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SUMMARY OF THE HYDRAULIC EVALUATION OF LWBR

(LWBR Development Program)

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BETTIS ATOMIC POWER LABORATORY
WEST MIFFLIN, PENNSYLVANIA

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SUMMARY OF THE HYDRAULIC EVALUATION OF LWBR
(LWBR Development Program)

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FOREWORD

The Shippingport Atomic Power Station located in Shippingport, Pennsylvania was the first large-scale, central-station nuclear power plant in the United States and the first plant of such size in the world operated solely to produce electric power. This program was started in 1953 to confirm the practical application of nuclear power for large-scale electric power generation. It has provided much of the technology being used for design and operation of the commercial, central-station nuclear power plants now in use.

Subsequent to development and successful operation of the Pressurized Water Reactor in the Atomic Energy Commission (now Department of Energy, DOE) owned reactor plant at the Shippingport Atomic Power Station, the Atomic Energy Commission in 1965 undertook a research and development program to design and build a Light Water Breeder Reactor core for operation in the Shippingport Station.

The objective of the Light Water Breeder Reactor (LWBR) program has been to develop a technology that would significantly improve the utilization of the nation's nuclear fuel resources employing the well-established water reactor technology. To achieve this objective, work has been directed toward analysis, design, component tests, and fabrication of a water-cooled, thorium oxide-uranium oxide fuel cycle breeder reactor for installation and operation at the Shippingport Station. The LWBR core started operation in the Shippingport Station in the Fall of 1977 and is expected to be operated for about 4 to 5 years or more. At the end of this period, the core will be removed and the spent fuel shipped to the Naval Reactors Expended Core Facility for a detailed examination to verify core performance including an evaluation of breeding characteristics.

In 1976, with fabrication of the Shippingport LWBR core nearing completion, the Energy Research and Development Administration, now DOE, established the Advanced Water Breeder Applications (AWBA) program to develop and disseminate technical information which would assist U.S. industry in evaluating the LWBR concept for commercial-scale applications. The program is exploring some of the problems that would be faced by industry in adapting technology confirmed in the LWBR program. Information being developed includes concepts for commercial-scale prebreeder cores which would produce uranium-233 for light water breeder cores while producing electric power, improvements for breeder cores based on the technology developed to fabricate and operate the Shippingport LWBR core, and other information and technology to aid in evaluating commercial-scale application of the LWBR concept.

All three development programs (Pressurized Water Reactor, Light Water Breeder Reactor, and Advanced Water Breeder Applications) are under the technical direction of the Office of the Deputy Assistant Secretary for Naval Reactors of DOE. They have the goal of developing practical improvements in the utilization of nuclear fuel resources for generation of electrical energy using water-cooled nuclear reactors.

Technical information developed under the Shippingport, LWBR, and AWBA programs has been and will continue to be published in technical memoranda, one of which is this present report.

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The principal hydraulic performance features of the Light Water Breeder Reactor are summarized in this report. The calculational models and procedures used for prediction of reactor flow and pressure distributions under steady-state and transient operating conditions are described. Likewise, the analysis models for evaluation of the static and dynamic performance characteristics of the hydraulically-balanced and hydraulically-buffered movable-fuel reactivity-control system are outlined. An extensive test program was conducted for qualification of the subject LWBR hydraulic evaluation models. The projected LWBR hydraulic performance is shown to fulfill design objectives and functional requirements. Results of design performance assessments concerning reactor flow levels, pressure and flow distributions, flow coastdown rates, and movable fuel scram characteristics (acceleration, terminal velocity and deceleration phases) are presented and compared with corresponding parameters measured during the LWBR precritical test program at the Shippingport plant site. These evaluations provide the basis of the hydraulic design assurance for LWBR operation in the Shippingport Atomic Power Station.

SUMMARY OF THE HYDRAULIC EVALUATION OF LWBR (LWBR Development Program)

J.W. Stout, S. Lerner, K.D. McWilliams, J.R. Turner

I. INTRODUCTION

The Light Water Breeder Ractor (LWBR) Program is developing the technology to breed fissile material (U-233) using the Th-232/U-233 fuel cycle in a pressurized, light-water reactor. This technology exhibits the potential for significant improvement in utilization of nuclear fuel resources in light-water thermal reactors and could be accommodated in commercial pressurized water plant designs. The LWBR core represents the first full-core power reactor usage of a thorium-base binary fuel of high density. Installation of the LWBR demonstration core was completed in the fall of 1977 at the Shippingport Atomic Power Station where it has been operated at a rated power of 236.6 Mw (thermal).

This report describes the hydraulic design and performance of the LWBR core. An introductory overview of the LWBR design is given in Reference 1; nuclear and thermal design summaries are presented in References 2 and 3. LWBR incorporates several unique design features engendered by the overall objective to demonstrate breeding in a light-water environment. These features are described in Section II which also presents summary descriptive discussions of the reactor mechanical design, the primary coolant system, and the principal hydraulic design objectives. The hydraulic models and analysis procedures, with supporting experimental qualification information, used in assessment of LWBR performance are described in Section III. Calculated results of LWBR hydraulic evaluations are given in Section IV and in Section V these results are compared with measured hydraulic performance parameters obtained during the LWBR start-up test program.

II. BACKGROUND DESCRIPTION AND HYDRAULIC OBJECTIVES

A. Unique Features

To achieve the degree of efficient neutron utilization needed to fulfill the goal of breeding in a light water reactor, a number of novel features were introduced which had a significant influence on reactor hydraulic objectives and hydro-mechanical design considerations. These features, discussed from nuclear design and thermal-hydraulic viewpoints in References 2 and 4, respectively, are outlined below as a preface to (a) presentation of reactor mechanical and primary coolant system descriptions and (b) definition of hydraulic objectives.

1. The LWBR core design employs a compact array of fuel rods in a seed-blanket arrangement for optimizing neutron usage and utilizes a radial reflector region to minimize neutron losses from the core. The reflector is loaded only with fertile thorium, while the seed and blanket are loaded with both thorium and fissile uranium-233. On a weight-percent basis fissile loading is higher in the seed than in the blanket. As a result of these fuel loading variations, significant

differences in volumetric power generation exist among the seed, blanket and reflector regions. Consequently, to more nearly balance region thermal capabilities and more efficiently use available coolant flow, orificing of flow to the blanket and reflector regions is required. The seed fuel modules are contained within Zircaloy guide tubes and Zircaloy shells are provided around reflector fuel modules to accomplish hydraulic separation, thereby enabling the blanket and reflector to be orificed. Further constraint on hydraulic design is imposed by the fact that these guide tubes and shells must be thin to minimize parasitic neutron absorption while their deflections must be limited in order to provide unobstructed pathways for fuel-rod coolant flow and for translation of the movable seed fuel used in reactivity control. LWBR hydraulic design has incorporated tri-plate, multiholed orifices at both top and bottom of blanket and reflector modules. This mode of orificing reduces pressure differentials and the resulting lateral loads on the reflector shells and on the movable fuel guide tubes by about 50 percent, as compared with pressure loads from only inlet orificing, while achieving the desired distribution of coolant flow.

2. For neutron conservation purposes, reactivity control in LWBR is achieved by variable fuel geometry, made practical by the seed-blanket concept, rather than by conventional poison rods, soluble poison or burnable poison. Variable geometry nuclear control has been accomplished (a) by providing control drive mechanisms for vertical translation of movable seed fuel modules through guide tubes located in the center of the surrounding, stationary blanket fuel modules and (b) by stepped zoning of seed and blanket fuel such that neutron-absorbing thorium wedges are created at the top of seed modules and at the bottom of blanket modules. This control concept is shown in Figure 1 and discussed more fully in Reference 2. To assure for shutdown or scram purposes a net downward force on the

FIGURE 1. VARIABLE GEOMETRY NUCLEAR CONTROL CONCEPT

movable elements approximately equal to their weight, LWBR uses a hydraulic technique for counterbalancing upward hydrodynamic friction and drag forces caused by axial flow of coolant through the movable fuel. Hydraulic balancing is achieved by applying core pressure drop across a balance piston attached above the movable fuel. A low-resistance flow path, termed the Bypass Inlet Flow (BIF) pressure supply line, is used to apply core inlet pressure above the balance piston with coolant leakage then flowing downward around the high-resistance piston to essentially outlet plenum pressure. A hydraulic buffer is included in the translating system design to decelerate the movable fuel and limit impact velocity at the end of its scram stroke. Figure 2 shows a schematic representation of the hydraulic balancing and hydraulic buffer systems. Additional description of these systems is found in Reference 5.

3. To achieve a more uniform radial flux distribution, a lower metal-to-water ratio and higher fissile fuel loadings have been utilized on the outboard sides of the nine peripheral blanket modules. This region, which is identified in Figure 3 (LWBR core cross-section), is consequently termed the power-flattening blanket. The standard blanket and power-flattening blanket regions are hydraulically coupled; that is, they are not separated by structural shells which would tend to increase parasitic neutron absorption. Fuel rod pitch is the same in the two regions, but a smaller rod diameter is used in the power-flattening blanket. The flow split between standard and power-flattening rod regions is therefore governed by their relative hydraulic resistances. Coolant velocity is thus higher in the power-flattening blanket than in the standard blanket. Crossflow between these hydraulically coupled regions, which potentially could be detrimental from a fuel rod vibratory wear consideration, is minimized by use of appropriately smaller holes in the standard blanket portions

FIGURE 2. SCHEMATIC ARRANGEMENT OF BIF BALANCING SYSTEM AND FLOW PATHS

LEGEND

MOVABLE SEED
ROD REGION

STANDARD BLANKET
ROD REGION

FILLER UNITS
(NON-FUEL)

REFLECTOR
ROD REGION

POWER FLATTENING
BLANKET ROD REGION

MODULE TYPES

DESCRIPTION

I SEED ASSEMBLY SURROUNDED BY STANDARD BLANKET ROD ASSEMBLY.

II, III SEED ASSEMBLY SURROUNDED BY STANDARD BLANKET AND POWER-
FLATTENING BLANKET ROD ASSEMBLY. TYPES II AND III DIFFER DUE
TO OVERALL BOUNDARY DIMENSIONS AND DISTRIBUTION OF THE TWO
BLANKET REGIONS.

IV, V REFLECTOR MODULES. TYPES IV AND V DIFFER DUE TO OVERALL
BOUNDARY DIMENSIONS.

FIGURE 3. CROSS SECTION OF LWBR CORE WITH
MODULE IDENTIFICATION

of the tri-plate orifices located upstream and downstream of the blanket regions. This orifice hole-size arrangement produces the same flow split at the inlet and outlet elevations as in the blanket rodged regions and consequently reduces the potential for crossflow.

4. To minimize neutron absorption in water, LWBR fuel rods are placed in a compact, triangular-pitched array. The rod-to-rod spacing in LWBR seed, standard blanket and reflector regions (~ 60 mils) is about half that employed in other current pressurized water reactor designs. Within this closely-spaced array such factors as rod bow, clad ovality and wear at support elevations, manufacturing tolerances on pitch and diameter, and rod diametral shrinkage or swelling result in localized flow redistributions of significance in thermal and hydraulic performance evaluation. These effects, discussed further in Reference 3, are treated as hot channel allowances in LWBR thermal analyses.

