FUEL PERFORMANCE IMPROVEMENT PROGRAM

**Quarterly Progress Report
April - June 1979**

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Richland, Washington 99352**

MASTER

July 1979

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ABSTRACT

Two series of test rods are under irradiation in the Halden Boiling Water Reactor (HBWR Series H-1 and Series H-4). Fuel rods for Series H-2 and H-3 have been fabricated and delivered to Halden. Plans for the first series of demonstration fuel assemblies for irradiation in the Big Rock Point Reactor have been modified to substitute two BRPR Series S-2 assemblies containing segmented rods, some with sphere-pac fuel, for Series S-1 assemblies that will be irradiated later. These assemblies are in the current BRPR reload and reactor startup is scheduled for September-October.

An effective method for applying graphite coating to the inside surface of cladding tubes has been demonstrated. Rod loading procedures for sphere-pac fuel that assure a uniform axial fuel density have been developed. Special hardware has been designed and tested that assures the spherical fuel can be retained in the fuel column during loading and handling operations.

The measured centerline fuel temperature in the pressurized sphere-pac fuel rod (HBWR Series H-4) is lower ($\sim 145^{\circ}\text{C}$) than for the reference pellet fuel rod operating at comparable linear heat ratings.

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SUMMARY

Progress in the Fuel Performance Improvement Program (FPIP) continues to be good with no delayed goals during the third quarter of FY-1979.

Irradiation of the HBWR Series H-1 in IFA-518 and Series H-4 in IFA-517 are on schedule; the fuel rods for Series H-2 and H-3 have been fabricated and delivered to Halden. Process development has been completed and fabrication of the fuel rods for BRPR Series S-2 and S-3 has begun. Several significant achievements and results can be cited for this report period:

- The base technology for applying an acceptable graphite coating to the inside surface of fuel rod cladding tubes has been demonstrated by a technique that is adaptable to production quantities.
- Fabrication of the fuel rods for HBWR went smoothly and they were delivered on time.
- A rod loading procedure was developed for sphere-pac that assures the desired 86.5-87% TD to the top of the fuel column.
- Special hardware in the form of a retainer disc has been designed and tested that assures the microspheres in sphere-pac fuel are retained in the fuel column during loading and handling.
- The measured centerline fuel temperature in the pressurized sphere-pac rod (HBWR Series H-4) is lower ($\sim 145^{\circ}\text{C}$) than for the reference rod and significantly lower ($\sim 345^{\circ}\text{C}$) than for a vipac rod operating at comparable linear heat ratings.
- A major program change that required an accelerated schedule for fabrication, licensing and shipment of three BRPR Series S-2 fuel assemblies to meet the projected September-October startup at Big Rock Point Reactor was incorporated into this quarter's activities without interfering with scheduled progress.

Summaries of activities according to individual task summaries have been collected below to provide a convenient synopsis of those activities for the reporting period.

Task 2A - Out-of-Reactor Experiments

A modified method for controlling the thickness of Dag 4 graphite coatings applied to the inside of zircaloy cladding has been used for the BRPR Series S-2 and S-3 rods with good results. The important control factors are a minimum drain time of ten minutes and a batch by batch coating thickness check and adjustment. Coatings obtained by this method probably represent the best that is reasonably attainable by the flood coating technique as well as the limit of the present capability to measure coating thickness. Further, it is considered that the base technology for applying an acceptable graphite coating to the inside surface of fuel rod cladding tubes has been demonstrated by a technique that is adaptable to production quantities.

The friction and wear test results show that although both types of graphite coatings tested exhibit reduced friction and wear when compared to noncoated zircaloy surfaces, it is doubtful that either graphite would function effectively as a lubricant over significant sliding distances in the environment that exists within an operating fuel rod. These indications suggest that comparative results obtained by this testing method may not be relatable to the observed capability of graphite coatings to improve FCI behavior.

Coating function studies have been scoped. Three types of tests are planned; time-to-failure; comparison; and characterized surface.

The fuel cladding interaction (FCI) simulator has been assembled and preliminary tests to evaluate mechanical characteristics of this device are being performed.

Task 2B - In-Reactor Experiments

The screening tests are being irradiated in two test rigs in the HBWR. The IFA-518 test rig contains two six-rod clusters and the power in the two clusters is varied by alternately shifting a silver absorber

shield over the two clusters. The IFA-517 test rig contains a single cluster of four rods. The peak operating linear heat generation rate in both assemblies is 44.5 kW/m.

Rod instrumentation in both assemblies includes fuel centerline thermocouples, elongation sensors, and internal pressure sensors. IFA-518 will be sequentially loaded with rods comprising HBWR Series H-1, H-2, and H-3, respectively. Peak burnups will be in the range of 1200 to 1400 GJ/kg (14 to 16 MWd/kg).

The HBWR Series H-4 rods will be irradiated in IFA-517 to peak burnups of about 430 GJ/kg (5 MWd/kg). The IFA-517 rig will also be used later for the power-ramping tests.

Fabrication of the fuel rods for HBWR Series H-2 and H-3 went smoothly and they were completed on time. The fuel rods were X-rayed, etched, neutron radiographed and shipped to Norway in April 1979. Profilometry was performed at the Kjeller Laboratory in Norway.

Task 2C - In Reactor Demonstrations

The schedule for fabrication, licensing, and shipment of the three BRPR Series S-2 fuel assemblies was accelerated to meet the projected September-October startup at Big Rock Point Reactor. Plans are to substitute the two BRPR Series S-2 assemblies which contain segmented rods, some include sphere-pac rods, for two of the four BRPR Series S-1 assemblies (which contain some vipac rods). Results of the neutronics analysis show that the Series S-1 and S-2 assemblies are virtually identical with respect to power peaking.

For purposes of economy, all of the rods required in the BRPR Series S-2 and S-3 are being fabricated with BRPR reload batch H-1 (Cycle 16 reload). Details and bundles maps for Series S-2 assemblies were shown in the January-March 1979 Quarterly Progress Report.^(1.10)

Two documents required for fabrication and qualification of Series S-2 and S-3 fuel rods are on schedule:

- The parts list for BRPR Series S-2 and S-3 fuel rods, XN-NF-PL-15, was approved with one minor drawing change requested.

