I on December 29, 1966. Following the return to power after replacing U-tube 8R, the primary coolant activity failed to decrease as it had following the removal of the first defective assembly. The defective assembly was determined to be in U-tube 6J and later tests indicated the failed assembly was in leg 6I of the U-tube. Since no abnormal power distributions (radial or axial) existed in the reactor following the removal of the second failed assembly, it was postulated that the 2.0 w/o standard assembly at core position 6I failed during the push power test of November 21, 1966. The assembly at 6I was operating at a peak power density of 13.6 kw/ft during the push power test. Since U-tube 6J did not contain HPD test assemblies and primary coolant activity levels were well within allowable limits and radiation levels in the plant were not excessive, reactor power operation was continued.

The fourth failure was detected in the 2.0 w/o standard assembly at core position 7K during the period January 13-16, 1967. During this period each U-tube was checked for leaks and U-tube 7J was found to be leaking in addition to U-tube 6J. Later tests indicated that the defective assembly was located in leg 7K of the U-tube. It is postulated that assembly 7K failed during the push power test on November 21, 1966. At that time the assembly was operating at a peak power density of 12.3 kw/ft, well within the design value of 17 kw/ft. Since radioactivity levels were still within limits, the reactor was returned to power.

The fifth failure was detected in the HPD test assembly at core location 3S on January 24, 1967. On this date the primary coolant activity began to increase and continued to increase until the reactor was shutdown for non-nuclear repairs. The leak was located in U-tube 3R during tests conducted the following day. Subsequent tests after the removal of the U-tube indicated that the defective assembly was the HPD test assembly at core position 3S. The test assembly was rated at 20.9 kw/ft peak power density. At the time of the primary coolant activity increase, the assembly was operating at a peak power density of 16.0 kw/ft. However, it is possible that the test assembly at 3S failed during the push power test on November 21, 1966 and was not discovered until later. However, tests during December and early January had shown the assembly not to be leaking. The test assembly was operating at 17.6 kw/ft peak power density during the push power test.

The reactor was not returned to power following the discovery of the test assembly failure at 3S and preparations were made to remove this assembly and the remaining non-defective test assembly at core position 8A. A subsequent decision was made to terminate power operation of the CVTR at this point.

A history of CVTR defective fuel assemblies is presented in Table 7.1.

7.2 Detailed Power History at Time of Failure

The thermal-hydraulic instrumentation of the CVTR facilitates an accurate determination of power levels, not only for the entire core, but also for individual pressure tubes. The power produced in a particular leg of a U-tube is obtained by applying the power sharing factor predicted by the TURBO* code calculation. A long period of good agreement between measured radial factors and TURBO* predicted radial factors has provided confidence in the TURBO* predicted power sharing factors. The ability to accurately determine pressure tube leg power allowed the frequent determination of peak power densities in the HPD test assemblies by applying appropriate nuclear hot channel factors to the leg power. Axial hot channel factors, F_z , employed in the determination of peak power density histories, are "best estimate" values based on flux wire irradiations and established

TABLE 7.1
TABULATION OF FUEL LEAKS DETECTED DURING CYCLE 2 OPERATION

Failure Number	Assembly Identification	Core Location	Date of Failure	Date of Location	Date of Shut-down Prior to Removal	Date of Removal	Equivalent Exposure* Hours at Time of Failure		Peak Exposure** at Time of Failure (MWD/MTU)	Peak Power Density at Time of Failure (kw/ft)
							EFPH	MWD/MTU		
1	HPD Test Assembly Group #3	3C	10/5/66	10/27/66	10/27/66	11/6/66	1,577	6,010	8,201	20.6
2	HPD Test Assembly Group #1	8Q	11/21/66	11/29/66	11/29/66	12/5/66	2,036	7,759	10,588	21.2
3	Standard 2.0 w/o	6I	11/21/66	12/29/66	1/24/67	2/23/67	11,844	11,714	19,127	13.6
4	Standard 2.0 w/o	7K	11/21/66	1/13 - 1/16/67	1/24/67	2/24/67	11,681	11,552	18,862	12.3
5	HPD Test Assembly Group #4	3S	11/21/66	1/24/67	1/24/67	2/25/67	1,968	7,500	9,317	17.6

*Equivalent Exposure hours are the core average EFPH multiplied by the average radial factor ($\overline{F_R}$) for the entire irradiation period.

**Peak Exposures are determined in the following manner for standard and test assemblies:

(1) Standard: Peak Burnup = Equivalent Exposure Hours $\times F_Z \times G_O \times F_E$

(2) 25.1 kw/ft: Peak Burnup = Equivalent Exposure Hours $\times F_Z \times 19/6 \times F_{W/O} \times F_E \times F_{\phi} \times 95.4/78.4$

(3) 20.9 kw/ft: Peak Burnup = Equivalent Exposure Hours $\times F_Z \times 19/6 \times F_E \times F_{\phi} \times 95.4/78.4$

where: $F_E = 1.045$ $F_{W/O} = 1.098$

$G_O = 1.25$ $F_{\phi} = 1.025$

trends. In certain cases, the axial hot channel factor employed was determined at the time of the thermal-hydraulic measurement used to establish peak power densities. Past experience with CVTR has demonstrated that during the fuel management program the axial factors at the test assembly locations were rather insensitive to rod motion, by design.

The history of peak power density for the 25.1 kw/ft HPD test assemblies at core positions 8Q and 3C is shown in Figure 7.1 for the period preceding and up to the time of the first failure, which was the assembly at 3C on October 5, 1966. The axial hot channel factor, F_z , used to determine the peak power density at the time of failure were obtained following the failure but at a similar control rod configuration.

The history of peak power density for the HPD test assembly located at core position 8Q is presented in Figure 7.2 for the period preceding and following the failure of the assembly at 8Q during the push power test (see Section 5.5.3 of this report) on November 21, 1966. The axial hot channel factor, F_z , used to determine the peak power density at the time of failure was determined from flux wires irradiated during the failure.