B. Core Mechanical Description

As shown in Figure 3, LWBR contains twelve movable seed assemblies arranged symmetrically on a 17.2-inch triangular pitch in the center of the core. Each seed assembly, encompassed by a thin, perforated, Zircaloy shell, is located within an annular blanket assembly bounded internally by a Zircaloy guide tube through which the seed assembly is translated. A fuel module consists of a movable seed assembly and its adjunct, stationary blanket assembly. The three center (Type I) modules have a regular hexagonal cross-section as detailed in Figure 4. The remaining nine (Type II and III) modules are symmetric non-regular hexagons wherein the extended sections on the outboard periphery contain the power-flattening blanket fuel rods; see Figures 5 and 6. The regular inboard sections of the Type II and III blanket assemblies and all portions of the Type I blanket assemblies contain the standard blanket fuel rods.

FIGURE 4. TYPE I MODULE CROSS SECTION

FIGURE 5. TYPE II MODULE CROSS SECTION

FIGURE 6. TYPE III MODULE CROSS SECTION

FIGURE 7. TYPE IV AND TYPE V REFLECTOR MODULE CROSS SECTION

The twelve principal power-producing seed/blanket modules containing both fertile and fissile fuel are surrounded by fifteen (Type IV and V) reflector modules. The reflector fuel rods, which initially contain only fertile fuel, are enclosed by Zircaloy shells as shown in Figure 7. As previously indicated, the reflector region serves to reduce neutron leakage from the LWBR core. The irregular space between the reflector modules and the core barrel is occupied by fifteen, non-fuel (stainless steel) filler units whose function is to limit coolant leakage flow through this area.

The fuel rods in the four core regions are approximately ten feet long and are arranged in closely-spaced, triangular-pitched arrays. Details of the fuel rod design are given in Reference 6. About half the rods in a fuel assembly or module are axially supported by a top baseplate with the remainder supported by a bottom baseplate. Lateral support is provided by stainless-steel, honeycomb support grids at nine axial levels in seed assemblies, eight levels in blanket assemblies and six levels in reflector modules. The rod support grid system is described in Reference 7. Longitudinal views of a movable fuel assembly, Type I and Type III blanket assemblies and a Type IV reflector module are shown in Figures 8 through 11, respectively. Representative core parameters are presented in Table 1.

In-core instrumentation is provided to measure (a) seed flowrates at seven locations, (b) blanket flowrates at six locations, (c) core pressure drop at one location and (d) BIF balance piston pressure differential at three locations indicated in Figure 12. Movable seed position is also monitored, and scram performance is measured periodically.

**FIGURE 8. MOVABLE FUEL ASSEMBLY
(BUFFER CYLINDER AND TRANSLATING ASSEMBLY INSTALLED)**

FIGURE 9. TYPE I BLANKET ASSEMBLY

FIGURE 10. TYPE III BLANKET ASSEMBLY

FIGURE 11. TYPE IV REFLECTOR ASSEMBLY

TABLE 1
LWBR CORE PARAMETERS

	<u>Seed</u>	<u>Standard Blanket</u>	<u>Power- Flattening Blanket</u>	<u>Reflector</u>
<u>Power Fraction</u>	0.380	0.321	0.298	0.001
<u>Flow* Fraction</u>	0.249	0.185	0.321	0.094
<u>Average Nuclear Heat Flux @ 100% Power x 10⁻³, BTU/hr-ft^{2**}</u>	59.5	61.5	60.2	4.2
<u>Maximum Nuclear Heat Flux @ 100% Power x 10⁻³, BTU/hr-ft^{2**}</u>	375.	267.	285.	77.
<u>Mass Velocity x 10⁻⁶, lb/hr-ft²</u>	2.409	2.225	2.696	0.737
<u>Fuel Rod Diameter, in.</u>	0.306	0.572	0.527	0.832
<u>Hydraulic Diameter (infinite array), in.</u>	0.187	0.198	0.309	0.246
<u>Pitch, in.</u>	0.369	0.630	0.630	0.901
<u>Total Number</u>	7428	3234	3581	3047
<u>Number of Grid Levels</u>	9	8	8	6
<u>Flow Area, ft²</u>	2.839	2.545	3.698	4.103
<u>Heat Transfer Area, ft²</u>	5184.	4220.	4307.	5667.
<u>Heated Wetted Perimeter, in.</u>	7141.	5803.	5923.	7964.
<u>Unheated Wetted Perimeter, in.</u>	438.	369.	558.	1049.

* Remaining flow goes to movable fuel balancing system (0.021) and bypass paths (0.130).

** Heat Fluxes are for 100 full power hours of operation except for the reflector which is at 18000 hours. The heat flux in the reflector initially containing only thorium is low at beginning of life. With increasing life time, the thorium is transmuted to uranium-233 and the power generated in the reflector fuel rods increases.

LEGEND:

- ◆ BIF PRESSURE DROP LOCATION (3)
- - BLANKET FLOWMETER LOCATION (6)
- - SEED FLOWMETER LOCATION (7)
- ⊙ - SPARE SEED FLOWMETER LOCATION (3)
- ◇ - CORE PRESSURE DROP LOCATION (1)

FIGURE 12. IN-CORE HYDRAULIC INSTRUMENTATION LOCATIONS

C. Primary Coolant System Description

1. Loop Characteristics

The Shippingport plant contains four primary, high-pressure, light-water, reactor coolant loops used to transfer heat from the LWBR core to the lower-pressure, secondary (or steam turbine) system. Steam generators provide the heat transfer link between the primary and secondary systems and maintain separation between primary and secondary coolant flow. A typical primary coolant loop is illustrated in Figure 13. Each of the four loops connected to the reactor vessel contains a pair of coolant stop valves in the inlet leg and another pair of stop valves in the outlet leg. A check valve and a venturi flowmeter are also provided. Loop piping is 18-inch (nominal OD), seamless, stainless steel pipe. Installed in each loop is a horizontal steam generator consisting of a heat exchanger, a separate steam drum, and interconnecting risers and downcomers. The heat exchanger has Ni-Cr-Fe alloy tubes with an overall conductance of about 3.1 MW/°F. Each loop contains a vertical, single-stage, canned-motor, centrifugal pump which may be operated at full (1788 rpm) or half speed. A pump head of 440 feet is developed at a full-speed flowrate of approximately 19,000 gpm. The operating power requirement is about 2 MW per pump. To improve the flow coastdown following inadvertent loss of pumping power, each main coolant pump power circuit is connected to a flywheel generator capable of providing the electrical equivalent of 13,000 lb-ft² of mechanical inertia at the pump shaft. Each of the four flywheel generators consists of a 1200-rpm synchronous machine coupled to a large flywheel. During normal operation the flywheel generator acts as an A-C motor driven from the normal plant power distribution system. Upon a loss of pumping power, the flywheel generator acts as a generator to supply power to its associated coolant pump for extending its flow coastdown characteristic. Primary system pressure is maintained by means of a pressurizer vessel, equipped with a spray nozzle at the top and submerged electrical heaters, which is connected to the A-loop outlet piping by a 6-inch surge line. Each coolant loop also has a valved, 3-inch diameter, reactor vessel bypass line connecting inlet and outlet piping between each pair of main stop valves.

FIGURE 13. TYPICAL SHIPPINGPORT PLANT REACTOR
COOLANT LOOP WITH LWBR CORE

2. Coolant Characteristics

The LWBR core is cooled by high-purity, demineralized, deoxygenated, light water. Coolant pH is maintained in the range of 10.1 to 10.3 (at 25°C) by the addition of concentrated ammonium hydroxide solution. Coolant specifications limit chloride, fluoride and boron concentrations each to a maximum of 0.1 ppm, dissolved oxygen to 0.14 ppm or less and total dissolved gases to 125 cc/kg*. Hydrogen gas is maintained in the range of 10 to 60 cc/kg*.

The nominal operating condition at beginning of life for the primary system, which contains about 2450 cubic feet of coolant, is a pressure of 2000 psia at the reactor vessel outlet with a system average temperature of 531°F. For the rated power of 236.6 MW (thermal) and a total coolant flowrate of about 31×10^6 pounds per hour, the corresponding coolant temperature at the reactor vessel inlet is 520°F.

3. Reactor Flowpaths

Figure 14 illustrates the principal coolant flowpaths through the reactor vessel and core. Coolant enters through four inlet nozzles located at the bottom of the reactor vessel. About 15 percent of the inlet coolant flows upward in the annulus between the reactor vessel and thermal shield and returns to the inlet region in the annulus between the thermal shield and lower core barrel to join the main flowstream above the inlet flow baffle. From the inlet region coolant flows upward through the single-pass reactor core to the outlet plenum in several parallel flow paths which include the 12 seed assemblies, the 12 blanket assemblies, the 15 reflector modules, the 6 BIF supply tubes, and various by-pass leakage sites. A representative distribution of coolant is indicated by the flow fractions given earlier in Table 1. The effluent from these paths combines in the outlet plenum and exits from the reactor vessel through four outlet nozzles. Further description of core hydraulic characteristics follows.

*At 0°C and 1 atmosphere.

FIGURE 14.(a) LWBR CORE IN SHIPPINGPORT REACTOR VESSEL
SHOWING CORE HYDRAULIC FLOW PATHS

FIGURE 14. (b) BOTTOM PLATE DETAILS

a. Seed Flowpath

To enter a seed assembly, coolant first goes through a fixed multiholed strainer plate followed by a venturi flowmeter located at the upstream end of a bottom plate extension (Figure 14). It then flows upward within the guide tube extension depicted in Figure 9 and passes through flowholes in the seed bottom coverplate and baseplate before traversing the power-producing fuel-rod/support-grid region of the movable fuel assembly which is enclosed by the perforated shell seen in Figure 8. Coolant exits into the outlet plenum, after passing the top baseplate/coverplate and support-shaft attachment web, through three outlet slots in the blanket support tube (cf. Figures 9 and 14). The annulus between the seed shell and the guide tube permits some flow to bypass the seed fuel rods. The seed shell is perforated to minimize radial pressure differentials.

b. Blanket Flowpath

Multiple-tube venturi-type flowmeters adjacent to the lower surface of the bottom plate (Figure 14) monitor flow entering the blanket stub tubes. Downstream of the flowmeters the blanket coolant encounters a multiholed, triplate orifice and a bottom coverplate/baseplate assembly prior to reaching the rod/grid region (Figure 9). The coolant leaves the blanket fuel-rod region through the multiholed, top baseplate/coverplate assembly and the top triplate orifice. Since no shells are used on the outer periphery of blanket assemblies, the Type I, II and III modules are hydraulically coupled (laterally interconnected) in the blanket (standard and power-flattening) power-producing region (cf. Figure 3). The blanket region is also in hydraulic communication with the leakage gaps among reflector modules and filler units which permit coolant to bypass the active core.

c. Reflector Flowpath

Coolant enters a reflector module through a stub tube which pilots into a hole in the bottom plate. It flows through a bottom triplate orifice and bottom coverplate/baseplate assembly upstream of the fuel-rod/support-grid region and a top baseplate/coverplate assembly and top triplate orifice downstream of the fueled region before exiting into the outlet plenum (see Figure 11).

d. BIF Flowpath

Six BIF supply tubes with lower ends in the inlet plenum are part of the system which conducts water at essentially inlet pressure to the plena above the 12 movable fuel balance pistons. Each supply tube consists of two concentric pipes. Flow in the center pipe is conducted to the suspension system of a specific fuel module by means of a crossover pipe located just below the vessel closure head (see Figure 14). Flow in the annulus between the two supply pipes goes to a second module suspension system via another crossover pipe. BIF flow is conducted upward within annuli in the module suspension system to six holes located at the top of the compression sleeve through which it enters the plenum above the balance piston (Figure 2). BIF flow must then pass through the annular clearances between the balance piston and compression sleeve and between the seed support shaft and buffer cylinder before exiting to the reactor outlet plenum through three oval holes in the blanket support tube (cf. Figures 2, 8, 9 and 14).