- The Irradiation Test Authorization (XN-NF-ITA-1) for BRPR Series S-2 and S-3 demonstration fuel rods was approved and issued and sent to CPC.

The sphere-pac fuel retainer disc with an interference fit has been shown to prevent microsphere intrusion under the lower insulator disc and also to prevent microspheres from falling into the upper plenum during horizontal rod handling. The retainer disc was modified to have an open center hole of 5.56 mm (0.219 in.) thus increasing disc flexibility and decreasing contact with hot fuel. In tests to determine the effect of the disc on the inside surface of the cladding, results showed that normal internal cladding oxidation did not cause any galling or seizing between the retainer and cladding.

The rod loading procedure for sphere-pac fuel was modified to permit overfilling followed by removal of the excess fuel column length by aspiration after the fuel column had been consolidated by vibration. This technique avoids a low, 70-80% TD, in the top 2.5 mm of the column. This technique was also used to remake a fuel rod that had a low density region.

Fabrication of reference and annular pellets for BRPR Series S-2 and S-3 is complete as is graphite coating of the cladding.

Task 3 - Performance Evaluation

The Series H-1 fuel rods under irradiation in the Halden Boiling Water Reactor (HBWR) assembly IFA-518 include 12 fuel rods with six of these rods hexagonally arrayed in an upper cluster and the remaining six rods in the lower cluster. Four fuel forms are included in HBWR Series H-1: reference, annular, annular-coated, and vipac. A movable neutron absorber shield covers one of the clusters at any one time allowing the power of the shielded cluster to be reduced approximately 40%. The IFA-518 assembly is instrumented with neutron detectors, fuel centerline thermocouples, cladding elongation sensors and internal pressure transducers.

As part of the in-reactor experiments, calculations of rod-average and local linear heat generation rates (LHGR) are made to compare performance of the various fuel designs. Errors in these calculations were noted after the first shield shift on September 26, 1978 when the silver shield was lowered from the upper to the bottom cluster.

The shield down position caused the calculated LHGRs at the thermocouple (LHRTF) to be lower for a given temperature than when the shield was up. The error in reference fuel was estimated to be ~18%. A slightly smaller error in LHGR was noted for annular-coated fuel and may be compared with reference fuel using Figures 7.1 and 7.2. The cause of this error and any other errors in the power equations was determined so that an accurate comparison can be made between the performance in each fuel rod.

Task 4 - Establishment of Technical Bases

A DOE supported task of the Fuel Refabrication and Development Program evaluated fuel properties for several fuel compositions and forms. The reference file structure for this work was compatible with that used in FPIP so evaluation of vipac and sphere-pac fuel properties are being incorporated into the FPIP reference file.

Theoretical results for evaluating thermal conductivity of sphere-pac fuel have been obtained that outline the different aspects of the dependence of thermal conductivity on key parameters.

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1.0 INTRODUCTION

C. E. Crouthamel (Exxon Nuclear Company)

1.1 PROGRAM OBJECTIVES

The objectives of the Fuel Performance Improvement Program (FPIP) are to identify and demonstrate fuel concepts with improved performance and to provide the supportive technical bases for developing commercial fuel designs that are capable of achieving high burnup for better utilization of uranium.

1.2 BACKGROUND

The Fuel Performance Improvement Program focuses on improving the fuel-cladding interaction behavior of both BWR and PWR light-water reactor fuels. A complete listing of reports previously published under the auspices of the FPIP is given at the end of this section.

1.3 PROGRAM DESCRIPTION

The program description and history have been documented in the previous quarterly reports^(1.4, 1.5) which also give the task structures and brief descriptions of each task.

1.4 PROGRAM ORGANIZATION

The program is performed by Consumers Power Company (CPC), Exxon Nuclear Company (ENC), and Battelle, Pacific Northwest Laboratory (PNL). CPC serves as the prime contractor for the U.S. Department of Energy (DOE)--most of the work has been subcontracted to ENC and PNL. Figure 1.1 illustrates the functional program organization by task.

As prime contractor, CPC has overall responsibility for directing the program. This is accomplished through a steering committee under the chairmanship of the program director. CPC has designated ENC to manage a program office for the execution of the program subject to policy directions and review by the steering committee. In addition,

ENC is responsible for the design, process development, and fabrication of fuel concepts under study.

PNL is primarily responsible for performing the research and development portion of the program and the interpretation of results.

The following sections of the report present the progress on program tasks.

FIGURE 1.1. Program Organization of the Fuel Performance Improvement Program

1.5 PREVIOUSLY PUBLISHED FPIP REPORTS

- 1.1 "Fuel Performance Improvement Program, Quarterly Progress Report, April-June 1977," compiled by C. E. Crouthamel, C00-4066-1, July 1977.
- 1.2 W. J. Bailey, C. L. Wilson, L. J. MacGowan, and P. J. Pankaskie, "State-of-the-Technology Review of Fuel Cladding Interaction," C00-4066-2, PNL-2488, December 1977.
- 1.3 W. J. Bailey and J. O. Barner, "Assessment of Fuel Concepts," C00-4066-3, PNL-2490, January 1978.
- 1.4 "Fuel Performance Improvement Program, Quarterly/Annual Progress Report, April-September 1977," compiled by C. E. Crouthamel, C00-4066-4, November 1977.
- 1.5 "Fuel Performance Improvement Program, Quarterly Progress Report, October-December 1977," compiled by C. E. Crouthamel, C00-4066-5, January 1978.
- 1.6 "Fuel Performance Improvement Program, Quarterly Progress Report, January-March 1978," compiled by C. E. Crouthamel, C00-4066-6, April 1977.
- 1.7 "Fuel Performance Improvement Program, Quarterly Progress Report, April-June 1978," compiled by C. E. Crouthamel, C00-4006-7, July 1978.
- 1.8 "Fuel Performance Improvement Program, Quarterly/Annual Progress Report, October-September 1978," compiled by C. E. Crouthamel, C00-4066-8, October 1978.
- 1.9 "Fuel Performance Improvement Program, Quarterly Progress Report, October-December 1978," compiled by C. E. Crouthamel, C00-4066-9, January 1979.
- 1.10 "Fuel Performance Improvement Program, Quarterly Progress Report, January-March 1979," compiled by C. E. Crouthamel, C00-4066-10, April 1979.
- 1.11 S. R. Wagoner, J. O. Barner and R. K. Welty, "Fuel Performance Improvement Program, Description and Characterization of HBWR Series H-1 Test Rods," C00-4066-11, June 1979.*

* In process, not approved for release.