The peak power densities based upon thermal-hydraulic instrumentation in the defective assemblies at the time of failure is tabulated in Table 7.2. Similar results based upon flux wire irradiation measurements for the standard fuel and the special test assemblies are presented in Figures 7.3, 7.4 and 7.5.

TABLE 7.2
PEAK POWER DENSITIES AT TIME OF FAILURE

Failure Number	Assembly Identification	Core Location	F_z at Failure	Tube Power (kw)	Power Split (%)	Peak Power Density (kw/ft)
1	25.1 kw/ft Test Assembly	3C	1.17 (est.)	1420	40	20.6
2	25.1 kw/ft Test Assembly	8Q	1.36	1250	40	21.3
3	2.0 w/o Standard Assembly	6I	1.70 (est.)	1790	50	13.6
4	2.0 w/o Standard Assembly	7K	1.70 (est.)	1615	50	12.3
5	20.9 kw/ft Test Assembly	3S	1.36 (est.)	1190	38	17.6

Figure 7.1 Detailed History of Peak Power Densities in Special Test Assemblies During Failure 1

Figure 7.2 Detailed History of Peak Power Densities in Special Test Assembly During Failure 2

Figure 7.3 Power Density Versus Burnup for Standard 2.0 w/o Assembly in Location 6I

Figure 7.4 Power Density Versus Burnup for Standard 2.0 w/o Assembly in Location 6K

Figure 7.5 Maximum and Average Power Densities of Special Test Assemblies Containing Fuel Leaks

7.3 Failed Fuel Detection

The CVTR Failed Fuel Element Detection and Location (FEDAL) System consisted of a general primary coolant monitor and two U-tube monitors capable of monitoring each individual U-tube in the core, in sequence. The FEDAL system design and the operational procedure are described in References 21 and 22.

Attempts to locate the first defective assembly indicated that the FEDAL system did not perform according to design, due to the transfer time involved in cycling a U-tube sample through the common header to the FEDAL system detector. The cycling time required was on the order of seven minutes, and, since the primary coolant cycled in less than one-half minute, a significant mixing occurred which tended to give successively higher activity readings for each sample point. This differed from the design concept which expected the defective U-tube sample activity to be consistently higher than the other sample points.

For this reason, it was necessary to develop an operational technique which would locate the defective assemblies using the existing equipment. In the technique developed, the two U-tube detectors were used to monitor two U-tubes while the defective assembly was forced to release a short burst (app. 5 minutes) of activity. The burst of activity was produced by increasing the power level approximately 10% for a short period (~ 5 minutes) and then returning to the original power level. If the reactor was in a shutdown condition, the burst could be produced by lowering the primary coolant pressure for the desired length of time and then returning to the required pressure. The burst of activity released into the system would be recorded on the U-tube FEDAL detectors. The traces obtained would have one of two general shapes. The trace from the detector monitoring a pressure tube with a defective fuel assembly would be similar to that depicted in Figures 7.6 and 7.7 for pressure tubes 6J and 7J, respectively. The trace from the detector monitoring pressure tube with a non-defective fuel assembly would be similar to that depicted in Figures 7.6 and 7.7 for pressure tubes 5J and 4J, respectively. Note that the traces of the defective assemblies (Figure 7.6) increase sharply, peak, and then level out as flow mixing of the primary coolant reduces the activity in that pressure tube.

PAIRED RELATIVE ACTIVITY TRACES OF 6J AND 5J, 12/28/66

Figure 7.6

PAIRED RELATIVE ACTIVITY TRACES OF 7J AND 4J, 12/28/66

Figure 7.7

The traces of the non-defective assemblies (5J and 4J) increase gradually as flow mixing carries activity to that pressure tube. It was desirable to monitor one defective and one non-defective assembly during each run as depicted on the figures, in order to compare the traces obtained. This procedure was continued until each pressure tube had been checked for defective assemblies.

The defective assembly was located by sampling the refueling machine cooling water activities before and after each assembly was removed from the defective pressure tube. Higher activities were obtained from the defective assemblies. In addition, the activities of the assemblies were checked periodically after encapsulation in the spent fuel pit.

7.4 Special Consideration of Unusual Occurrences

The purpose of this section is to provide consideration of any anomalies occurring at the time of failure and also to present comments of interest regarding the fuel failures.

1. Observation of Figure 7.1 illustrates that when Failure #1 occurred in location 3C, the test assembly at location 8Q was operating at a higher power level. Also, when Failure #2 occurred in 8Q, it was operating at a slightly lower power level than it had previously operated at the time 3C failed.
2. The failures in the standard fuel occurred in the south half of the core which had been operating at power densities that were typically 2% higher than the north half.
3. HPD test assembly Group #3, which was located at core position 3C and developed the first leak, contained four fuel rods having fuel densities of 96% of the theoretical density, and two rods having fuel densities of 93% of the theoretical density. HPD test assembly group #1, located at core position 8Q, was the second test assembly to fail and was also rated at 25.1 kw/ft, peak power density. Group #1 test assembly contained rods with fuel densities of 93% of the theoretical density. The 20.9 kw/ft, peak, test assemblies contained rods with fuel densities of 93% of the theoretical density.