D. Hydraulic Design Objectives

The principal hydraulic objectives in designing LWBR to function satisfactorily at rated power, considering steady-state and transient plant operation and postulated accident conditions, are:

1. Provide coolant flow, including means for appropriate flow distribution, adequate to meet fuel rod thermal performance requirements as given in References 3 and 6. These requirements include prevention of critical heat flux, flow instability and steady-state bulk boiling and attainment of a reasonable balance in performance capability of seed and blanket regions.
2. Assure an acceptable pressure distribution which will not result in excessive structural loads or deflections, particularly in the relatively thin shells utilized in this compact breeder configuration.

3. Achieve hydraulic balancing and buffering of the movable fuel. The net force on the movable fuel assembly must be compatible with control-drive-mechanism load requirements, nuclear protection (scram) requirements, and fuel deceleration requirements as highlighted in Section III.B.1 and detailed in Reference 5.

III. HYDRAULIC MODELS AND ANALYSIS TECHNIQUES

The models and analysis techniques used in the hydraulic design and evaluation of LWBR are presented in two parts. The first describes the HAFMAT and PARK* computer models/analyses which are utilized in the calculation of steady-state and transient reactor flow and pressure drop. The second part discusses the BIF system performance analyses.

A. Reactor Flow and Pressure Distribution

1. Analysis Goals

The models developed for the calculation of reactor flow and pressure distribution have been utilized in several analysis phases to satisfy the LWBR hydraulic objectives stated above. For example, design-phase assessments were performed to establish blanket and reflector orifice sizes required to achieve flow and pressure distributions which would provide adequate cooling of all fuel rods and assure acceptable pressure differentials across structures. Subsequent assessments have been performed to calculate hydraulic parameters required in detailed performance evaluations of postulated accident conditions. Such assessments have generated pressure-drop forcing functions for hot-channel XITE or HOTROD** thermal evaluations for reactivity insertion transients and loss of flow accidents, cf. Reference 3, and have provided explicit shell pressure differentials for determination of mechanical clearances to assure movable fuel scram and fuel rod cooling. These hydraulic models and relevant qualification detail are outlined in the remainder of Section III.A.

* References 8 and 9, respectively.

** References 10 and 11, respectively.

2. Steady-State Flow and Pressure-Drop Model

a. Description of Hydraulic Computer Code HAFMAT

The HAFMAT computer program is used to calculate the steady-state core flow and pressure distributions. This program contains the continuity of flow, momentum and thermal energy balance equations as described in Reference 8. In using this program, a network of flowpaths must first be established. The flowpaths are called branches and are analogous to pipes in a conventional flow system. Each branch has one inlet node and outlet node. The flow area, length, hydraulic pressure-loss factors and the amount of heat addition must be specified for each branch. Boundary conditions consist of specifying either pressure or flow and enthalpy at the boundary points. An iterative method is used to arrive at a converged solution.

b. Description of LWBR Hydraulic Model

1. Primary Coolant System Network Description

As noted previously, the LWBR core has four different fuel-rod diameter zones with inlet and exit orificing in the blanket and reflector regions. To facilitate hydraulic calculations and to aid in understanding the effect of component pressure-loss coefficients on core flow distribution (both axial and transverse) and structural pressure differentials, a detailed computer model was established. (The breeding objective necessitated several iterations among the mechanical, nuclear and hydraulic designers.). Figure 15 shows a block-diagram flow schematic of the hydraulic model. Branch and node designations are defined in the legend of the figure. The flow paths represented in the model include the following (starting on the left side of the figure and proceeding to the right): the bypass inlet flow balancing system, the movable seed with the hydraulically-coupled annulus leakage path, the Type I blanket assemblies, the Type II blanket and power-flattening blanket assemblies, the Type III blanket and

FIGURE 15

power-flattening blanket assemblies, the blanket leakage path, the Type IV reflector modules, the Type V reflector modules, a core by-pass leakage path and the four primary loops including the pumps and steam generators. The hydraulic parameters associated with each of these major paths are representative of the entire core. For example, the seed flow path represents all 12 seed assemblies. Similarly, the Type I blanket represents all 3 Type I blanket assemblies, and so on for the other flow paths. All blanket assemblies as well as the blanket leakage path are hydraulically interconnected by crossflow branches to represent the physical reality of an open matrix among the two different fuel-rod diameter zones and the leakage paths formed by the reflector shells, filler units and core barrel.

2. Hydraulic Model Details

Table 2 lists the nominal crud-free beginning-of-life pressure-loss coefficient and flow area, hydraulic diameter, friction length, and power fraction for each component of the hydraulic model shown on Figure 15. The majority of the LWBR core pressure-loss coefficients were determined from results of hydraulic testing of each major component (e.g. grids, orifice plates, flowmeters, baseplates, balance pistons, etc., as described in Reference 12). These single-component pressure-loss coefficients were subsequently confirmed and/or fine-tuned by a full-scale mockup hydraulic test. This test, entitled REM-3, is illustrated in Figure 16. REM-3 represented approximately 1/12 of the LWBR core and contained a movable fuel assembly with balance piston, a Type III blanket assembly and suspension system with buffer cylinder, a BIF supply tube with crossover piping, a control drive mechanism with associated power supply, a Type V reflector module, and a typical filler unit. Pressure-loss coefficients for the reactor vessel nozzles, thermal shield, inlet flow baffle, seed outlet slots, SID plate, hold-down barrel and core barrel were determined from a 0.4-scale, air-flow model test of LWBR similar to that performed for PWR-2, Reference 13. Loss coefficients for loop piping and steam generators were based on calculation and prior PWR operating experience. For a few flow branches, principally leakage paths, loss coefficients were based on

TABLE 2
SUMMARY OF HAFMAT INPUT PARAMETERS

	HAFMAT Branch Number	Total Flow Area for Core Region, ft ²	Loss Coeff. K Nom.	Hydraulic Diameter D _H (Inches)	Friction Length L (feet)	Power Fraction
<u>BIF</u>						
Bypass Line and strainer	33	.408	10.1			
Balance piston	33	.1083	14.5			
<u>SEED</u>						
Strainer*	34	3.736	0			
Flowmeter	35	1.51	.225			
Bottom baseplate	36	2.589	1.73			
1 grid + 12" rod + ent. K	25	2.58	3.313	.192	.867	0
.8 grid + 11" rod	3	2.58	2.64	.192	.810	0
5.4 grid + 69" rod	2	2.58	17.820	.197	5.03	.393
1.8 grid + 23" rod + exit K	1	2.58	6.135	.195	1.677	0
Exit losses	39	2.589	0			
Top baseplate	38	2.589	1.240			
<u>SEED ANNULUS</u>	40	.296	2.057	.21		
<u>LEAKAGE</u>	41	.296	.921	.21		
	42	.296	5.775	.21		
	43	.296	1.967	.21		
	44	.296	1.49	.21		
<u>TYPE-I BLANKET</u>						
Flowmeter	46	.3087	.258			
Bottom orifice	47	.713	3.85			
Bottom baseplate	48	.50766	.515			
4 grids	49	1.060	5.592			
20% Rod length + ent. K	6	1.060	.018	.246	1.703	.0127
60% rod length	5	1.060	0	.246	5.109	.1142
20% rod length + exit K	4	1.060	.343	.246	1.703	0
4 grids	50	1.060	5.592			
Top baseplate	51	.5024	1.155			
Top orifice	52	.710	4.05			
<u>TYPE-II BLANKET</u>						
Flowmeter	53	.4975	.245			
Bottom orifice	54	.408	4.00			
Bottom baseplate	55	.3058	.559			
4 grids	56	.618	5.592			
20% rod length + ent. K	9	.618	.018	.246	1.703	.0074

*Loss coefficient is contained in flowmeter.

TABLE 2 (continued)

	HAFMAT Branch Number	Total Flow Area for Core Region, ft ²	Loss Coeff. K Nom.	Hydraulic Diameter D _H (Inches)	Friction Length L (feet)	Power Fraction
<u>TYPE-II BLANKET</u>						
60% rod length	8	.618	0	.246	5.109	.0667
20% rod length + exit K	7	.618	.343	.246	1.703	0
4 grids	57	.618	5.592			
Top baseplate	58	.2805	1.143			
Top orifice	59	.398	4.25			
<u>TYPE-II P.F. BLANKET</u>						
Flowmeter	53	.4975	.245			
Bottom orifice	60	.720	3.60			
Bottom baseplate	61	.57582	.488			
4 grids	62	.936	3.620			
20% rod length + ent. K	12	.936	.014	.330	1.703	.0076
60% rod length	11	.936	0	.330	5.109	.0680
20% rod length + exit K	10	.936	.209	.330	1.703	0
4 grids	63	.936	3.620			
Top baseplate	64	.5424	1.025			
Top orifice	65	.740	3.51			
<u>TYPE-III BLANKET</u>						
Flowmeter	66	1.1667	.245			
Bottom orifice	67	.613	3.95			
Bottom baseplate	68	.4077	.49			
4 grids	69	.883	5.592			
20% rod length + ent. K	15	.883	.018	.246	1.703	.0106
60% rod length	14	.883	0	.246	5.109	.0954
20% rod length + exit K	13	.883	.343	.246	1.703	0
4 grids	70	.883	5.592			
Top baseplate	71	.362	1.113			
Top orifice	72	.605	4.20			
<u>TYPE-III P.F. BLANKET</u>						
Flowmeter	66	1.1667	.245			
Bottom orifice	73	2.15	3.51			
Bottom baseplate	74	1.602	.432			
4 grids	75	2.744	3.620			
20% rod length + ent. K	18	2.744	.014	.330	1.703	.0221
60% rod length	17	2.744	0	.330	5.109	.1993
20% rod length + exit K	16	2.744	.209	.330	1.703	0

TABLE 2 (continued)