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2.0 TASK 1 - ASSESS FUEL - CLADDING INTERACTION STATE-OF-THE-TECHNOLOGY
AND POTENTIAL IMPROVED CONCEPTS

W. J. Bailey (Pacific Northwest Laboratory)

The objective of Task 1 was to assess the state-of-the-technology of fuel cladding interaction and to identify fuel concepts with potential for improved performance capabilities in order to provide direction and technical support for the Fuel Performance Improvement Program. With the completion of the topical reports entitled, "State-of-the-Technology Review of Fuel - Cladding Interaction" (COO-4066-2/PNL-2488) and "Assessment of Fuel Concepts" (COO-4066-3/PNL-2490), activities associated with this task were completed.

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3.0 TASK 2 - TESTING AND DEMONSTRATION PROGRAM

M. D. Freshley (Pacific Northwest Laboratory)

The objective of the testing and demonstration program, which forms the basis for the Fuel Performance Improvement Program (FPIP), is to conduct a comprehensive, well balanced testing program to provide the data needed for establishing technical design bases for several fuel concepts with improved performance capabilities. The testing and demonstration program includes Task 2A, Out-of-Reactor Experiments; Task 2B, In-Reactor Experiments; and Task 2C, In-Reactor Demonstrations.

Task 2A activities provide out-of-reactor supportive data needed for the Task 2B irradiation experiments in the Halden Boiling Water Reactor (HBWR) and the Task 2C demonstration irradiations in the Big Rock Point Reactor (BRPR). Task 2B involves an in-reactor testing program that consists of screening tests conducted in the HBWR followed by ramping of individual rods in the HBWR ramping rig; and primary tests that consist of the irradiation of segmented rods to significant burnups in BRPR followed by ramping in a suitable test reactor. Task 2C activities involve the design, process development, fabrication, quality control, and licensing of a significant number of fuel rods comprising the primary fuel concepts for demonstration irradiations in BRPR.

3.1 TESTING PROGRAM STATUS SUMMARY

Coordination of subtask activities is continuing in order to maintain scheduled progress. Activities are currently focused on graphite coating studies and transferring the coating application technology to a production process; friction and wear testing of different coating materials; coating function and FCI simulation studies; data reduction and evaluations in connection with the irradiation of the HBWR Series H-1 (IFA-518) and Series H-4 (IFA-517) experiments; completion of fabrication of the fuel rods for HBWR Series H-2 and H-3; and process development and fabrication of fuel rods for BRPR Series S-2 and S-3.

The fuel rod matrix developed for the HBWR Series H-1, H-2, H-3 and H-4 irradiation tests in HBWR as part of Task 2B and the fuel rod matrices developed for BRPR Series S-2 and S-3 demonstration irradiations in BRPR as part of Task 2C were presented previously.^(1.9)

4.0 TASK 2A - OUT-OF-REACTOR EXPERIMENTS

L. R. Bunnell, D. D. Marchant, C. L. Mohr, and K. M. Olson
(Pacific Northwest Laboratory)

The purpose of this task is to conduct application-oriented out-of-reactor studies in support of the development and demonstration of fuel concepts that will improve fuel performance by minimizing failures involving fuel-cladding interaction (FCI). Current activities include assistance in coated cladding production for the BRPR Series S-2 and S-3 irradiations, friction and wear testing, coating function studies, and the development of an FCI simulator.

4.1 GRAPHITE COATING STUDIES

The previous quarterly report^(1.10) described a modified method for controlling the thickness of Dag 4 graphite coatings applied to the inside surface of zircaloy cladding. This approach, based on short draining and drying times, proved ineffective when applied to cladding for the BRPR Series S-2 and S-3 fuel rods. In order to meet the fabrication schedule for Series S-2 and S-3 rods, the coating application method was modified based on what had been learned previously. The following measures were used:

1. The dag was continuously agitated to eliminate any tendency toward either settling or gelation.
2. The mixing container was modified to prevent air exposure while withdrawing dag.
3. The draining and drying times were carefully standardized at a minimum of 10 and 5 minutes, respectively.
4. At the beginning of each day's work, a trial group of tubes was coated and measured before coating more tubes to either give assurance that the day's production would meet thickness specifications or signal a need for a change in solids content.
5. Tubes were coated in sequential order to identify trends in coating thickness with time, and solids content of the dag was monitored in order to relate this parameter to coating thickness.

These measures proved effective during coating of the large number of tubes processed. The acceptance rate was >90% and essentially all of the rejects were produced on a single day when the test group indicated marginal results and an adjustment should have been made. The average coating thickness per tube when averaged over the 200 acceptable tubes coated, was 6.0 μm with a standard deviation of 0.6 μm . This compares favorably with the specified thickness of $6.4 \pm 1.3 \mu\text{m}$ for the tube average. In the case of the segmented fuel rods for BRPR, the coating thickness specifications were met even more closely. The group of twelve tubes had an average coating thickness of 6.3 μm and a standard deviation of 0.14 μm . Based on all thickness measurements (ten per tube), the standard deviation becomes 0.6 μm . This was accomplished by adhering to the methods described above, with some selecting and re-coating of tubes to meet a more stringent specification of $6.4 \pm 0.5 \mu\text{m}$ on the tube average.

These coating results probably represent the best that is reasonably attainable by the flood coating technique as well as the limit of the present capability to measure coating thickness. Further, it is considered that the base technology for applying an acceptable graphite coating to the inside surface of fuel rod cladding tubes has been demonstrated by a technique that is adaptable to production quantities.