TABLE 7.3

PEAK POWER DENSITIES AT 25 INCHES ABOVE BOTTOM OF FUEL

Date of Flux Wire Irrad.	Burnup (EFPH)		U-tube Leg	Maximum F_z			Axial at 25" (F_z)	Power Density (kw/ft)		Remarks
	Cycle 1	Cycle 2		95.4"	78.4"	Height		At Max. F_z	At 25" Abv. Btm.	
5/18/66	90		6I	1.196		46"	1.107	11.34	10.50	Irradiation made during Push Power Test
6/8/66	481		6I	1.215		55"	1.026	11.37	9.60	
7/14/66	1174		6I	1.194		38"	1.165	10.24	9.99	
8/2/66	1568		6I	1.210		43"	1.120	11.33	10.48	
11/17/66		456	6I	1.127		45"	1.093	8.88	8.61	
11/21/66		535	6I	1.700		25"	1.700	13.60	13.60	
5/18/66	85		8Q		1.160	46"	1.063	16.94	15.53	Irradiation made during Push Power Test
6/9/66	497		8Q		1.179	52"	1.008	16.98	14.52	
7/15/66	1192		8Q		1.151	40"	1.111	16.27	15.70	
8/03/66	1590		8Q		1.174	45"	1.074	19.08	17.45	
11/21/66		535	8Q		1.360	25"	1.360	21.22	21.22	

4. The peak power densities obtained during the push power test and reported in Section 7.2 of this report, occurred at an axial position 25 inches above the bottom of the fuel assembly. This peak power density occurred lower in the core than during normal operation, due to the banked control rods, and was considerably higher than previously experienced at the 25 inch axial position. A history of power densities experienced at 25 inches above the bottom of the fuel in U-tube legs 8Q and 6I is presented in Table 7.3.
5. The two standard 2.0 w/o fuel assemblies which failed had achieved a lower exposure than the standard assemblies in the same U-tube. The defective assembly located at core position 6I had received an exposure of 11,714 MWD/MTU. The assembly located in the other leg of the U-tube (6K) had received an equivalent exposure of 12,407 MWD/MTU. The defective assembly located at core position 7K had received an equivalent exposure of 11,552 MWD/MTU. The assembly located in the other leg of the U-tube (7I) had received an equivalent exposure of 12,295 MWD/MTU.

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APPENDIX I

TABULATION OF CALCULATED POWER DISTRIBUTIONS

TABLE I.1

TABULATION OF TURBO* CALCULATED LOCAL-TO-AVERAGE RADIAL POWER DISTRIBUTIONS FOR CYCLE 1

Cycle 1 Burnup	0 EFPH	0 EFPH	0 EFPH	100 EFPH	270 EFPH	970 EFPH	1570 EFPH
Control Rods (Fully Inserted)	II, IV V, VII	V	IV, V	IV, V	V	None	None
Eigenvalue (λ)*	0.9034	1.0001	0.9748	0.9677	0.9892	1.0003	0.9880
(Fuel Assembly) Locations	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)
2E	0.897	1.022	0.906	0.901	1.022	0.914	0.919
2G	0.775	1.076	0.748	0.753	1.092	0.950	0.955
2I	0.778	1.135	0.728	0.739	1.159	0.990	0.994
3A	1.253	1.076	1.154	1.099	1.024	1.026	1.034
3C	0.851	0.927	0.968	0.931	0.888	0.855	0.849
3E	0.888	1.084	1.077	1.075	1.084	1.016	1.019
3G	1.005	1.147	1.067	1.079	1.161	1.045	1.047
3I	1.038	1.205	1.085	1.102	1.225	1.074	1.075
4A	1.077	0.809	0.907	0.895	0.798	0.876	0.879
4C	1.121	0.928	1.024	1.019	0.923	1.026	1.027
4E	0.942	0.886	0.953	0.961	0.891	0.936	0.936
4G	0.827	0.963	1.019	1.032	0.973	0.938	0.938
4I	1.170	1.380	1.450	1.465	1.386	1.262	1.256
5A	1.050	0.720	0.827	0.826	0.718	0.881	0.882
5C	0.915	0.649	0.743	0.748	0.654	1.012	1.010
5E	0.824	0.678	0.760	0.771	0.686	0.932	0.929
5G	1.008	0.892	0.996	1.009	0.900	0.933	0.931
5I	1.581	1.425	1.589	1.594	1.415	1.333	1.322

$$*\lambda_{(J.C.)} = 0.9884$$

TABLE I.2

TABULATION OF TURBO* CALCULATED LOCAL-TO-AVERAGE RADIAL POWER DISTRIBUTIONS
FOR CYCLE 2 (8 FUEL ASSEMBLY REPLACEMENT)

Cycle 2 Burnup	0 EFPH	100 EFPH	200 EFPH	900 EFPH	1450 EFPH
Cycle 1 Burnup	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH
Control Rods (Fully Inserted)	IV, V	IV, V	V	None	None
Eigenvalue (λ)*	0.9752	0.9685	0.9866	1.0045	0.9937
(Fuel Assembly) Locations	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)
2E	0.799	0.808	0.916	0.808	0.814
2G	0.640	0.653	0.948	0.812	0.819
2I	0.602	0.620	0.985	0.829	0.836
3A	1.153	1.130	1.060	1.041	1.047
3C	0.901	0.889	0.853	0.809	0.802
3E	0.988	0.994	1.008	0.935	0.938
3G	0.930	0.948	1.030	0.917	0.921
3I	0.911	0.937	1.058	0.916	0.919
4A	1.310	1.237	1.114	1.189	1.193
4C	1.384	1.231	1.207	1.317	1.321
4E	0.929	0.931	0.876	0.912	0.912
4G	0.926	0.944	0.906	0.868	0.868
4I	1.237	1.270	1.232	1.112	1.109
5A	0.929	0.903	0.799	0.940	0.938
5C	0.821	0.806	0.717	1.064	1.059
5E	0.930	0.938	0.850	1.143	1.136
5G	1.228	1.253	1.143	1.171	1.164
5I	1.380	1.417	1.299	1.214	1.205

$$*\lambda_{(J.C.)} = 0.9884$$

TABLE I.3

TABULATION OF TURBO* CALCULATED LOCAL-TO-AVERAGE RADIAL POWER DISTRIBUTIONS
FOR MODIFIED CYCLE 2 (12 FUEL ASSEMBLY REPLACEMENT)