	HAFMAT Branch Number	Total Flow Area for Core Region, ft ²	Loss Coeff. K Nom.	Hydraulic Diameter D _H (Inches)	Friction Length L (feet)	Power Fraction
<u>TYPE-III P.F. BLANKET (cont'd)</u>						
4 grids	76	2.7440	3.620			
Top baseplate	77	1.611	.957			
Top orifice	78	2.21	3.51			
<u>BLANKET LEAKAGE</u>						
	27	.112	6.5			
	29	1.0	17.6			
	88	1.0	33.4			
	141	1.0	22.8			
	143	1.0	35.3			
	145	1.0	20.6			
	26	.183	2.9			
<u>TYPE-IV REFLECTOR</u>						
Bottom orifice	79	.4015	4.794			
Bottom baseplate	80	2.727	1.176			
6 grids	81	2.727	8.34			
100% rod length + ent.K + exit K	19	2.727	.626	.267	8.515	.0020
Top baseplate	83	2.727	1.423			
Top orifice	84	.4391	6.021			
<u>TYPE-V REFLECTOR</u>						
Bottom orifice	85	.1829	4.986			
Bottom baseplate	86	1.273	1.370			
6 grids	87	1.376	8.34			
100% rod length + ent. K + exit K	20	1.376	.626	.267	8.515	.0009
Top baseplate	89	1.273	1.810			
Top orifice	90	.1808	5.057			
<u>CORE LEAKAGE</u>	132	1.0	2.316			
<u>CROSSFLOW</u>						
	28	.1	1.0			
	82	.1	1.0			
	100-131	1.0	1.0			
	133-136	.65	1.5			
	140	.1	1.0			
	142	.1	1.0			
	144	.1	1.0			
	146	.1	1.0			

TABLE 2 (continued)

	<u>HAFMAT Branch Number</u>	<u>Total Flow Area for Core Region, ft²</u>	<u>Loss Coeff. K Nom.</u>	<u>Hydraulic Diameter D_H (Inches)</u>	<u>Friction Length L (feet)</u>	<u>Power Fraction</u>
<u>OUTLET PLENUM</u>						
Lower slot BST	45	6.15	5.59			
Sid Plate	92	4.97	.04			
Hold down barrel (lower holes)	91	4.97	.1715			
Hold down barrel (upper holes)	93	4.97	.4468			
Core Barrel	94	4.97	.0292			
Outlet Nozzle	37	4.97	.375			
<u>HOT LEG</u>						
Loop A	21	1.225	5.548	15		-.25
Loop B	22	1.225	9.486	15		-.25
Loop C	23	1.225	9.486	15		-.25
Loop D	24	1.225	5.548	15		-.25
<u>COLD LEG</u>						
Loop A	96	1.225	2.651			
Loop B	97	1.225	1.406			
Loop C	98	1.225	1.406			
Loop D	99	1.225	2.651			
Inlet Nozzle	95	1.1	1.5			
Inlet Nozzle	139	1.1	1.5			
Inlet Nozzle	138	1.1	1.5			
Inlet Nozzle	137	1.1	1.5			
<u>INLET PLENUM</u>						
Ext. Branch	147	4.4	0			
Baffle plate	30	4.4	.32			
1/2 thermal shield	31	4.4	5.15			
1/2 thermal shield	32	4.4	5.15

FIGURE 16. FULL-SCALE MODULE TEST (REM-3)

analytical predictions with the analytical techniques being confirmed by REM-3 testing. Uncertainties were applied to all coefficients and are generally on the order of ± 5 to ± 10 percent depending on the amount of testing performed and dimensional tolerances.

The LWBR fuel rod axial and time dependent crud thickness distribution allowances are given in Table 3. The position of the movable seed modules is expected, based on irradiation test data indicating enhanced deposition on out-of-core surfaces, to result in thicker crud deposition on approximately the lower and upper 20 percent (2 feet) of the fuel rod length which at times are below or above the active core. Crud is assumed to build up linearly over the first 2000 Effective Full Power Hours, EFPH, to 30 percent of early-in-life equilibrium values. This rate of increase is based on prior experience for PWR-2 in the Shippingport plant. At 2000 EFPH equilibrium deposition is assumed to have been reached. These equilibrium values are utilized from 2000 EFPH to 10,000 EFPH, whereupon a step change to late-in-life equilibrium values (reflecting fuel position influence) is made for use in evaluations at 10,000 EFPH and beyond (see Table 3).

The beginning-of-life crud-free HAFMAT parameters listed in Table 2 are assumed to change with lifetime as a result of crud as follows:

- a. The crud thicknesses listed in Table 3 are used to increase fuel rod diameters and hence reduce the fuel-rod-coolant flow area.
- b. Coolant channel friction factors, determined using 1.1 times Moody values, are increased to reflect enhanced roughness resulting from crud buildup.

TABLE 3
LWBR CORE CRUD THICKNESS (MILS) AS A FUNCTION OF AXIAL POSITION AND CORE LIFETIME

						Equilibrium BOL	Equilibrium EOL
Core EFPH	Precritical	0	100	1000	2000	2000 to 10,000	$\geq 10,000$
% Crud Buildup	0	10	11	20	30	100	100
Seed	T	0	0.07	0.08	0.15	0.23	0.8
	M	0	0.02	0.03	0.05	0.07	0.2
	B	0	0.15	0.17	0.30	0.45	1.5
Blanket	T	0	0.17	0.19	0.34	0.51	1.7
	M	0	0.02	0.03	0.05	0.07	0.2
	B	0	0.08	0.09	0.17	0.26	1.7
PF Blanket	T	0	0.24	0.27	0.49	0.73	2.4
	M	0	0.02	0.03	0.05	0.07	0.2
	B	0	0.12	0.13	0.24	0.37	1.2
Reflector	T	0	0.74	0.82	1.48	2.22	7.4
	M	0	0.03	0.03	0.05	0.08	0.3
	B	0	0.37	0.41	0.74	1.11	3.7

NOTES: T = Top of fuel rod region, 20%
M = Middle of fuel rod region, 60%
B = Bottom of fuel rod region, 20%
BOL = Beginning of life
EOL = End of life

c. The crud deposits are assumed to increase the rod support grid pressure loss coefficients as follows:

EFPH	Multiplier on Grid Pressure-Loss Coefficient
0	1.02
100	1.022
1000	1.04
2000	1.06
<u>> 2000</u>	1.20

d. Baseplate and orifice-plate hydraulic loss coefficients are assumed not to be influenced by crud based on PWR operating experience.

The primary-loop pump operating characteristics also are specified in the LWBR hydraulic model. The full-speed head characteristics of the four main coolant pumps are tabulated below and are based on vendor test data and PWR operating experience.

TABLE 4: LWBR FULL-SPEED PUMP OPERATING CHARACTERISTICS

Flow (gpm)	Minimum (feet)	Pump Head Nominal (feet)	Maximum (feet)
0	522.1	535.4	548.8
8,350	567.2	579.3	591.5
12,840	538.1	550.1	562.1
15,410	498.3	509.0	519.7
17,980	446.7	456.3	466.0
18,620	433.4	442.8	452.1
19,260	419.4	428.2	437.0
20,550	389.8	397.4	404.9
23,110	323.3	329.6	335.8

Table 5 summarizes the direction in which the pump-head tolerances as well as the component pressure-loss-coefficient tolerances are assumed in defining the limiting design conditions. There are two basic design conditions: one to determine the flow distribution for thermal limit evaluations and one to determine reflector shell and guide tube pressure loads for structural analyses. The tolerances assumed for the thermal limit evaluation are selected to minimize the pressure drop forcing function

Table 5
Limiting Values of HAFMAT Model Parameters

ITEM	Limiting Design Cases											
	Thermal			Structural								
	Orifice Setting			Reflector Shell Load			Guide Tube Load With Movable Fuel Fulldown			Guide Tube Load with Movable Fuel Fullup		
	100 EFPH			EOL			EOL			EOL		
Crud Pump Head Curve	min			max			max			max		
Loop Loss Coeff.	max			min			min			min		
Reactor Inlet Loss Coefficient	max			min			min			min		
Baffle Loss Coef. and Thermal Shield	max			min			min			min		
	Seed	Blkt.	Ref1.	Seed	Blkt.	Ref1.	Seed	Blkt.	Ref1.	Seed	Blkt.	Ref1.
Strainer Loss Coefficient	max	--	--	max	--	--	max	--	--	min	--	--
Flow Meter Loss Coefficient	max	max	--	max	min	--	max	min	--	min	max	--
Orifice Loss Coefficient	--	max	max	--	min	T(B)min*	--	T-max	max	--	T-min	max
Base Plate Loss Coefficient	max	max	max	max	min	B(T)max	max	B-min	max	max	B-max	max
Rod & Grid Loss Coefficient	min	min	min	max	max	min	max	T-max	max	max	T-min	max
Other Exit Loss Coefficient	max	--	--	max	max	min	max	B-min	max	max	B-max	max
Outlet Loss Coef.	max	--	--	max	--	--	max	max	max	max	max	max
Seed Annulus Loss Coefficient	max	--	--	max	--	--	max	--	--	max	--	--
Leakage Loss Coef.	max	--	--	max	--	--	max	--	--	max	--	--
BIF Loss Coef.	max	--	--	max	--	--	max	--	--	max	--	--
Inlet Water Temp.	max	--	--	max	--	--	max	--	--	max	--	--
Core Power	max	--	--	max	--	--	max	--	--	max	--	--

T = Top B = Bottom

*The first value applies to Type V module wherein shell ΔP is maximized at bottom end, the second value to Type IV wherein ΔP is maximized at top of shell.

across the fuel region (e.g. minimum pump head, maximum loop losses, maximum module parasitic losses such as orifice plates and baseplates, and minimum fuel region losses). This pressure-drop forcing function is further described in Section III.A.4. Tolerances for the three structural cases are selected to maximize the pressure differential across the structural component (e.g. maximum pump head, minimum loop losses and core losses arranged appropriately as identified in Table 5). Results of the LWBR HAFMAT analyses are presented in Section IV. Minor modifications to the primary HAFMAT model described above were at times incorporated for special sensitivity, detailed-effects or as-built studies.

3. Transient Hydraulic Model

Hydraulic analysis of transients was performed using the PARK program with a model of the LWBR core and plant. The program, Reference 9, is designed to analyze the steady-state and transient performance of an entire nuclear reactor power plant. It is capable of simulating the overall reactor plant dynamic response to changes in plant conditions such as turbine-generator load changes, motion of the movable fuel assemblies, and changes in main coolant pump speed or primary coolant flow rate. Finite-differencing methods together with a momentum-forced solution technique are employed for solving the one-dimensional conservation equations of mass, energy, and momentum.

The PARK program was used to assess the flow decay in the fuel regions for imposed flow coastdowns representing loss-of-coolant-flow events, as described in Reference 6. For this purpose, the computational aspects of the program and reactor model include:

(a) determination of the initial steady-state condition for the plant;

(b) description of the LWBR core in terms of four parallel coolant flow regions (seed, standard blanket, power-flattening blanket, and reflector);

(c) representation of core bypass flowpaths;

(d) incorporation of four primary coolant loops with piping and valves, steam generators, main coolant pumps, check valves, and pressurizer; and

(e) calculation of coolant flow rates in all branches of the primary side hydraulic circuit based upon fundamental conservation laws, specification of the core and plant hydraulic resistances, and specification of the pump head versus flow and speed versus time characteristics.

The reactor hydraulic description for PARK is similar to that used for HAFMAT but includes less geometric detail. The hydraulic resistances are consistent with those in HAFMAT and appropriate flow areas and volumes are specified. The calculated steady-state flows and pressure drops for the primary coolant loops and principal core flow paths match the results obtained from HAFMAT with good accuracy (approximately one percent). Maximum pressure-loss coefficients and minimum pump-head characteristics are used to obtain a lower flow solution for the reactor system.