4.2 FRICION AND WEAR TESTING OF GRAPHITE COATINGS

Initially, Dag 154 was evaluated and used as the coating type for the inside cladding surface in the first series of irradiations being conducted in the Halden Boiling Water Reactor (HBWR). Subsequently, after further evaluation of potential graphite coating types, Dag 4 was substituted for Dag 154 and is the preferred type because of its superior adherence characteristics and abrasion resistance during pellet loading.

Because pellet-to-cladding friction may be involved in the effectiveness of graphite coatings to improve FCI behavior, a series of tests were performed to evaluate the friction and wear characteristics of the different types of graphite coatings of interest to this program. A pin-on-disk type of

apparatus, utilizing a round-tipped (1.3 cm radius) UO_2 pellet riding on coated or noncoated zircaloy disks, was used for these tests. With this apparatus, a variable load is applied and measurements are made over relatively long sliding distances at relatively high speeds. It was recognized that the test conditions which would most closely simulate those that exist between UO_2 pellets and zircaloy cladding in an actual fuel rod are high load and small differential movement at very low speeds. However, it was expected that the testing method used and the different environmental conditions employed during the tests would 1) provide an improved understanding of the characteristics and function of the graphite coatings, 2) provide a relative measure of the effectiveness of the different graphite coatings to reduce pellet-to-cladding friction and wear, and 3) provide a comparison between the behavior of coated and noncoated cladding.

The friction and wear test results obtained with this testing method, show that although both types of graphite coatings tested exhibit reduced friction and wear when compared to noncoated zircaloy surfaces, it is doubtful that either graphite would function effectively as a lubricant over significant sliding distances in the environment that exists within an operating fuel rod. These indications suggest that the comparative results obtained by this testing method may not be relatable to the demonstrated effectiveness of graphite coatings to improve FCI behavior during irradiation.^(4.1) In order to obtain data that may provide a better indication of the effectiveness of graphite coatings to improve FCI behavior, measurements of static friction under high loads will be made.

Most of the tests with the pin-on-disk were conducted through an identical regimen or sequence of applied load, disk rotation, and time, as shown in Table 1. One-half rpm (3.1 cm/min) was the slowest reproducible speed available with the test apparatus; the faster speed, 15 rpm (94 cm/min), was used in order to obtain sufficient sliding distance in a short time to make comparisons of rider wear. The loads used with the test apparatus were higher than what is normally used for this type of friction measuring device but were lower than the actual contact forces that exist between the pellet fuel and cladding in an operating fuel rod.

TABLE 4.1. Standard Friction Test Loading and Speed Sequence

<u>Load, Kg.</u>	<u>Speed of Disk, RPM (cm/min)</u>		<u>Time, Min.</u>	<u>Revolutions</u>
3.3	1/2	(3.14)	5	2.5
6.8	1/2	(3.14)	5	2.5
10.0	1/2	(3.14)	5	2.5
13.2	1/2	(3.14)	5	2.5
10.0	1/2	(3.14)	5	2.5
6.8	1/2	(3.14)	5	2.5
3.3	1/2	(3.14)	5	2.5
3.3	15	(94.3)	5	75
6.8	15	(94.3)	5	75
10.0	15	(94.3)	5	75
13.2	15	(94.3)	5	75
10.0	15	(94.3)	5	75
6.8	15	(94.3)	5	75
3.3	15	(94.3)	<u>5</u>	<u>75</u>
Totals	(3410 cm total sliding distance)		70 min	543 rev

During the testing, a coating was assumed to have failed when the friction coefficient reached a value of 0.2. This arbitrary standard was verified as being reasonable by inspecting the disk wear track after testing. With no coating failure during the test, the wear track was narrow with only smooth shiny graphite visible. At failure, the wear track clearly showed bare zircaloy. Further running after failure produced a wide wear track of bare zircaloy metal with UO_2 (identified by SEM) embedded in the surface, and the rider showed a correspondingly large wear scar. Coating performance was also evaluated by calculating the amount of UO_2 worn from the rider during the test, as indicated by the wear scar diameter.

The tests run under the standard sequence are summarized in Table 4.2. Note that not all of the tests were performed in what is considered to be the most relevant environment, i.e., 300°C helium. Some of the tests were performed in vacuum and air atmospheres to evaluate the effect of these conditions on friction and wear behavior in order to provide a better understanding of how the coatings function. Dag 154 and Dag 4 both lubricate extremely well when tested in ambient air, as shown by Tests 2 and 3. In particular, the difference between the rider wear of Test 1 with no coating ($49 \times 10^{-6} \text{ cm}^3/\text{m}$ of sliding) and Test 2 and 3 with Dag 154 and Dag 4, respectively (both showed $\sim 0.02 \times 10^{-6} \text{ cm}^3/\text{m}$ of sliding) is very large. However, the graphite coatings performed poorly as lubricants when tested in vacuum and helium. A well-known characteristic of graphite lubricants is that they require the presence of a condensable vapor (most commonly water) to be effective. When tested in vacuum and helium, rider wear was actually larger with both coatings than with no coating indicating that graphite can act as an abrasive under dry conditions.

The highest friction coefficient observed during this series, 0.42, occurred at the conditions of primary interest, i.e. 300°C in helium, with both coated and noncoated disks. However, both Dag 154 and Dag 4 lasted longer in 300°C helium than under any other conditions except room temperature air. The coatings in Tests 19, 20 and 22 failed at 17.5 revolutions, the point in the standard sequence where speed is significantly increased.

Three disks had specially treated coatings. For Test 23, a Dag 4 coating was preconditioned by running in ambient air with 3.3 to 13.2 kg load for 56 revolutions before testing through the standard sequence in 300°C helium. The preconditioning extended the coating life by a factor of about 5 (compare with Tests 21 and 22); rider wear was about the same. Burnishing the coatings, which oriented the graphite basal planes parallel to the disk surface but also removed most of the graphite, decreased the lifetime of Dag 154 (Test 24) but did not affect the performance of Dag 4 (Test 25).