Modified Cycle 2 Burnup	100 EFPH	0 EFPH	100 EFPH	400 EFPH	1400 EFPH	2100 EFPH	2400 EFPH
Cycle 1 Burnup	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH
Control Rods (Fully Inserted)	II, IV V, VII	II, V	II, V	II	None	None	None
Eigenvalue (λ)*	0.9141	0.9757	0.9665	0.9918	1.009	0.9950	0.9892
(Fuel Assembly) Locations	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)
2E	0.774	0.982	0.994	0.868	0.808	0.812	0.814
2G	0.675	1.016	1.029	0.872	0.847	0.850	0.852
2I	0.701	1.075	1.083	0.900	0.899	0.901	0.902
3A	1.157	1.168	1.162	1.122	0.969	0.980	0.982
3C	0.746	0.923	0.920	0.870	0.756	0.750	0.746
3E	0.795	1.031	1.040	0.960	0.922	0.926	0.927
3G	0.905	1.009	1.018	0.901	0.959	0.959	0.959
3I	1.262	1.409	1.377	1.175	1.332	1.331	1.332
4A	1.372	1.270	1.218	1.273	1.088	1.099	1.102
4C	1.373	1.322	1.277	1.374	1.225	1.235	1.237
4E	0.893	0.834	0.839	0.887	0.879	0.880	0.881
4G	0.762	0.678	0.689	0.681	0.885	0.881	0.881
4I	1.037	0.891	0.905	0.826	1.180	1.166	1.163
5A	1.052	0.880	0.870	1.001	0.849	0.855	0.857
5C	0.919	0.759	0.756	1.098	0.969	0.972	0.972
5E	0.977	0.775	0.788	1.109	1.073	1.070	1.068
5G	1.232	0.948	0.972	1.049	1.144	1.133	1.130
5I	1.368	1.029	1.060	1.035	1.216	1.199	1.195

$$*\lambda_{(J.C.)} = 0.9884$$

TABLE I.4

TABULATION OF TURBO* CALCULATED LOCAL-TO-AVERAGE RADIAL POWER DISTRIBUTIONS FOR CYCLE 3

Cycle 3 Burnup	0 EFPH	0 EFPH	100 EFPH	800 EFPH	800 EFPH	1400 EFPH
Modified Cycle 2 Burnup	1640 EFPH	1640 EFPH	1640 EFPH	1640 EFPH	1640 EFPH	1640 EFPH
Cycle 1 Burnup	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH	1570 EFPH
Control Rods (Fully Inserted)	None	II, V	V	V	None	None
Eigenvalue (λ)*	1.0186	0.9575	0.9870	0.9739	1.0005	0.9894
(Fuel Assembly) Locations	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)	(F_R)
6A	0.823	0.833	0.697	0.710	0.836	0.835
6C	0.924	0.713	0.626	0.639	0.948	0.942
6E	1.006	0.720	0.760	0.763	1.029	1.022
6G	1.061	0.877	1.038	1.028	1.072	1.064
6I	1.119	0.947	1.191	1.170	1.122	1.112
7A	1.082	1.206	0.999	1.016	1.083	1.086
7C	1.192	1.240	1.085	1.099	1.198	1.200
7E	0.843	0.799	0.811	0.813	0.850	0.850
7G	0.840	0.655	0.881	0.875	0.846	0.845
7I	1.117	0.851	1.232	1.212	1.116	1.110
8A	1.071	1.283	1.028	1.040	1.023	1.030
8C	1.147	1.351	1.130	1.141	1.103	1.111
8E	0.899	1.008	0.955	0.956	0.894	0.897
8G	0.933	0.987	1.041	1.034	0.933	0.934
8I	1.298	1.344	1.494	1.481	1.303	1.304
9E	0.812	0.990	0.900	0.900	0.805	0.810
9G	0.852	1.027	0.980	0.975	0.851	0.855
9I	0.979	1.169	1.150	1.148	0.987	0.994

$$\lambda_{(J.C.)} = 0.9884$$

APPENDIX II

TABULATION OF CALCULATED ACTIVITIES AND NUCLEAR FACTORS

All nuclear factors required for input to the CAVIAR code were established for fuel Cycles 1 and 2 from the TURBO* design calculations reported in the previous sections. These factors were obtained for all anticipated control rod configurations expected during fuel cycling of the first two cycles. The nuclear factor is the calculated ratio of the power produced in a fuel assembly to the manganese activation at the flux wire location for the particular pressure tube in question. These factors are used in the CAVIAR code to relate the measured flux wire activation to the relative power produced in the fuel assemblies.

This technique was used successfully throughout Core I operation and resulted in substantial agreement between design and measurements¹³. The validity of the technique was originally demonstrated by direct comparison with critical experiments performed at the Westinghouse Reactor Evaluation Center in conjunction with the Pre-Construction Research and Development Program*.

The particular nuclear factors generated in conjunction with the fuel management program are tabulated in detail. All values were obtained directly from TURBO* calculations and are related to the quadrants which contained the 25.1 kw/ft peak test assemblies, the North-West and South-East quadrants. The nuclear factors for the pressure tubes which contained the 20.9 kw/ft peak test assemblies are tabulated separately and were obtained from hand calculations which scaled the relative power in those assemblies by accounting for both enrichment and geometric differences.

*References 15, 16, 17, and 18.