Since the HAFMAT model provides only steady-state solutions, the PARK program is used to obtain the flow coastdown for loss-of-flow transients. The transient flow versus time is obtained in accordance with the pump affinity laws, Reference 14, by specification of pump speed versus coastdown time. The pump speed characteristic was derived from an analysis of the electrical power distribution system for the LWBR reactor plant, considering electromechanical devices such as the main coolant pumps, their associated flywheel generators, circuit breakers, transformers, and the turbine generator.

A loss-of-flow event can be caused by the closure of a main coolant isolation valve, the electrical or mechanical failure of a main coolant pump, or the loss of electrical power to one or more main coolant pumps. A flywheel generator is provided for each reactor coolant pump to ensure that the rate of flow during coastdown is adequate for removal of heat

remaining in the fuel rods following reactor scram. The action of the flywheel generator is incorporated into the pump speed characteristic. The details concerning the causes of loss-of-flow events and their evaluation are given in Reference 6. The flow coastdown results for principal loss-of-flow transients are presented in Section IV.A.3. These flow coastdowns (transient decay flow forcing functions) are used with the XITE program to generate a hot-channel transient pressure drop for thermal evaluation, as described in Section III.A.4.

4. Hot-Channel Pressure-Drop Forcing Function

Thermal analysis computer programs and hot-channel models of a fuel region (e.g., seed or blanket) are used to evaluate thermal capability for a given input pressure drop and power distribution. The procedure is described in Reference 3. The axial pressure drop imposed across the nominal fuel region is determined from a core flow distribution analysis using the HAFMAT model and analysis techniques previously discussed in Section III.A.2.

The local hot-channel flow and pressure drop within each fuel region are coupled to the calculated average nominal-channel thermal-design flow and lower-bound axial pressure drop according to the equation:

$$\Delta P_{Total}^{HC} = \frac{\Delta P_{Total}^{ANC} - \Delta P_{Elev}^{ANC}}{(F_p)^2 (F_c)^2} + \Delta P_{Elev}^{ANC} \quad (Eq. 1)$$

where

- ΔP_{Total} = pressure drop (including elevation),
HC = hot channel (for XITE or HOTROD thermal analysis programs, Ref.3),
ANC = average nominal channel (for a given fuel region based on HAFMAT analysis),
 ΔP_{Elev} = elevation pressure drop,
 F_p = flow maldistribution reduction factor, and
 F_c = flow-reduction factor for preferential crudding.

The flow-maldistribution factor accounts for potential non-uniformity in flow distribution within a given fuel region. The factor has values of 1.035 (seed region), 1.04 (blanket regions), and 1.07 (reflector region). These values were shown to be conservative by measurements obtained from air-flow tests which used a simulated 0.4-scale model of the LWBR core in a manner similar to the testing described in Reference 13. The flow-reduction factor for preferential crudding varies from 1.005 to 1.050 according to the time-dependent crud model, Reference 6, as shown in Table 6.

Table 6.
Preferential-Crud Flow-Reduction Factors

<u>EFPH</u>	<u>Percentage Crud Buildup</u>	<u>Preferential Crud Factor</u>
0	10	1.005
100	11	1.0055
1000	20	1.010
2000	30	1.015
<u>> 2000</u>	100	1.050

The core flow distribution and resulting nominal region pressure drops, as predicted by HAFMAT and modified by Equation 1, are used to produce a core-lifetime-dependent, pressure-drop forcing function for the hot channel in each of the four primary fuel regions of the core. The forcing function is applied to the XITE or HOTROD hot-channel models for each core region to establish the local coolant conditions and power to reach a thermal limit. The most important operational transients are the overpower and loss-

NOTATION:

[Digital Computation Symbol] = DIGITAL COMPUTATION
 HC = HOT CHANNEL
 ANC = AVERAGE NOMINAL CHANNEL
 SUBSCRIPT O = STEADY STATE
 † = TIME
 ΔP = PRESSURE DROP
 G = MASS VELOCITY
 $\frac{P(t)}{P_0}$ = NORMALIZED POWER TRANSIENT
 ** REFERENCES 15 AND 16

$\frac{W(t)}{W_0}$ = NORMALIZED MASS FLOW COASTDOWN

$\frac{\phi(t)}{\phi_0}$ = NORMALIZED HEAT FLUX COASTDOWN

$\frac{\Delta P(t)}{\Delta P_0}$ = NORMALIZED TRANSIENT-DECAY PRESSURE-DROP FORCING FUNCTION

* TO ALLOW FOR FLOW-MALDISTRIBUTION AND PREFERENTIAL-CRUD FLOW-REDUCTION FACTORS

FIGURE 17. COMPUTER FLOW DIAGRAM FOR LOSS-OF-COOLANT-FLOW ANALYSIS

of-coolant-flow accidents. Since the reactor flow remains constant for an overpower condition, the hot-channel pressure drop as derived above is the appropriate forcing function. In the case of loss of coolant flow, a transient-decay pressure drop is determined for the hot channel using the steady-state hot-channel pressure drop and a normalized transient-decay pressure-drop forcing function. The transient forcing function is obtained from a complementary, nominal-channel XITE calculation using as input the region thermal design flow obtained from the HAFMAT analysis and the transient flow coastdown obtained from the PARK analysis discussed in Section III.A.3. A two-percent diversion of core flow to the bypass inlet flow system occurring during a scram initiation was included in the nominal channel model. Figure 17 illustrates the procedure for evaluating the pressure-drop forcing function for analysis of limiting operational accidents.

B. Bypass Inlet Flow (BIF) System Performance

1. Analysis Goals

A calculational model solving the hydraulic flow network and the equations of motion for the movable-fuel/BIF system was established for use in sizing of BIF components (e.g. balance piston and buffer cylinder diameters and labyrinth configurations) and in evaluation of the steady-state and dynamic characteristics of the movable fuel system relative to its performance requirements. The model accounts for the upward hydraulic forces due to coolant flow through the seed fuel-rod assembly, the steady-state and transient hydraulic forces exerted on the balance piston in the BIF flowpath, the weight of the movable fuel assembly, the buoyant force, the hydrodynamic mass of coolant in the BIF supply line, hydromechanical drag on the leadscrew during scram, and mechanical friction between potentially contacting components resisting motion of the movable assembly. The performance criteria are outlined as follows.

a. Net Force

To ensure that the movable fuel will always move downward and/or remain in the full-down position when released from the control drive mechanism, the minimum net force on the movable fuel assembly must act in the downward direction. The maximum net downward force is limited to 2297 pounds as a design upper limit for the control drive mechanism.

b. Scram Displacement

To assure rapid negative reactivity insertion upon receipt of a scram signal with the balance piston elevated above the buffer, the motion of the movable fuel assembly must meet or exceed the displacement-versus-time performance characteristic shown in Figure 18. This characteristic is defined by a 0.350-second delay period (i.e. zero velocity for 0.350 seconds following removal of mechanism supply voltage)* followed by an instantaneous, constant (terminal) velocity of 6 inches per second. Achievement of a shorter delay time and/or a higher terminal velocity would represent a gain in scram capability.

Figure 18. Functional Requirements for Scram Displacement

*The delay period is to allow for leadscrew/mechanism unlatching time (0.200-second allowance) and movable-assembly acceleration time (0.150-second allowance).

c. Contact Velocity

To limit impact loads on movable fuel assembly and core structural components, the contact velocity with which the balance piston strikes the buffer stop at the end of a scram is required to be no more than 9 inches per second.

2. BIF Model

The analysis model for BIF system performance is presented next in summary form. A detailed description of the BIF model has been published in Reference 5; however, because of the significance of this unique system to LWBR thermal and hydraulic performance, a complementary outline of major features of the BIF model is included for completeness in this hydraulic overview report.

The hydraulic network incorporated in the BIF analysis model is depicted in Figure 19. The node points co-ordinate with locations indicated on the Figure 2 schematic representation of the BIF system. The model represents a one-twelfth segment of the LWBR core containing three parallel flowpaths, namely, the BIF system, the movable seed, and the fixed fuel (blanket/reflector/leakage) flowpaths. Primary system HAFMAT analyses provide (a) the flow rates and pressure drops for calculation of appropriate BIF model resistances in the fixed and movable fuel flowpaths and (b) the total reactor flowrate employed as the forcing function for the BIF model analyses. Resistances in the BIF system flowpath (such as in the supply tubes and crossover pipes and across the balance piston) have been determined by several hydraulic tests; see Reference 12, for example. BIF hydraulic parameters utilized in the analysis model are given in Table 7. A free-body diagram illustrating the pressure-drop and force relationships used in computer solution of the Newtonian equation of motion for the movable fuel assembly is shown in Figure 20.

FIGURE 19. FLOW NETWORK USED IN BIF SYSTEM ANALYSIS MODEL

Table 7

Parameters Used in BIF Analysis Model

<u>BIF Parameter</u>		<u>Nominal</u>
Bypass line hydraulic resistance at T=520°F*	psi/(lbs/sec) ²	.017
Bal. piston hydraulic resistance at T=520°F*	psi/(lbs/sec) ²	.62
Upper buffer hydraulic resistance at T=175°F*	psi/(lbs/sec) ²	1.00
Lower buffer hydraulic resistance at T=175°F*	psi/(lbs/sec) ²	.62
Therm.barrier hydraulic resistance at T=520°F*	psi/(lbs/sec) ²	4.78
Leak Rate	lbs/sec	2.1
Movable Fuel Assem. Weight in Air	lbs	2483
Hydrodynamic Mass	slugs	153
Dynamic Friction	lbs	163

* Hydraulic resistance varies with coolant temperature and Reynolds number.

PRESSURE DROP BALANCE :

$$P_6 - P_3 = \Delta P_{SF} + \Delta P_{MF} + \Delta P_{8-3} = \Delta P_{BYP} + \Delta P_{BP} + \Delta P_{BUF} + \Delta P_{9-3}$$

FORCE BALANCE :

$$F_{NET} = W + F_{BP} + F_{BUF} - F_{MF} - FRIC - D = M\ddot{x}$$

FIGURE 20. MOVABLE FUEL ASSEMBLY FORCE AND PRESSURE DROP RELATIONSHIPS

Qualification of the BIF analysis model was obtained by test of a complete, prototypic BIF system at reactor operating conditions. This testing was part of the LWBR REM-3 program. It achieved qualification of the BIF model which enhanced the design assurance provided by (a) the aforementioned hydraulic testing of LWBR components in the BIF bypass line and (b) specification and control of tight manufacturing tolerances on balance piston and buffer cylinder diameters. The REM-3 test was performed with full-size components representing approximately 1/12 of the LWBR core as described earlier in Section III.A.2.b.2. The REM-3 fuel rods contained copper pellets to simulate the weight of LWBR fuel pellets. Over 30,000 feet of mechanism travel, 605,000 stop-start cycles, and 600 scrams were accumulated during REM-3 testing. Performance of the REM-3 BIF system was predicted using the BIF analysis model with appropriate REM-3 input parameters (such as a movable fuel assembly weight of 2320 pounds in air). Results of the analyses are compared with REM-3 data in Figures 21 and 22. The data provided early indication that the LWBR BIF system functional requirements would be met. In addition, the agreement between predicted and measured results provided support for use of the BIF analysis model for LWBR performance evaluation.