TABLE 4.2. Friction and Wear Tests Run Through Standard Sequence of 543 Revolutions

Test No.	Zircaloy Disk Coating	Test Conditions	Life of Coating In Revolutions	Friction Coefficient, μ	Wear of UO ₂ Rider in cm ³ Worn per m of Sliding Distance (543 Rev. = 34.1 m)
1	None	250°C, air	0	0.16-0.31	49.0x10 ⁻⁶
2	Dag 154(1)	250°C, air	>543	0.04-0.1	0.02x10 ⁻⁶
3	Dag 4	250°C, air	>543	0.06-0.1	0.02x10 ⁻⁶
4	None	250°C, vacuum(2)	0	0.21-0.3	2.9x10 ⁻⁶
5	Dag 154	250°C, vacuum	8.5	0.05-0.32	3.9x10 ⁻⁶
6	Dag 4	250°C, vacuum	12.5	0.04-0.31	13.5x10 ⁻⁶
7	None	300°C, vacuum	0	0.16-0.27	2.7x10 ⁻⁶
8	Dag 154	300°C, vacuum	8.9	0.06-0.33	3.6x10 ⁻⁶
9	Dag 4	300°C, vacuum	6	0.04-0.22	5.3x10 ⁻⁶
10	None	250°C, helium(3)	0	0.26-0.37	9.1x10 ⁻⁶
11	Dag 154	250°C, helium	0	0.21-0.42	16.0x10 ⁻⁶
12	Dag 154	250°C, helium	3.5	0.05-0.34	14.0x10 ⁻⁶
13	Dag 4	250°C, helium	2	0.08-0.31	13.0x10 ⁻⁶
14	Dag 4	250°C, helium	3.5	0.04-0.36	18.0x10 ⁻⁶
15	None	300°C, helium	0	0.19-0.34	2.1x10 ⁻⁶
16	None	300°C, helium	0	0.21-0.32	4.0x10 ⁻⁶
17	None	300°C, helium	0	0.21-0.42	7.0x10 ⁻⁶
18	Dag 154	300°C, helium	3.8	0.05-0.42	9.5x10 ⁻⁶
19	Dag 154	300°C, helium	17.5	0.03-0.24	9.9x10 ⁻⁶
20	Dag 154	300°C, helium	17.5	0.01-0.31	7.5x10 ⁻⁶
21	Dag 4	300°C, helium	8.5	0.02-0.37	7.0x10 ⁻⁶
22	Dag 4	300°C, helium	17.5	0.03-0.42	7.0x10 ⁻⁶
23	Preconditioned Dag 4	300°C, helium	93 (does not include preconditioning)	0.01-0.24	6.9x10 ⁻⁶ (37.1 m of sliding including preconditioning)
24	Burnished Dag 154	250°C, air	168	0.05-0.42	39.0x10 ⁻⁶
25	Burnished Dag 4	250°C, air	>543	0.03-0.07	0.03x10 ⁻⁶

- (1) Coatings were $\sim 6 \mu\text{m}$ thick and outgassed for 24 hr at 425°C prior to testing. Coatings were applied to a zircaloy surface polished through 600 grit.
- (2) Pressure was 0.01 Pa or lower.
- (3) Helium was backfilled through a 825°C zirconium getter into an evacuated chamber.

There was no apparent dependence of friction and wear behavior on load during these tests, but a "run-in" effect results from the standard sequence because of the initial low loads and speeds. Table 4.3 summarizes the results of tests conducted through the standard time and speed sequence with a single heavy load of 13.2 Kg. In room temperature air, all of the UO_2 riders showed greater wear under the higher load, as expected, but no other significant differences except for the early failure of Dag 154. These results suggest that the behavior of Dag 154 in air is more dependent on the loading sequence than Dag 4. In 300°C helium, Dag 154 showed somewhat longer coating life and slightly less rider wear than Dag 4. Rider wear against coated disks was again slightly higher than against noncoated disks.

TABLE 4.3. Friction and Wear Tests Run Through Standard Sequence, but with Constant Load of 13.2 Kg

Test No.	Zircaloy Disk Coating	Test Conditions	Life of Coating In Revolutions	Friction Coefficient, μ	Wear of UO_2 Rider in cm^3 Worn per m of Sliding Distance (543 Rev. = 34.1 m)
26	None	25°C, air	0	0.27-0.35	64.0×10^{-6}
27	Dag 154	25°C, air	4	0.06-0.35	72.0×10^{-6}
28	Dag 154	25°C, air	17.5	0.08-0.34	98.0×10^{-6}
29	Dag 4	25°C, air	>543	0.01-0.02	0.05×10^{-6}
30	None	300°C, helium	0	0.14-0.27	7.0×10^{-6}
31	Dag 154	300°C, helium	17.5	0.04-0.22	9.2×10^{-6}
32	Dag 4	300°C, helium	4.5	0.09-0.26	12.0×10^{-6}
33	Dag 4	300°C, helium	2.5	0.02-0.28	12.0×10^{-6}

Endurance tests were run in air under an even heavier load of 26.3 kg applied by rapidly increasing loads from 3.3 to 26.3 kg in the first 15 revolutions. These tests show (Figure 4.1) Dag 4 to be superior to Dag 154 and that both coatings are superior to noncoated zircaloy. Overall the friction and wear behavior of Dag 4 is superior to Dag 154 when tested in ambient air, i.e., it maintains a lower friction coefficient over longer sliding distances.

FIGURE 4.1. Endurance Test Results for Graphite-Coated and Noncoated Zircaloy Run in Air Opposite Heavily Loaded UO_2

Dry atmospheres significantly decrease coating endurance and somewhat variable behavior results. Under the extremely dry, high pellet-to-cladding load conditions that exist within a fuel rod during irradiation, it is doubtful that either graphite would function effectively as a lubricant over significant sliding distances. In the environment of primary interest, i.e. 300°C helium, both coatings were superior to no coating and performed about the same, showing friction coefficients of 0.1 over a sliding distance of a few centimeters. Thus, the fuel-to-cladding lubrication afforded by the graphite coating may contribute to the improved FCI behavior of graphite-coated cladding.