TABLE II.1

TABULATION OF CALCULATED ACTIVITIES AND NUCLEAR FACTORS FOR CYCLE 1

Tube Location	Rod Groups II, IV, V, VII Inserted, Burnup = 0 EFPH		Rod Group V Inserted Burnup = 0 EFPH		Rod Group IV, V Inserted Burnup = 0 EFPH	
	Activation	Nuclear Factor	Activation	Nuclear Factor	Activation	Nuclear Factor
2E		156.1		139.6		149.9
2F	0.2487		0.3166		0.2613	
2G		134.7		147.0		123.8
2I		136.2		141.0		127.7
2J	0.2470		0.3482		0.2466	
3A		168.7		145.7		146.6
3B	0.3214		0.3196		0.3405	
3C		114.6		125.4		123.0
3E		135.6		139.4		142.9
3F	0.2833		0.3364		0.3259	
3G		153.4		147.5		141.7
3I		145.8		139.9		138.6
3J	0.3080		0.3727		0.3386	
4A		147.4		133.4		133.4
4B	0.3161		0.2623		0.2940	
4C		153.4		153.0		150.6
4E		122.1		108.0		108.3
4F	0.3336		0.3548		0.3806	
4G		107.2		117.4		115.9
4I		146.3		150.8		150.4
4J	0.3460		0.3959		0.4170	
5A		151.2		140.8		140.0
5B	0.3002		0.2211		0.2555	
5C		131.8		127.0		125.8
5E		100.5		91.04		90.81
5F	0.3547		0.3219		0.3622	
5G		122.9		119.9		118.9
5I		163.0		153.8		153.5
5J	0.4197		0.4007		0.4478	
8A		130.6		112.7		113.5
8B	0.2978		0.2962		0.3155	
8C		153.0		167.4		164.1

TABLE II.1 (cont'd.)

Tube Location	Rod Groups IV, V Inserted Burnup = 100 EFPH		Rod Group V Inserted Burnup = 270 EFPH		All Rod Groups Withdrawn Burnup = 970 EFPH	
	Activation	Nuclear Factor	Activation	Nuclear Factor	Activation	Nuclear Factor
2E		149.3		136.0		134.3
2F	0.2611		0.3252		0.2945	
2G		124.8		145.3		139.5
2I		128.3		138.9		135.0
2J	0.2493		0.3608		0.3174	
3A		145.3		141.3		139.9
3B	0.3272		0.3135		0.3172	
3C		123.1		122.5		116.6
3E		142.2		135.8		135.0
3F	0.3270		0.3454		0.3256	
3G		142.8		145.4		138.8
3I		139.0		137.2		133.3
3J	0.3432		0.3862		0.3485	
4A		133.3		130.5		126.8
4B	0.2904		0.2645		0.2988	
4C		151.8		150.9		148.5
4E		108.3		105.8		108.2
4F	0.3836		0.3646		0.3743	
4G		116.4		115.4		108.4
4I		150.3		147.1		143.5
4J	0.4215		0.4077		0.3805	
5A		139.9		137.8		126.2
5B	0.2552		0.2255		0.3021	
5C		126.8		125.5		145.0
5E		91.14		89.67		105.3
5F	0.3661		0.3311		0.3828	
5G		119.2		117.5		105.5
5I		153.3		149.9		146.0
5J	0.4498		0.4084		0.3950	
8A		112.5		109.4		108.2
8B	0.3032		0.2805		0.2939	
8C		164.3		163.6		155.7

TABLE II.1 (cont'd.)

Tube Location	All Rod Groups Withdrawn Burnup = 1570 EFPH	
	Activation	Nuclear Factor
2E		132.2
2F	0.3005	
2G		137.4
2I		132.8
2J	0.3239	
3A		138.2
3B	0.3237	
3C		113.5
3E		132.9
3F	0.3316	
3G		136.6
3I		131.2
3J	0.3544	
4A		125.1
4B	0.3039	
4C		146.1
4E		106.7
4F	0.3796	
4G		106.8
4I		140.5
4J	0.3865	
5A		124.4
5B	0.3066	
5C		142.5
5E		103.7
5F	0.3874	
5G		104.0
5I		143.0
5J	0.3998	
8A		107.0
8B	0.3000	
8C		151.5

TABLE II.2

TABULATION OF CALCULATED ACTIVITIES AND NUCLEAR FACTORS FOR CYCLE 2

Tube Location	Rod Groups IV, V Inserted Burnup = 0 EFPH		Rod Groups IV, V Inserted Burnup = 100 EFPH		Rod Group V Inserted Burnup = 200 EFPH	
	Activation	Nuclear Factor	Activation	Nuclear Factor	Activation	Nuclear Factor
2E		144.9		145.9		134.2
2F	0.2386		0.2394		0.2952	
2G		116.0		118.0		138.9
2I		121.3		122.9		134.6
2J	0.2146		0.2181		0.3168	
3A		141.6		142.8		140.4
3B	0.3523		0.3423		0.3265	
3C		110.7		112.3		113.1
3E		139.3		140.0		134.9
3F	0.3068		0.3071		0.3232	
3G		131.2		133.6		137.9
3I		131.2		132.6		132.7
3J	0.3005		0.3057		0.3448	
4A		163.5		163.7		162.1
4B	0.3466		0.3269		0.2973	
4C		172.8		174.9		175.7
4E		107.2		107.5		105.8
4F	0.3751		0.3748		0.3579	
4G		106.8		109.0		109.4
4I		140.9		142.6		141.3
4J	0.3798		0.3854		0.3771	
5A		134.8		135.9		135.1
5B	0.2981		0.2873		0.2557	
5C		119.1		121.3		121.3
5E		112.8		113.4		112.5
5F	0.3567		0.3578		0.3271	
5G		148.9		151.4		151.2
5I		143.2		144.7		143.9
5J	0.4169		0.4237		0.3904	
8A		109.6		110.5		108.7
8B	0.3265		0.3172		0.3026	
8C		147.8		150.0		151.0

TABLE II.2 (cont'd.)