As indicated earlier, in LWBR design applications the BIF model is used for assessment of system performance with respect to five principal design requirements; i.e., minimum and maximum net force, maximum delay time, minimum terminal velocity, and maximum contact velocity. Design biases, reflecting such factors as component dimensional tolerances and hydraulic resistance uncertainty, are applied to HAFMAT and BIF model parameters in a direction to be most restrictive for the given design determination. These conditions are shown in Table 8 for HAFMAT parameters and in Table 9 for BIF model parameters. Results of the BIF analysis performed for LWBR design evaluation are presented subsequently in Section IV.D.

FIGURE 21. MOVABLE FUEL ASSEMBLY REM-3 ACCELERATION AND BUFFERING CHARACTERISTICS - COMPARISON WITH BIF MODEL PREDICTIONS AND BIF PERFORMANCE CRITERIA-SCRAM FROM HEIGHT OF 36 INCHES

FIGURE 22. MOVABLE FUEL ASSEMBLY REM-3
SCRAM PERFORMANCE-COMPARISON WITH
BIF MODEL PREDICTIONS

Table 8
HAFMAT Input for BIF Design

Item	Minimum Net Force			Minimum Terminal Velocity and Scram Displacement			Maximum Net Force and Contact Velocity		
1. Pump Head Curve	max			max			max		
2. Loop Loss Coef.	min			min			min		
3. React.Inlet Loss Coefficient	min			min			min		
4. Baffle Loss Coef.	min			min			min		
	S	B	R	S	B	R	S	B	R
5.Strainer Loss Coef.	min	--	--	min	--	--	max	--	--
6.Flowmeter Loss Coefficient	min	max	--	min	max	--	max	max	--
7.Orifice Loss Coefficient	--	max	max	--	max	max	--	max	max
8.Baseplate Loss Coefficient	max	max	max	max	max	max	min	max	max
9.Rod and Grid Loss Coefficient	max	max	max	max	max	max	min	max	max
10.Other Exit Loss Coefficient	min	--	--	min	--	--	max	--	--
11.Outlet Loss Coef.	min			min			min		
12.Seed Annulus Loss Coefficient	max			max			min		
13.Leak.Loss Coef.	max			max			max		
14.BIF Loss Coef.	max			min			max		
15.Inlet Water Temp.	175°F			500°F			520°F		
16.Core Power	0%			130%			0%		

S - Seed Region

B - Blanket/Power-Flattening Blanket Region

R - Reflector Region

Table 9
Conditions of Limiting BIF Design* Parameters

<u>Operating Characteristics</u>	<u>Core Flowrate</u>	<u>BIF Design Parameters</u>							<u>Upper Buffer Hydraulic Resistance</u>	<u>Lower Buffer Hydraulic Resistance</u>	<u>Hydro- dynamic Mass</u>	<u>Thermal Barrier Hydraulic Resistance</u>
		<u>Balance Piston Hydraulic Resistance</u>	<u>Bypass Line Hydraulic Resistance</u>	<u>Balance Piston Diameter</u>	<u>Dynamic Friction</u>	<u>Bypass Line Leak Rate</u>	<u>Seed Flowmeter Resistance</u>	<u>Movable Fuel Resistance</u>				
Minimum scram displacement	Maximum	Minimum	Maximum	Minimum	Maximum	Maximum	Minimum	Maximum	N/A**	N/A	N/A	Maximum
Minimum accel-eration	Maximum	Minimum	Maximum	Minimum	Maximum	Maximum	Minimum	Maximum	N/A	N/A	Maximum	Maximum
Maximum contact velocity	Zero	Minimum	Minimum	Minimum	Zero	Maximum	Minimum	Minimum	Minimum	Minimum	Maximum	Minimum
Minimum net force (zero elevation)	Maximum	N/A	Maximum	Minimum	N/A	Maximum	N/A	N/A	Minimum	Maximum	N/A	N/A
Maximum net force (5 and 12 to 84-inch elevation)	Zero	Maximum	Minimum	Maximum	N/A	Zero	Maximum	Minimum	N/A	N/A	N/A	N/A

*Each row of tabulation represents the worst cases to be used in the design of operating characteristics enumerated in the first column.

**N/A = Not Applicable

IV. HYDRAULIC EVALUATION RESULTS

A. Reactor Flow and Pressure Distribution Assessments

Table 10 summarizes the HAFMAT-predicted flows and pressure drops for the LWBR thermal design and structural design cases under 4-loop operating conditions. The fuel region flows shown in the table represent predicted rodded region flows and not flowmeter flows. Three-loop flows are approximately 79 percent of the 4-loop values. The thermal design predictions are shown for three lifetimes. For the precritical, non-cruded core at a coolant temperature of 520°F, the projected total reactor flow is 30.9×10^6 pounds per hour with a core pressure drop of 45.8 pounds per square inch. The later two lifetimes, reflecting increased crud levels, result in reduced rodded region flow rates and increased pressure differentials. As shown in Section III.A.4., the thermal-design rod-region pressure differential is reduced to account for the possible effects of preferential crud and flow maldistribution. This adjusted pressure differential is in turn used to determine core power capability according to the procedures and models described in Reference 3. It is demonstrated in References 3 and 6 that the LWBR power objective is satisfactorily fulfilled; for example, results of a design-basis assessment of the most limiting reactivity insertion transient (RIT) reported in Reference 6 show a thermal power margin of 7 percent compared with a minimum margin requirement of 5 percent. Thus, it is concluded that the hydraulic as well as the thermal objectives/requirements of Section II.D.1 and Reference 3 are mutually compatible and achievable.

B. Guide Tube and Reflector Shell Pressure Distributions

Relative pressures through the seed and blanket assemblies are shown in Figures 23 and 24 for the seed assembly in full-up and full-down positions, respectively. When the seed is in a full-down position, the lateral pressure differential across the guide tube is inward (i.e., from the blanket towards the seed) with a maximum value of 27 psid. When the seed is in a full-up position, the pressure differential is outward from the seed to the blanket with a maximum value of 24.5 psid. Relative pressures through the

Table 10
HAFMAT Predictions of LWBR
Flowrates and Pressure Differentials

Thermal-Design Cases (520°F inlet temperature, zero core power)

Case No. Description	1 Pre-Critical No Crud (Clean)		2 Orifice Setting BOL Crud		3 Equilibrium Crud	
Core EFPH	0		100		<u>></u> 10,000	
Crud Buildup (percent)	0		11		100	
Region	pressure drop (psi)	flow rate (lb/sec)	pressure drop (psi)	flow rate (lb/sec)	pressure drop (psi)	flow rate (lb/sec)
Seed	45.5	2151.	46.0	2136.	48.4	2053
Rod Region	42.8	1742.	43.4	1726.	45.8	1632
Annulus	42.8	409.	43.4	411.	45.8	421
Blanket	45.8	1584.	46.2	1582.	48.6	1566
Type I	22.0	656.	22.4	655.	24.7	648
Type II	21.9	382.	22.4	382.	24.7	378
Type III	21.6	546.	22.1	546.	24.4	540
P.F. Blkt.	45.8	2767.	46.2	2757.	48.6	2741
Type II	21.8	703.	22.3	701.	24.6	696
Type III	21.0	2064.	21.5	2056.	23.8	2045
Reflector	45.8	806.	46.2	809.	48.6	826
Type IV	1.8	556.	1.9	558.	2.3	570
Type V	1.5	250.	1.6	251.	1.9	256
Leakage	45.8	1103.	46.2	1115.	48.6	1153
Core	45.8	151.	46.2	152.	48.6	156
Blkt.	45.8	952.	46.2	963.	48.6	997
BIF	45.5	175.	46.0	176.	48.4	180
Tot. Reactor	64.7	8585.	65.1	8575.	67.2	8520

Table 10 (continued)

Structural-Design Cases (520°F inlet temperature, 130 percent core power)

Case No. Description	4 Reflector Shell Load		5 Blanket Guide Tube Load with Movable Fuel Full-Down		6 Blanket Guide Tube Load with Movable Fuel Full-Up	
Core EFPH	$\geq$ 10,000		$\geq$ 10,000		$\geq$ 10,000	
Crud Buildup (percent)	100		100		100	
Region	pressure drop (psi)	flow rate (lb/sec)	pressure drop (psi)	flow rate (lb/sec)	pressure drop (psi)	flow rate (lb/sec)
Seed	56.6	2096.	58.6	2133.	58.1	2131.
Rod Region	54.1	1683.	56.1	1713.	56.0	1711.
Annulus	54.1	413.	56.1	420.	56.0	420.
Blanket	56.8	1815.	58.8	1787.	58.4	1793.
Type I	32.9	753.	33.5	741.	32.6	744.
Type II	32.8	438.	33.4	431.	32.6	433.
Type III	32.5	624.	33.2	614.	32.3	616.
P.F. Blkt.	56.8	3166.	58.8	3120.	58.4	3126.
Type II	32.7	806.	33.4	794.	32.6	796.
Type III	31.9	2360.	32.6	2326.	31.8	2329.
Reflector	56.8	916.	58.8	909.	58.4	906.
Type IV	2.8	631.	2.9	627.	3.0	625.
Type V	2.3	284.	2.5	282.	2.5	281.
Leakage	56.8	685.	58.8	678.	58.4	679.
Core	56.8	87.	58.8	88.	58.4	88.
Blkt.	56.8	598.	58.8	590.	58.4	591.
BIF	56.6	56.	58.6	57.	58.1	57.
Tot. Reactor	73.5	8734.	75.2	8684.	75.0	8691.

FIGURE 23. GUIDE TUBE STATIC PRESSURE DISTRIBUTION, SEED IN FULL-UP POSITION

FIGURE 24. GUIDE TUBE STATIC PRESSURE DISTRIBUTION, SEED IN FULL-DOWN POSITION

FIGURE 25. TYPE IV REFLECTOR SHELL STATIC PRESSURE DISTRIBUTION

FIGURE 26. TYPE V REFLECTOR SHELL STATIC PRESSURE DISTRIBUTION

reflector and blanket modules are shown in Figures 25 and 26 for the two possible design tolerance conditions previously discussed in Section III.A.2.b.2, Table 5. The maximum differential pressure (ΔP) acting inward on the reflector shell occurs at the bottom of the shell (cf. Figure 26) and is equal to 13.5 psid while the maximum outward ΔP occurs at the top of the shell (cf. Figure 25) and is equal to 16.5 psid. These two extremes would not occur together on the same reflector shell since the only difference between the data plotted on Figures 25 and 26 is the tolerance direction imposed on the top and bottom reflector orifices. The reflector-shell and blanket-guide-tube structural assessments were performed using the above ΔP 's and found to be acceptable as demonstrated in Reference 6.

C. Flow Coastdown Analyses

Loss-of-flow accidents are classified as (a) a complete loss-of-flow accident (CLOFA) involving the simultaneous loss of forced flow in all primary loops, (b) a sequential loss of flow accident (SLOFA) involving the sequential loss of forced flow in all primary coolant loops, and (c) a partial loss of flow accident (PLOFA) involving the loss of forced flow in some, but not all, primary coolant loops. Based on design of the reactor electrical distribution system and the reactor protection system, analysis (Reference 6) has shown that a CLOFA is highly unlikely for LWBR and that the SLOFA is no more limiting than the worst-case CLOFA or PLOFA. Both four-loop and three-loop flow coastdown transients have been evaluated for various postulated loss-of-flow events. The rate of flow coastdown following inadvertent loss of pumping power is reduced because each LWBR main coolant pump has an associated flywheel generator (FWG).