4.3 COATING FUNCTION STUDIES

The mechanism by which graphite coatings applied to the inside surface of zircaloy cladding function to improve FCI-behavior is not well understood. Graphite coatings can be expected to reduce pellet-to-cladding friction which in turn, should reduce stress concentrations in the cladding. Attempts(4.2) have been made to measure the effect of reducing pellet-to-cladding friction and it was concluded that while the friction reduction is certainly measurable it may not be the dominant mechanism. The main function of the graphite coating may be to act as a chemical agent which acts to protect the cladding from the active SCC species such as fission product iodine. The coating function studies, as outlined below, have been undertaken in an effort to increase the understanding of how graphite coatings improve FCI behavior.

Determination of the function of the graphite coating as it affects stress corrosion cracking (SCC) of zircaloy cladding requires well defined laboratory studies correlated with the FCI-simulator tests. The information derived from these tests will be used to supplement both the examination and interpretation of the irradiation tests. As a basis for the design of the laboratory tests, they should do the following:

- Be reproducible so that the effects of well-controlled environmental conditions such as iodine concentration and chemical form can be evaluated.
- Apply the stresses and strains in a tangential direction thus simulating the hoop stresses in the cladding.
- Allow examination of the reacting surfaces before and after the tests, thereby leading to a better understanding of the role of surface mechanical defects and chemical inhomogeneities in the cladding and/or coating.
- Develop data that is relatable to both the FCI Simulator results and the irradiation tests.

All four of these criteria can be met using slotted zircaloy tubes as test samples and applying radial stress to the samples by pulling on "Dee's" placed inside them.

A universal testing machine is being used to apply the mechanical loads and record the resulting responses. The mode of mechanical loading can be constant stress, constant strain rate, or constant strain. The tests are being conducted at representative cladding temperatures and the test apparatus has the capability of controlling the oxygen, hydrogen, and iodine content of the inert gas atmosphere. However, the apparatus does not test with UO_2 in contact with either the cladding or the coating. These tests are best done in the FCI simulator.

Three types of tests are being conducted:

- Time-to-failure as a function of strain rate on noncoated cladding, using I_2 and possibly Cd or Cs as SCC agents and comparing with results obtained without these agents present.
- Comparison tests using coated cladding to determine the effects of coating.
- Tests on specimens with highly characterized surfaces to determine the role of surface condition on SCC.

The data on strain-to-failure as a function of strain ratio in the controlled chemical environments will be used to quantify any differences that may exist between coated and noncoated cladding. The more detailed analyses of quantities such as iodine concentration in the coating and cladding, as well as the chemical species present, will be used to construct a model which fits the observations.

4.4 FUEL-CLADDING INTERACTION SIMULATOR STUDIES

The FCI simulator, which is a load application system designed to simulate the loading conditions in the cladding at the pellet-pellet interface, with cracked fuel, has been assembled and a series of preliminary tests are currently being performed to evaluate the mechanical characteristics of this device. The mechanical loading system, which is based on a split mandrel concept, uses a deformable plug in combination with split annular metal segments and an annulus of UO_2 at the inner cladding boundary to deform the cladding. (Alumina is currently being used as a standin for

UO₂). Comparative tests are being performed to evaluate the effects of the selected fuel design options, coatings, and chemical environment on FCI behavior.

The segments which comprise the split mandrel, are 9.5 mm (0.37 in) long which gives a total test section length of 19.1 mm (0.75 in) for simulating the pellet-pellet interface. The deformable plug is designed to expand at the center of the two sets of segments and thus simulates the hourglassing which occurs at the ends of the fuel pellets.

Several types of deformable plugs have been used. Copper tubes with a silicone rubber center have been tried, as have silicone filled aluminum tubes. The load vs. deflection relationship for the filled tubes was very good as long as the tubes did not rupture, which allowed the rubber to be extruded. The range of application for the filled tubes is limited by this characteristic to total strains of less than 4%. Solid aluminum plugs have been successfully used to strains of 10%. The current work indicates that 6061-T6 aluminum alloy will be used as the deformable plug because it extrudes less between segments than pure aluminum.

The FCI simulator uses hydraulic actuators to deform the loading plug and to internally strain the cladding. The hydraulic power supply for the simulator is capable of providing oil to the actuators at a preselected flow rate. Thus, the plug deforms at a constant volume and causes the mandrel segments to expand in a nonlinear manner. The simulator can be operated over a rather broad range of hydraulic oil flow rates; however, any base rate that is selected will change as the plug becomes deformed. Scoping tests are planned to cover the range of strain rates which occur in LWR fuel rod cladding during normal power changes.

The response of the FCI simulator during testing is shown in Figure 4.2. The pressure shown is the hydraulic pressure of the actuators that are used to compress the deformable plug as oil is supplied at a constant rate. The strain shown is the maximum change in diameter of the cladding specimen measured at the pellet-pellet interface.

Rupture of the cladding is shown (Figure 4.2) as a decrease in the system hydraulic pressure. Rupture is also detected by a reduction in the flow of the helium carrier gas which transfers the iodine to the cladding surface during testing. When fracture of the cladding occurs, the gas escapes at the rupture and is no longer carried through the specimen. A simple bubbler at the gas exit provides an indication of cladding failure.

Tests are being performed to separate the load-deformation characteristics of the deformable plug from those of the cladding in order to relate the measured hydraulic pressure to cladding stress. As soon as these calibration tests are completed, the comparative FCI evaluation tests will be conducted in accordance with the test plan.