Tube Location	All Rod Groups Withdrawn Burnup = 900 EFPH		All Rod Groups Withdrawn Burnup = 1450 EFPH	
	Activation	Nuclear Factor	Activation	Nuclear Factor
2E		133.5		132.0
2F	0.2620		0.2667	
2G		134.1		132.8
2I		131.5		129.3
2J	0.2728		0.2797	
3A		138.9		137.4
3B	0.3243		0.3294	
3C		107.9		105.3
3E		134.8		133.3
3F	0.3001		0.3046	
3G		132.2		130.8
3I		129.7		128.2
3J	0.3054		0.3103	
4A		158.3		157.1
4B	0.3249		0.3284	
4C		175.4		174.0
4E		108.6		107.4
4F	0.3635		0.3672	
4G		103.3		102.3
4I		138.5		136.5
4J	0.3473		0.3515	
5A		124.0		122.5
5B	0.3278		0.3310	
5C		140.4		138.3
5E		132.4		130.4
5F	0.3736		0.3767	
5G		135.6		133.6
5I		140.9		138.4
5J	0.3725		0.3766	
8A		107.5		106.4
8B	0.3005		0.3052	
8C		144.1		140.6

TABLE II.3

TABULATION OF CALCULATED ACTIVITIES AND NUCLEAR FACTORS FOR MODIFIED CYCLE 2

Tube Location	Rod Groups II, V Inserted Burnup = 0 EFPH		Rod Groups II, V Inserted Burnup = 100 EFPH		Rod Group II Inserted Burnup = 400 EFPH	
	Activation	Nuclear Factor	Activation	Nuclear Factor	Activation	Nuclear Factor
2E		131.9		133.6		134.3
2F	0.3222		0.3220		0.2795	
2G		136.5		138.2		134.9
2I		131.4		133.3		131.8
2J	0.3537		0.3515		0.2955	
3A		139.4		141.0		140.6
3B	0.3626		0.3565		0.3453	
3C		110.1		111.7		109.0
3E		135.5		137.2		138.7
3F	0.3290		0.3279		0.2993	
3G		132.7		134.3		130.1
3I		167.1		168.5		167.5
3J	0.3647		0.3535		0.3035	
4A		162.7		163.4		160.2
4B	0.3376		0.3226		0.3438	
4C		169.3		171.3		172.9
4E		112.5		113.5		116.7
4F	0.3205		0.3199		0.3287	
4G		91.50		93.20		89.67
4I		126.7		128.4		127.7
4J	0.3044		0.3050		0.2796	
5A		134.8		136.3		126.2
5B	0.2824		0.2759		0.3433	
5C		116.3		118.5		138.3
5E		115.6		116.7		138.4
5F	0.2902		0.2921		0.3465	
5G		141.3		144.0		130.9
5I		140.4		142.6		141.8
5J	0.3172		0.3216		0.3156	
8A		107.9		109.1		108.8
8B	0.3360		0.3304		0.3200	
8C		147.0		149.1		145.5

TABLE II.3 (cont'd.)

Tube Location	All Rods Out Burnup = 1400 EFPH		All Rods Out Burnup = 2100 EFPH		All Rods Out Burnup = 2400 EFPH	
	Activation	Nuclear Factor	Activation	Nuclear Factor	Activation	Nuclear Factor
2E		121.4		120.1		121.0
2F	0.2878		0.2927		0.2910	
2G		127.3		125.7		126.6
2I		121.2		119.7		120.6
2J	0.3210		0.3255		0.3234	
3A		128.0		126.7		127.6
3B	0.3275		0.3345		0.3329	
3C		99.86		96.98		96.95
3E		122.3		121.0		121.9
3F	0.3263		0.3310		0.3289	
3G		127.2		125.3		126.1
3I		157.4		156.2		157.4
3J	0.3661		0.3687		0.3662	
4A		146.6		145.6		146.9
4B	0.3210		0.3267		0.3247	
4C		165.0		163.5		164.8
4E		99.20		98.29		99.08
4F	0.3835		0.3874		0.3845	
4G		99.79		98.42		99.14
4I		130.4		128.0		128.6
4J	0.3915		0.3941		0.3912	
5A		114.3		113.1		113.9
5B	0.3213		0.3273		0.3253	
5C		130.4		128.5		129.3
5E		120.6		119.1		119.8
5F	0.3850		0.3886		0.3856	
5G		128.5		126.1		126.8
5I		131.6		129.2		129.6
5J	0.3998		0.4014		0.3987	
8A		99.06		98.05		98.75
8B	0.3035		0.3100		0.3085	
8C		133.3		129.5		129.4

APPENDIX III

FUEL ASSEMBLY BURNUP ACCUMULATION

TABLE III.1

CORE I FUEL ASSEMBLY BURNUP ACCUMULATION

Core Location	Fuel Enrichment (w/o U-235)	PREVIOUS IRRADIATION			Equivalent Exposure Hours For _____	Total Equivalent Exposure Hours At End Of Core I
		Core Location	Period	Equivalent Exposure, Hrs.		
2E	2.0					8508
2G	2.0					7967
2I	2.0					7744
2K	2.0					7744
2M	2.0					7816
2O	2.0					8348
3A	2.0					8198
3C	2.0					8902
3E	2.0					9125
3G	2.0					9711
3I	2.0					9991
3K	2.0					9991
3M	2.0					9730
3O	2.0					9120
3Q	2.0					8790
3S	2.0					8058
4A	2.0					8952
4C	2.0					8880
4E	1.5					8207
4G	1.5					9766
4I	1.5					10611
4K	1.5					10611
4M	1.5					10450
4O	1.5					8737
4Q	2.0					9004
4S	2.0					9073
5A	2.0					9545
5C	2.0					9986
5E	1.5					9839
5G	1.5					11499
5I	1.5					11896
5K	1.5					11896
5M	1.5					11503
5O	1.5					9832
5Q	2.0					10148
5S	2.0					9693
6A	2.0					9629
6C	2.0					10097
6E	1.5					9929
6G	1.5					11611
6I	1.5					12243
6K	1.5					12243
6M	1.5					11543
6O	1.5					9880

TABLE III.1 (cont'd.)