Figure 27 identifies the three most-limiting four-loop flow coastdowns which are as follows:

1. A 4-0 CLOFA (3 flywheel generators operating) caused by a loss of the 138-KV transmission lines.

FIGURE 27. CORE FLOW COASTDOWN CURVES PREDICTED FOR LOSS OF FLOW ACCIDENTS (FOUR-LOOP OPERATION)

2. A 4-0 CLOFA (3 flywheel generators operating) caused by loss of all 2.4-KV power.

3. A 4-3 PLOFA (without flywheel generator) caused by a mechanical failure of the pump and check valve.

For Case 1, it is postulated that the two 138-KV transmission line towers, which independently provide power to the main coolant pumps operating as electrically isolated pairs, have a simultaneous 3-phase fault on the primary side of the transformer servicing each pair (see Reference 6 for details). The 3-phase fault induces a flow coastdown "dip", due to initial loss of voltage caused by the fault, which compounds the severity of the transient. Since prime relay protection is assumed to fail, the fault is cleared by backup relays. This CLOFA, involving concurrent loss of power to all four 2.4-KV pump buses, together with the fault (the loss of prime relay protection and the additional analytical assumption of the loss of a flywheel generator from the loop with the highest flow) constitutes a very conservative accident condition for evaluation of the CLOFA capability.

For Case 2 it is postulated that catastrophic destruction of the 138-KV transformer and No. 1 station transformer removes all sources of 2.4-KV pumping power. The accident could also occur if the reactor operator simultaneously opened all four feeder breakers or all four pump breakers. For these cases there is no voltage dip, hence no flow coastdown dip, and the pumps coastdown under the combined inertia of the pump (1000 lb-ft² at pump shaft) and flywheel generators. Loss of the flywheel generator for the loop with the highest flow is also assumed in this case.

The two CLOFA cases (138-KV fault and the loss of all 2.4-KV power) are the most severe of the CLOFA cases involving four pumps operating at fast speed with flywheel generators. The 138-KV fault CLOFA has a faster flow coastdown for the first 1.5 seconds of the transient while the CLOFA with loss of all 2.4-KV power has a faster coastdown beyond 1.5 seconds as shown in Figure 27. A natural flow coastdown of all pumps without assistance from the flywheel generators is also shown in Figure 27 to enable visualization of the improvement in flow-coastdown characteristics achieved by use of flywheel generators during LWBR operation.

Case 3 is a PLOFA caused by mechanical failure (i.e., severed impeller or seized rotor) of a single pump. The normalized impeller speed (of the failed pump) is assumed to decrease from 1.0 to 0.001 in 0.15 second. The severity of this accident is compounded by the assumption that the check valve remains stuck open permitting back-flow through the loop. Since the failure is to the pump impeller, the flywheel generator provides no advantage in maintaining flow and the loop flow decays rapidly.

The loss-of-flow events represented by Cases 1, 2, and 3 were evaluated using the procedures and criteria described in Reference 3 to determine thermal margin. The procedure includes the use of the HOTROD and transient XITE thermal computer programs with (1) an initial reactor flow/-pressure distribution compatible with the steady-state HAFMAT hydraulic analysis, (2) design transient decay functions for pressure drop (Sections III.A.3 and III.A.4) and fuel-rod surface heat flux, (3) appropriate hot channel factors and design allowances, and (4) the Shippingport critical heat flux correlation. The minimum design power margin is determined assuming the hottest portion of the fuel region is operating at nine percent above core rated power when the transient occurs. The calculated design power margins at nine-percent overpower are 13, 20, and 13 percent for Cases 1, 2, and 3, respectively, compared to the minimum requirement for five percent margin. Thus the results indicate acceptable LWBR performance with respect to the objective for safe operation at rated power if a loss-of-flow accident should occur during four-loop operation.

D. Bypass Inlet Flow (BIF) System Performance Assessments

1. Net Force

Evaluation of steady-state net force on the movable fuel assembly using the previously described analysis model has shown (cf. Reference 5) the BIF system to be slightly underbalanced hydraulically; that is, typically the downward hydraulic force is slightly less than the upward hydraulic force. The minimum net force consequently occurs at maximum core resistance and flowrate. The net force is also a function of balance piston

elevation. Visualization of this position dependency is aided by Figure 28 which shows the pressure drop across the upper buffer piston would be minimized and that across the lower buffer piston would be maximized when the movable assembly is at its lowest or 0-inch elevation (the maximum engagement position between lower buffer piston and cylinder). Since the cross-sectional area of the lower buffer piston is significantly smaller than that of the upper buffer piston, the downward hydraulic force (pressure drop times area) is minimized at the 0-inch elevation and thus net downward force is a minimum at this position. A plot of minimum net force calculated for the LWBR movable fuel assembly under 4-loop full-speed pump operating conditions is shown as a function of balance piston elevation in Figure 29. The minimum net force is seen to remain in the downward direction in accord with BIF system requirements.

As a consequence of the slightly underbalanced system, the maximum net force occurs at zero flow and is equal to the maximum weight of the movable fuel assembly in water which has been predicted to be 2260 pounds. The maximum net force therefore is calculated to be less than the 2297-pound requirement and is also less than a comparable 2351 pound load under which the LWBR control drive mechanism has been life tested.

2. Scram Displacement

Based on LWBR BIF system parameters, scram displacement and terminal velocity ranges were predicted by use of the BIF analysis model. The results are shown in Figures 30 and 31. The predicted LWBR performance is seen to exceed minimum scram requirements. Terminal velocity is observed in Figure 31 to decrease with increasing core ΔP and flow. At full flow the calculated terminal velocity range is about 9 to 16 inches per second compared with a minimum acceptable value of 6 inches per second.

3. Contact Velocity

Projected performance of the movable fuel buffer was also evaluated through the BIF model. Resultant upper-bound and lower-bound contact velocities are shown in Figure 32 as a function of core pressure

FIGURE 28. BIF BUFFER OPERATION

FIGURE 29. MINIMUM NET FORCE THROUGHOUT
MOVABLE FUEL ASSEMBLY TRAVEL

FIGURE 30. SCRAM DISPLACEMENT RANGE
VERSUS TIME

FIGURE 31. TERMINAL VELOCITY PREDICTIONS

drop. The maximum predicted contact velocity of 8.8 inches per second occurs at a zero flow/pressure-drop condition; the calculated range for contact velocity at full flow is about 0.6 to 4 inches per second. These values are below the allowable limit of 9 inches per second. Typical, analytical time/displacement performance characteristics of the LWBR buffer at 0- and 100-percent core flow are illustrated in Figures 33 and 34, respectively.

Additional description of the performance of the LWBR BIF system, including buffer, scram-displacement, and net-force characteristics, is given in Reference 5.

FIGURE 32. CONTACT VELOCITY PREDICTIONS

FIGURE 33. TYPICAL BUFFER PERFORMANCE AT ZERO CORE FLOWRATE

FIGURE 34. TYPICAL BUFFER PERFORMANCE AT FULL CORE FLOWRATE

V. LWBR PRECRITICAL TEST PROGRAM HYDRAULIC RESULTS

A. Reactor Pressure Drop and Flow Characteristics

Prior to initial core criticality and with the reactor plant at 531°F and 2000 psia, LWBR in-core and primary-loop instrumentation was used to monitor seed and blanket assembly flowrates, core and BIF pressure drops, and loop flowrates under 4-loop and 3-loop fast-speed pump operating conditions. A representative listing of the measured data is presented in Table 11. The adequacy of the reactor hydraulic design was assessed by comparison of these measurements with HAFMAT analytical parameters as follows.

1. Table 12 compares the range of measured core flows*, core pressure drop, and total reactor flow** with appropriate limiting HAFMAT design predictions for 4-loop operation. As seen in Table 12, all measured flowrates are above the predicted thermal-design values while measured core pressure drop is below the predicted structural-design value.

2. Table 13 compares the as-measured seed, blanket and reactor 3-loop flow fractions*** with predicted flow fractions. The measured data are in close agreement with prediction.

Since the measured core flowrates were greater than the lower-limit (thermal-design) values and the core pressure drop was less than the upper-limit (structural-design) value, the reactor hydraulic design was concluded to be acceptable. These precritical test measurements have therefore confirmed the conservative hydraulic analyses and procedures incorporated in design performance assessments for evaluating achievement of LWBR hydraulic objectives (such as adequate coolant flowrates and satisfactory pressure distribution on core structures). LWBR performance assessment based on the as-measured hydraulic characteristics and other as-built/measured parameters indicates a thermal margin of 17 percent for the limiting reactivity insertion transient (RIT), Reference 3, compared with the previously calculated design margin of 7 percent, Reference 6.

*Seed and blanket in-core flowmeter readings.

**Sum of loop flowmeter readings.

***3-loop flow as a fraction of 4-loop flow.

Table 11
Precritical Measurements of LWBR Hydraulic Characteristics

<u>Hydraulic Characteristic</u>	<u>Measured Values*</u>	
	<u>4-Loop Operation</u>	<u>3-Loop Operation</u>
1. Seed I-1 Flow (lb/sec)	181-192	147-157
2. Seed I-2 Flow (lb/sec)	187-193	150-159
3. Seed II-1 Flow (lb/sec)	180-184	145-151
4. Seed II-3 Flow (lb/sec)	183-188	148-154
5. Seed III-1 Flow (lb/sec)	186-188	149-155
6. Seed III-2 Flow (lb/sec)	180-183	143-154
7. Seed III-3 Flow (lb/sec)	184-187	147-153
8. Blanket I-1 Flow (lb/sec)	261-262	210-212
9. Blanket I-2 Flow (lb/sec)	270-274	219-225
10. Blanket II-1 Flow (lb/sec)	462-473	370-387
11. Blanket II-3 Flow (lb/sec)	452-462	372-377
12. Blanket III-1 Flow (lb/sec)	554-568	442-470
13. Blanket III-3 Flow (lb/sec)	555-568	448-471
14. Core Pressure Drop (psi)	47.5-47.9	28.6-29.8
15. BIF II-3 Pressure Drop (psi)	45.5-45.8	26.9-28.2
16. BIF III-1 Pressure Drop (psi)	46.3-46.9	27.8-29.4
17. BIF III-3 Pressure Drop (psi)	46.5-47.2	27.9-29.6
18. Loop A Flow (lb/sec)	2311-2325	2456-2458**
19. Loop B Flow (lb/sec)	2206-2236	2358-2361**
20. Loop C Flow (lb/sec)	2211-2214	2355-2369**
21. Loop D Flow (lb/sec)	2369-2372	2467-2489**

Test Conditions: 531°F, 2000 psia, movable fuel in full-down position, full-speed pump operation.

* The indicated range covers normal measurement fluctuations; in addition, the range for 3-loop operation encompasses changes resulting from variation in isolated-loop designation.