FIGURE 4.2. FCI Simulator Hydraulic System Pressure and Maximum Diametral Cladding Strain as a Function of Time for a Constant Oil Flow Rate (temperature 300°C, 6061 solid aluminum deformable plug)

5.0 TASK 2B - IN-REACTOR EXPERIMENTS

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The objective of the in-reactor experiments task is to conduct irradiation tests to study the detailed behavioral characteristics and performance limits of those fuel concepts which have potential for improved operating behavior in LWRs. Comparative parametric fuel rod experiments are being performed with emphasis on the annular-coated-pressurized design and, at a more modest level, the sphere-pac design as a back-up. The performance of both designs is compared with a reference fuel design. The performance characteristics of the improved concepts are being evaluated in screening tests which involve three sequential series of power cycling/ramping irradiations of instrumented rods in the Halden Boiling Water Reactor (HBWR) and the irradiation of instrumented sphere-pac rods in HBWR to measure central fuel temperature and cladding elongation behavior of this fuel type. The results of these comparative irradiations will be utilized to select the most promising fuel concept(s) and test conditions for evaluation in the primary power-ramping experiments. Rod segments for the primary tests are being irradiated in the Big Rock Point Reactor (BRPR) as part of the Task 2C in-reactor demonstrations.

Details of the general testing criteria, the selection of the irradiation facilities, and the detailed test descriptions of irradiations currently in progress were presented in previous progress reports.^(1.8, 1.9) Regarding the initial irradiations which are in progress, the emphasis for HBWR Series H-1 was placed upon evaluating the individual effects of design differences between reference, annular, annular-coated, and vipac designs.

5.1 SUMMARY DESCRIPTION OF HBWR TESTS

The screening tests are being irradiated in two test rigs in the HBWR. The IFA-518 test rig contains two six-rod clusters and the power in the two

clusters is varied by alternately shifting a silver absorber shield over the two clusters. The IFA-517 test rig contains a single cluster of four rods. The peak operating linear heat generation rate in both assemblies is 44.5 kW/m.

Rod instrumentation in both assemblies includes fuel centerline thermocouples, elongation sensors, and internal pressure sensors. IFA-518 will be sequentially loaded with rods comprising HBWR Series H-1, H-2, and H-3, respectively. Peak burnups will be in the range of 1200 to 1400 GJ/kg (14 to 16 MWd/kg).

The HBWR Series H-4 rods will be irradiated in IFA-517 to peak burnups of about 430 GJ/kg (5 MWd/kg). The IFA-517 rig will also be used later for the power-ramping tests.

5.2 FABRICATION - HBWR SERIES H-1, H-2, H-3, and H-4 RODS

5.2.1 HBWR Series H-1 Characterization Report

The HBWR Series H-1 characterization report which describes the Series H-1 test rods undergoing irradiation in HBWR was completed. As the report is considered to be a working document, a concerted effort was made to include the information necessary for future fuel performance analysis. The major topics included in this report are as follows:

- Experiment Description
- Process Development for HBWR Series H-1 Rods
- Fabrication and Characterization Procedures and Methods
- Physical, Chemical and Ceramographic Characterization of the HBWR Series H-1 Fuel and Fuel Rods.

5.2.2 Rod Fabrication HBWR Series H-2 and H-3

Fabrication of the remaining 10 fuel rods for HBWR Series H-2 and H-3 was completed. The fuel rods were then x-rayed, etched, neutron radiographed and shipped to Norway in April 1979. Profilometry was performed at the Kjeller Laboratory in Norway.

A description of the rods in this final shipment are:

- 2 - Reference rods with elongation sensors
- 2 - Reference rods with pressure transducers
- 4 - Annular-Coated-Pressurized rods with elongation sensors
- 2 - Annular-Coated-Pressurized rods with pressure transducers
- 1 - Sphere-pac rod with elongation sensor
- 1 - Sphere-pac rod with pressure transducer.

Neutron radiographs made in the Triga reactor at the Hanford Engineering Development Laboratory showed no pellet chipping or other anomalies in the fuel columns. Also, the smear densities in the sphere-pac fuel columns are extremely small (no longer than approximately 1.5 mm).

The fuel rods were stored briefly because the lower end caps made at Halden required finishing of the weld preparation area. The weld design was changed slightly from Series H-1 to reduce the possibility of voids. This improvement was verified by the fact that no bottom end cap welds were rejected.

While awaiting the delivery of the lower end caps, the loaded annular and reference fuel rods were fitted with temporary end caps and stored in a helium atmosphere glove box. Control samples consisting of pellets and coated (Dag 4) cladding were also placed in the storage glove box. Just prior to welding the permanent lower end caps to the rods, the samples were analyzed for water and hydrogen content. These results showed that the pellets contained <2ppm H₂O. The coated cladding sample was found to contain 5190 ppm H₂O* (0.03 ppm H-fuel weight basis) as compared to 4180 ppm H₂O (0.24 ppm H-fuel weight basis) observed just prior to storage. Thus, the fuel rods were judged to be acceptable for closure. In addition to the process control samples, the helium atmosphere of the glove box storage area was sampled just before and just after storage with the following results:

- Just before storage - 99.7% Helium
- Just after rods were removed - 93.6% Helium and
from storage 89.2% Helium.

* Average of five analyses.

Each fuel rod is identified by a serial number. Some important physical dimensions of each fuel rod are described in Table 5.1.

Fabrication of the HBWR Series H-2, H-3 and H-4 test rods proceeded smoothly as planned, based upon the experience gained in fabricating HBWR Series H-1. There were only eight deviating reports issued during this fabrication effort. Table 5.2 presents a summary of these reports.