[illegible]

TABLE III.2

CYCLE 1 FUEL ASSEMBLY BURNUP ACCUMULATION

Core Location	Fuel Enrichment (w/o U-235)	PREVIOUS IRRADIATION			Equivalent Exposure Hours For Cycle 1	Total Equivalent Exposure Hours At End of Cycle 1
		Core Location	Period	Equivalent Exposure, Hrs.		
2E	2.0					10378
2G	2.0					9925
2I	2.0					9888
2K	2.0					9888
2M	2.0					9776
2O	2.0					10221
3A	2.0	Fresh Assembly				1845
3C	HPD	Fresh Assembly - HPD Test Group #3				1563
3E	2.0					11278
3G	2.0					11972
3I	2.0					12365
3K	2.0					12365
3M	2.0					11978
3O	2.0					11260
3Q	2.0	Fresh Assembly				1892
3S	HPD	Fresh Assembly - HPD Test Group #4				1603
4A	2.0					10580
4C	2.0					10770
4E	1.5					10132
4G	1.5					11793
4I	2.0	3A	Core I	8198.488	2650.321	10849
4K	2.0	3C	Core I	8901.733	2650.316	11552
4M	1.5					12630
4O	1.5					10808
4Q	2.0					11090
4S	2.0					10870
5A	2.0					11267
5C	2.0					11739
5E	1.5					11737
5G	1.5					13703
5I	2.0	3Q	Core I	8789.859	2853.337	11643
5K	2.0	3S	Core I	8057.891	2853.335	10911
5M	1.5					13660
5O	1.5					11690
5Q	2.0					11924
5S	2.0					11436
6A	2.0					11377
6C	2.0					11879
6E	1.5					11846
6G	1.5					13837
6I	2.0	8A	Core I	8250.833	3004.524	11255
6K	2.0	8C	Core I	8968.381	3004.526	11973
6M	1.5					13733
6O	1.5					11767

TABLE III.2 (cont'd.)

[illegible]

TABLE III.3

CYCLE 2 FUEL ASSEMBLY BURNUP ACCUMULATION

Core Location	Fuel Enrichment (w/o U-235)	PREVIOUS IRRADIATION			Equivalent Exposure Hours For <u>Cycle 2</u>	Total Equivalent Exposure Hours At End of <u>Cycle 2</u>
		Core Location	Period	Equivalent Exposure, Hrs.		
2E	2.0					11645
2G	2.0					11231
2I	2.0					11331
2K	2.0					11317
2M	2.0					11092
2O	2.0					11516
3A	1.5	Assemblies Inserted in November 1966 After Removal of HPD Test Group #3 and Standard 2.0 w/o				1081
3C	1.5					1148
3E	2.0					12758
3G	2.0					13430
3I	2.0	Fresh Assembly				1946
3K	2.0	Fresh Assembly				1919
3M	2.0					13428
3O	2.0					12763
3Q	2.0					3615
3S	HPD					2697
4A	2.0	Fresh Assembly				1827
4C	2.0	Fresh Assembly				1993
4E	1.5					11562
4G	1.5					12991
4I	2.0	3A	Core I	8198.488	1461.130	12310
4K	2.0	3C	Core I	8901.733	1436.573	12989
4M	1.5					13923
4O	1.5					12362
4Q	2.0	Fresh Assembly				2055
4S	2.0	Fresh Assembly				1887
5A	2.0					12719
5C	2.0					13344
5E	2.0	4A		10580.167	1687.702	12268
5G	2.0	4C		10770.251	1668.880	12439
5I	2.0	3Q	Core I	8789.859	1802.343	13446
5K	2.0	3S	Core I	8057.891	1769.241	12680
5M	2.0	4Q		11090.392	1777.767	12868
5O	2.0	4S		10870.235	1778.087	12648
5Q	2.0					13528
5S	2.0					12895
6A	2.0					12816
6C	2.0					13460
6E	2.0	7A		11097.507	1770.580	12868
6G	2.0	7C		11364.246	1740.181	13104
6I	2.0	8A	Core I	8250.833	1808.199	13064
6K	2.0	8C	Core I	8968.381	1771.957	13745
6M	2.0	7Q		11244.928	1713.910	12959
6O	2.0	7S		11044.656	1750.065	12795

TABLE III.3 (cont'd.)

[illegible]

APPENDIX IV

TABULATION OF RELATIVE FUEL ROD POWER SHARING FROM CRITICAL
EXPERIMENTS FOR HIGH POWER DENSITY FUEL ASSEMBLIES

TABLE IV.1

RELATIVE POWER PRODUCTION IN 0.6-INCH PITCH CLUSTER

<u>Position</u>	<u>Run 1</u>	Enrichment Relative Power		<u>Run 3</u>	<u>Run 4</u>
		<u>Run 2</u>			
1	3.7 0.999	2.7 0.873		1.6 0.693	2.7 1.206
2	3.7 1.006	3.7 1.121		3.7 1.311	1.6 0.798
3	3.7 0.997	2.7 0.875		1.6 0.664	2.7 1.206
4	3.7 1.000	3.7 1.125		3.7 1.336	1.6 0.779
5	3.7 1.005	2.7 0.872		1.6 0.667	2.7 1.221
6	3.7 0.994	3.7 1.134		3.7 1.329	1.6 0.791
Power Ratio	1.000	1.290 ± .007		1.965 ± .024	1.434 ± .014
Enrichment Ratio	1.000	1.3704		2.3125	1.6875

TABLE IV.2

RELATIVE POWER PRODUCTION IN 0.6-INCH PITCH CLUSTER WITH FLUX TILT

<u>Position</u>	<u>Run 5</u>	Enrichment Relative Power		<u>Run 7</u>	<u>Run 8</u>
		<u>Run 6</u>			
1	3.7 1.004	2.7 0.877		1.6 0.665	2.7 1.218
2	3.7 0.970	3.7 1.095		3.7 1.291	1.6 0.756
3	3.7 0.967	2.7 0.846		1.6 0.641	2.7 1.181
4	3.7 0.992	3.7 1.128		3.7 1.338	1.6 0.783
5	3.7 1.037	2.7 0.891		1.6 0.683	2.7 1.255
6	3.7 1.030	3.7 1.163		3.7 1.381	1.6 0.807
Power Ratio	1.000	1.296 ± .006		2.016 ± .014	1.558 ± .007
Enrichment Ratio	1.000	1.3704		2.3125	1.6875