** Flow in the specified loop with one of the remaining three loops isolated.

Table 12

Comparison of Measured and Predicted
4-Loop Hydraulic Characteristics

<u>Hydraulic Characteristic</u>	<u>Limiting Predicted Values*</u>	<u>Measured Range</u>
1. Core Flows - lb/sec/assembly		
a. Seed	177	180-193
b. Type I Blanket	253	261-274
c. Type II Blanket	410	452-473
d. Type III Blanket	510	554-568
2. Core Pressure Drop - psi	53.5	47.5-47.9
3. Total Reactor Flow - lb/sec	8471	9097-9147

*The lower-limit predicted levels for flowrates were established by the thermal-design case while the upper-limit level for core pressure drop was based on the structural-design case. The predicted flow values are taken from HAFMAT flowmeter branches to correspond with LWBR flowmeter measurements; predictions include no adjustments to allow for measurement errors. Operating parameters specified for the HAFMAT calculations were consistent with those existing at the time of the measurements (i.e. zero power, 531°F, 2000 psia, no crud, movable fuel in full-down position, fast-speed pumps).

Table 13

Measured 3-Loop Flow Fractions Compared to Predictions

<u>Region</u>	<u>3-Loop Flow as a Fraction of Full Flow*</u>	
	<u>Measured</u>	<u>Predicted</u>
Seed	.792-.837	.784-.795
Type I Blanket	.808-.822	.786-.796
Type II Blanket	.801-.822	.786-.796
Type III Blanket	.800-.827	.786-.796
Total Reactor Flow	.786-.801	.786-.796

*For the same operating conditions as indicated under Table 12.

The hydraulic parameters previously identified are being recorded at 15-minute intervals during LWBR power operation. The data is being evaluated periodically to ensure that the core hydraulic characteristics remain within acceptable bounds and that no unexpected trends are developing*.

V. B. Bypass Inlet Flow (BIF) System Performance Characteristics

BIF system hydraulic data which confirmed predicted system performance characteristics were obtained from control drive mechanism testing performed during LWBR precritical operations. The test program included individual scrams of each LWBR movable fuel assembly from a height of 53 inches and a 12-module bank scram from a height of 11 inches. The LWBR primary plant was maintained at zero-power, full-flow (4 pumps on fast speed), hot operating conditions ($T = 531^{\circ}\text{F}$, $P = 2000$ psia). Typical core and BIF-system pressure drops recorded during the test period are indicated in Table 14.

*After more than 3 years of operation and 18,000 EFPH, core flows remain at acceptable levels. Seed flow is essentially the same as measured during startup but blanket flow has decreased approximately 2 percent while core pressure drop has increased about 5 percent. The predictions discussed in Section IV.A, which include lifetime-dependent allowances for crud deposition, show a reduction of approximately 4.6 percent in seed flow, no change in total blanket flow (includes blanket leakage) and an increase in core pressure drop of about 6 percent over the same time period. The increase in seed flow and reduction in blanket flow with respect to the predictions is likely a result of less crud buildup in the seed than predicted (as indicated by the data) which caused decreased flow redistribution into the blanket and slightly enhanced deposition therein. In addition, leakages in the blanket region decrease slightly with increasing lifetime thereby decreasing total measured blanket flow. These small lifetime leakage variations were not included in the prediction model to minimize the coolant-channel pressure-drop forcing functions for thermal evaluations. It is also noted that, to reduce burden on fuel rod cladding, primary plant nominal operating pressure has been programmatically decreased over lifetime as follows.

EFPH	Pressure (psia)
0-4,254	2000
4,254-7,029	1940
7,029-10,771	1870
10,771-18,298	1815

Extended lifetime operation beyond 18,298 EFPH is in progress which entails further reduction in plant pressure (to 1615 psia) and also reductions in system average temperature (to 521°F) and in plant power levels (to 80% power or lower).

Table 14
Measured LWBR BIF and Core Pressure Drop Data

Module	Day	Time	In- or Out- of-Buffer	Core ΔP (psi)	BIF ΔP (psi)
II-3	7/19	2318	Out ↓	47.64	44.54
		2323		48.18	44.44
		2328		47.46	44.75
		2333		48.31	44.75
		2353		48.19	45.04
	7/20	0008	In ↓	48.51	44.87
		0028		48.28	46.48
		0033		48.21	46.53
		0108	Out ↓	48.10	45.77
		0113		47.74	44.84
		0118		48.37	44.74
		0123	In ↓	47.89	44.48
		0143		47.87	44.85
		0158		48.04	44.72
		0238		47.82	44.81
		0258		48.01	44.47
		0303		48.17	44.45
		0308		47.72	46.32
III-1	7/21	2249	Partway In (11") ↓	48.32	45.96
		2259		48.19	45.86
		2324		48.14	46.31
		2349		47.80	46.12
		2354		48.21	46.16
		2359		47.52	45.91
	7/22	0004	In ↓	48.46	45.86
		0044		48.52	46.02
		0125		48.05	47.44
		0129	Lifting ↓	47.64	46.99
		0144		47.95	47.08
		0209		47.79	46.00
		0214	Out ↓	48.23	46.31
		0219		48.24	46.39
		0234		48.17	46.20
		0244		47.92	46.13
III-3	7/21	1552	In Lifting ↓ Out ↓ ↓ ↓ ↓ ↓ ↓ ↓ ↓ ↓ ↓	47.44	47.13
		1557		47.68	47.47
		1602		47.78	47.20
		1607		48.10	46.25
		1612		47.89	45.72
		1617		48.07	45.57
		1622		48.04	45.23
		1627		47.81	45.08
		1647		47.91	45.37
		1657		48.03	45.50
		1702		47.97	45.50
		1727		47.60	45.16

This testing provided confirmatory, design-assurance information concerning circuitry time delays, mechanism unlatching time, movable fuel assembly acceleration time, terminal velocity, deceleration characteristics, and contact velocity.

The noise generated by the ratcheting of the mechanism out-motion latch against the leadscrew during scram was the principal parameter monitored as a function of time during test. The noise signal was detected by an accelerometer located at the top of the mechanism motor tube extension. The signal was amplified by a charge amplifier and recorded on a light-beam oscillograph high-speed recorder. A high-pass filter in conjunction with a micro-farad capacitor was used on the signal to obtain distinct peaks characteristic of the out-motion latch ratcheting. The period between signal peaks represented a downward displacement of the movable fuel equal to the leadscrew pitch of 0.584 inches. The speed of the movable fuel assembly during any portion of the scram was calculated by dividing the displacement by the elapsed time between peaks as indicated on the strip chart recording. Contact velocities were determined from the last two peaks. Terminal velocities were determined by calculating an average velocity over approximately 35 inches of motion after the acceleration phase and prior to the deceleration phase of scram.

1. Scram Displacement

Figure 35 shows the measured displacement-versus-time scram characteristics for the fastest and slowest LWBR movable fuel assemblies. The scram performance is within the predicted range for the test conditions and satisfies the BIF functional requirements. The terminal velocities attained by each movable fuel assembly are listed in Table 15. The minimum was 11.45 inches per second while the maximum was 13.03 inches per second; all were thus significantly larger than the minimum 6-inches-per-second criterion. The delay/-acceleration time to reach a velocity of 6 inches per second for the slowest accelerating assembly (III-2) was 0.217 seconds or 0.133 seconds faster than the 0.350-second design criterion. The delay time for the fastest accelerating assembly (II-3) was 0.04 seconds shorter than that for module III-2.

FIGURE 35. COMPARISON OF LWBR MOVABLE FUEL ASSEMBLY SCRAM ACCELERATION DATA WITH BEST-ESTIMATE PREDICTIONS AND FUNCTIONAL REQUIREMENTS

Table 15
Scram Data Obtained During Mechanism Hot Testing

<u>Movable Assembly</u>	<u>Terminal Velocity (in./sec)</u>	<u>Contact Velocity (in./sec)</u>
I-1	11.76	1.39
I-2	11.72	0.77
I-3	11.53	1.24
II-1	11.45	1.04
II-2	12.00	0.83
II-3	12.17	1.33
III-1	12.60	1.46
III-2	12.29	0.97
III-3	12.51	1.36
III-4	12.93	1.67
III-5	12.56	1.10
III-6	13.03	1.67

2. Contact Velocity

The contact velocities determined for each movable fuel assembly during the precritical testing are also listed in Table 15. They range from a minimum of 0.77 inches per second to a maximum of 1.67 inches per second and lie within the expected range of 0.6 to 4 inches per second (cf. Figure 25 at a core ΔP of about 48.3 pounds per square inch). Acceptable LWBR buffer performance is therefore indicated.

C. Flow Coastdown Characteristics

Flow coastdown tests were conducted prior to initial operation of the reactor. The test evaluation showed that the measured flow coastdown rates without flywheel generators were slightly faster than expected (unfavorable, as shown for the worst-case coolant loop, D, in Figure 36). The measured flow coastdown rates with flywheel generators were slower than expected (favorable, as shown in Figures 37 and 38). The net effect on total core flow coastdown in four coolant loops with one flywheel generator inoperative was slightly faster than expected (unfavorable). The input pump speed specification for each main coolant loop described by PARK was adjusted for the limiting loss-of-flow cases to reflect these favorable and unfavorable results obtained from the testing. For conservatism in adjusting the pump speed to match the data, it was assumed that loops without operating flywheel generators had the worst-case measured flow coastdown obtained from loop D and that loops operating with flywheel generators had the minimum improvement in flow coastdown (loop B). To offset the slightly unfavorable

FIGURE 36. NORMALIZED FLOW COASTDOWN FOR LOOPS B AND D WITHOUT FLYWHEEL GENERATOR

FIGURE 37. NORMALIZED FLOW COASTDOWN FOR LOOPS B AND C WITH FLYWHEEL GENERATOR

FIGURE 38. NORMALIZED FLOW COASTDOWN FOR LOOPS A AND D WITH FLYWHEEL GENERATOR

net result for a limiting four-loop CLOFA, margin accrued from other as-built/as-measured parameters (e.g., higher module steady-state flow, faster movable-fuel-assembly scram displacement and improved physics characteristics) was applied to the final thermal evaluation of loss-of-flow accidents performed prior to initial core operation at power. The results indicated adequate LWBR performance under LOFA conditions; that is, a minimum power margin of 24 percent was determined for the most limiting LOFA case, Reference 3, compared to the previously calculated design-basis value of 13 percent.

VI. SUMMARY/CONCLUSIONS

Based on qualified design analyses and confirmatory LWBR precritical test measurements described in this report, it is concluded that LWBR hydraulic performance objectives have been achieved with margin relative to limiting functional requirements. Assurance of reliable operation of LWBR in the Shippingport Atomic Power Station has thereby been provided in part by the reported assessments. Flow and pressure distributions resulting from LWBR hydraulic evaluations have been shown, considering thermal, structural and reactivity-control performance, to support adequately LWBR rated-power operation under the most-restrictive potential accident (RIT and LOFA) conditions. The hydraulic performance analyses have included determination of (a) hydraulic loads on structural shells resulting from inlet and outlet orificing of blanket and reflector fuel regions, (b) scram velocity of the hydraulically-balanced movable fuel and (c) contact velocity of the hydraulically-buffered movable fuel, in addition to evaluation of reactor flowrates and pressure drops for steady-state and transient operation. An extensive test program has provided qualification for the analysis procedures.

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