TABLE IV.3

RELATIVE POWER PRODUCTION IN 0.4-INCH PITCH CLUSTER

<u>Position</u>	<u>Run 9</u>	Enrichment Relative Power	<u>Run 11</u>	<u>Run 12</u>
		<u>Run 10</u>		
1	2.7 1.001	2.7 0.871	1.6 0.670	2.7 1.216
2	3.7 0.996	3.7 1.133	3.7 1.342	1.6 0.777
3	3.7 1.000	2.7 0.865	1.6 0.661	2.7 1.235
4	3.7 0.992	3.7 1.126	3.7 1.337	1.6 0.783
5	3.7 0.999	2.7 0.870	1.6 0.653	2.7 1.214
6	3.7 1.010	3.7 1.135	3.7 1.337	1.6 0.777
Power Ratio	1.000	1.302 $\pm$.006	2.024 $\pm$.018	1.569 $\pm$.013
Enrichment Ratio	1.000	1.3704	2.3125	1.6875

TABLE IV.4

RELATIVE POWER PRODUCTION IN 0.4-INCH PITCH CLUSTER WITH FLUX TILT

<u>Position</u>	Enrichment Relative Power	
	<u>Run 13</u>	<u>Run 14</u>
1	2.7 0.876	2.7 1.228
2	3.7 1.099	1.6 0.780
3	2.7 0.844	2.7 1.168
4	3.7 1.133	1.6 0.796
5	2.7 0.902	2.7 1.241
6	3.7 1.147	1.6 0.788
Power Ratio	1.289 ± .010	1.539 ± .027
Enrichment Ratio	1.3704	1.6875

TABLE IV.5

RELATIVE FLUX AROUND FUEL RODS IN 0.6 INCH PITCH CLUSTER

<u>Position</u>	<u>Relative Flux</u>			
	<u>Run 1</u>	<u>Run 2</u>	<u>Run 3</u>	<u>Run 4</u>
2a	0.980	0.915	0.987	0.978
b	1.009	0.987	1.020	0.984
c	1.061	1.004	1.032	1.022
d	1.048	1.008	1.004	1.059
e	1.012	0.970	0.981	1.052
f	0.979	0.928	0.950	1.003
Average	1.015	0.969	0.996	1.016
Rod Enrichment	3.7	3.7	3.7	1.6
3a	0.936	0.973	0.967	0.988
b	0.983	0.980	0.966	1.022
c	0.994	1.017	0.966	0.990
d	1.054	1.099	1.038	0.970
e	1.033	1.086	1.056	0.958
f	0.968	1.034	1.034	0.974
Average	0.995	1.031	1.004	0.984
Rod Enrichment	3.7	2.7	1.6	2.7
5a	1.001			
b	1.053			
c	1.023			
d	0.981			
e	---			
f	---			
Average	---			
Rod Enrichment	3.7			
6a	0.998			
b	0.972			
c	0.941			
d	0.979			
e	1.018			
f	1.036			
Average	0.991			
Rod Enrichment	3.7			

TABLE IV.6

RELATIVE FLUX AROUND FUEL RODS IN 0.6" PITCH CLUSTER WITH FLUX TILT

<u>Position</u>	<u>Relative Flux</u>			
	<u>Run 5</u>	<u>Run 6</u>	<u>Run 7</u>	<u>Run 8</u>
2a	0.912	0.916	0.962	0.924
b	0.956	0.938	0.971	0.939
c	0.987	0.986	1.012	0.966
d	1.013	1.003	0.981	0.976
e	0.974	0.981	0.958	0.971
f	0.918	0.928	0.939	0.961
Average	0.960	0.959	0.970	0.956
Rod Enrichment	3.7	3.7	3.7	1.6
3a	0.974	0.950	0.950	0.948
b	0.939	0.950	0.957	0.965
c	0.930	0.989	0.962	0.982
d	0.947	1.039	0.996	1.006
e	0.998	1.021	1.021	0.994
f	0.995	0.988	0.981	0.975
Average	0.964	0.989	0.978	0.978
Rod Enrichment	3.7	2.7	1.6	2.7
5a	1.102	0.969	0.996	0.933
b	1.064	1.010	1.014	0.955
c	1.027	1.075	1.070	1.028
d	0.980	1.117	1.089	1.051
e	0.998	1.057	1.072	1.055
f	1.048	1.059	1.025	1.012
Average	1.037	1.048	1.044	1.006
Rod Enrichment	3.7	2.7	1.6	2.7
6a	1.098	0.940	0.955	1.016
b	1.087	0.949	0.965	1.048
c	1.018	1.024	1.019	1.089
d	0.968	1.060	1.053	1.102
e	0.993	1.055	1.063	1.070
f	1.075	0.996	0.990	1.034
Average	1.040	1.004	1.007	1.060
Rod Enrichment	3.7	3.7	3.7	1.6

TABLE IV.7

RELATIVE FLUX AROUND FUEL RODS IN 0.4" PITCH CLUSTER

<u>Position</u>	<u>Relative Flux</u>	
	<u>Without Tilt</u> <u>(Run 9)</u>	<u>With Tilt</u> <u>(Run 13)</u>
2a	0.929	0.853
b	0.951	0.887
c	1.068	0.995
d	1.101	1.046
e	1.060	1.018
f	0.965	0.940
Average	1.012	0.956
Rod Enrichment	3.7	3.7
3a	0.909	
b	1.043	
c	0.923	
d	1.052	
e	1.040	
f	0.931	
Average	0.983	
Rod Enrichment	3.7	
5a	0.906	
b	0.986	
c	1.056	
d	1.078	
e	1.022	
f	0.919	
Average	0.994	
Rod Enrichment	3.7	
6a	0.922	0.917
b	0.959	0.985
c	1.060	1.126
d	1.099	1.159
e	1.066	1.085
f	0.955	0.991
Average	1.010	1.044
Rod Enrichment	3.7	3.7

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