TABLE 5.1. Halden - HBWR Series H-2, H-3, & H-4 Fuel Rods

ROD SERIAL NUMBER	DESCRIPTION	Overall Length		Rod Diameter O.D.					
		mm	(in)	0° top	0° middle	0° bottom	90° top	90° middle	90° bottom
H2UCR023	Reference with elongation sensor	715.289	(28.161)	12.327 (0.4853)	12.327 (0.4853)	12.329 (0.4854)	12.334 (0.4856)	12.324 (0.4852)	12.334 (0.4856)
H2UCR024	Reference with pressure transducer	715.239	(28.159)	12.332 (0.4855)	12.324 (0.4852)	12.329 (0.4854)	12.332 (0.4855)	12.334 (0.4856)	12.332 (0.4855)
H3UCR025	Reference with elongation sensor	715.315	(28.162)	12.327 (0.4853)	12.327 (0.4853)	12.324 (0.4852)	12.327 (0.4853)	12.324 (0.4852)	12.329 (0.4854)
H3UCR026	Reference with pressure transducer	715.188	(28.157)	12.327 (0.4853)	12.329 (0.4854)	12.339 (0.4858)	12.319 (0.4850)	12.329 (0.4854)	12.324 (0.4852)
H2UCG027	Annular-Coated-Pressurized with elongation sensor	715.442	(28.167)	12.332 (0.4855)	12.332 (0.4855)	12.329 (0.4854)	12.327 (0.4853)	12.327 (0.4853)	12.319 (0.4850)
H2UCG028	Annular-Coated-Pressurized with pressure transducer	715.620	(28.174)	12.329 (0.4854)	12.327 (0.4853)	12.337 (0.4857)	12.329 (0.4854)	12.334 (0.4856)	12.327 (0.4853)
H2UCG029	Annular-Coated-Pressurized with elongation sensor	715.061	(28.152)	12.327 (0.4853)	12.324 (0.4852)	12.327 (0.4853)	12.332 (0.4855)	12.327 (0.4853)	12.329 (0.4854)
H3UCG030	Annular-Coated-Pressurized with elongation sensor	715.162	(28.156)	12.329 (0.4854)	12.327 (0.4853)	12.324 (0.4852)	12.324 (0.4852)	12.327 (0.4853)	12.329 (0.4854)
H3UCG031	Annular-Coated-Pressurized with pressure transducer	715.061	(28.152)	12.332 (0.4855)	12.324 (0.4852)	12.324 (0.4852)	12.339 (0.4858)	12.327 (0.4853)	12.329 (0.4854)

TABLE 5.1. Halden - HBWR Series H-2, H-3, & H-4 Fuel Rods (Continued)

ROD SERIAL NUMBER	DESCRIPTION	Overall Length		Rod Diameter O.D.					
		mm	(in)	0° top	0° middle	0° bottom	90° top	90° middle	90° bottom
H3UCG032	Annular-Coated-Pressurized with elongation sensor	715.086	(28.153)	12.334 (0.4856)	12.334 (0.4856)	12.332 (0.4855)	12.327 (0.4853)	12.332 (0.4855)	12.332 (0.4855)
H2UCS033	Sphere-pac-Pressurized with elongation sensor	715.721	(28.178)	12.334 (0.4856)	12.344 (0.4860)	12.342 (0.4859)	12.339 (0.4858)	12.342 (0.4859)	12.342 (0.4859)
H3UCS034	Sphere-pac-Pressurized with elongation sensor	715.315	(28.162)	12.339 (0.4858)	12.332 (0.4855)	12.337 (0.4857)	12.332 (0.4855)	12.334 (0.4856)	12.332 (0.4855)
H2SPS040	Sphere-pac-Pressurized with elongation sensor	715.620	(28.174)	12.334 (0.4856)	12.337 (0.4857)	12.342 (0.4859)	12.337 (0.4857)	12.334 (0.4856)	12.334 (0.4856)
H2XXS041	Sphere-pac-Pressurized with thermocouple and elongation sensor	715.442	(28.167)	12.342 (0.4859)	12.332 (0.4855)	12.342 (0.4859)	12.339 (0.4858)	12.332 (0.4855)	12.332 (0.4855)
H2XXS042	Sphere-pac-Pressurized with thermocouple and pressure transducer	715.315	(28.162)	12.342 (0.4859)	12.339 (0.4858)	12.344 (0.4860)	12.344 (0.4860)	12.339 (0.4858)	12.344 (0.4860)

TABLE 5.2. Halden HBWR Series H-2, H-3, & H-4 Fuel Rods

Deviating Material Reports

Deviation	Description	Resolution
Hardware - Porous Frits	Dimensions	Slight deviations will not affect function of part.
Fuel - Instrumented Pellets	Center hole dimensions	Slight deviations will not affect function of part.
Fuel - Annular and Reference Pellets	N ₂ , O ₂ gas analysis	Equipment not available for analysis. Sufficient pellets have been placed in protective archive containers for future analysis.
Fuel - Sphere-pac	Off-gas measurements	Off-gas measurements were affected by atmospheric contamination during analysis.
Fuel - Sphere-pac	H ₂ analysis	H ₂ measurements were affected by atmospheric contamination. Moisture analysis for 5 samples (4, 2, 3, 3, 4 ppm H ₂ O. $\bar{X}$ = 3.2 ppm, σ = 0.836 ppm, and upper 95:95 = 6.7 ppm H ₂ O. Sphere-pac crystal is of high density and has little porosity.
Fuel - Sphere-pac	Immersion density	Coarse = 98.16% TD Medium = 98.17% TD Analytical procedure not capable of analyzing fines ($\leq 45\mu$).
Rod - Sphere-pac	Test equipment not available to check rod internal pressure.	Welding equipment was qualified using calibrated pressure transducer.
Rods - Sphere-pac	Small gaps (~ 1 mm) between lower insulator discs and end caps.	Small amount of fuel between lower insulator is estimated to be at low packing density ($\sim 65\%$ TD) and will not provide much power or very high temperature.

5.3 IRRADIATION STATUS - HBWR TESTS

5.3.1 IFA-518 Irradiation

Irradiation of the 12 fuel rods in HBWR Series H-1 continued satisfactorily with the beginning and completion of the third cycle. The shield was raised for the start of the third cycle on April 5, 1979. The cycle ended May 22, 1979 with an estimated peak burnup of 497 GJ/Kg (5.8 MWd/Kg) in the lower cluster and 415 GJ/Kg (4.8 MWd/Kg) in the upper cluster. The HBWR operated at ~90% full power during the third cycle's steady-state operation with an average linear heat rate of 33 kW/m and a peak linear heat rate of 41 kW/m in the reference rod. The next cycle will begin in July with the shield remaining in the upper position.

Movement of the silver shield prior to reactor startup on April 5, 1979 uncovered the lower cluster which was "reconditioned" during the previous cycle of operation. During the initial power ascension after the shield shift, fuel-cladding mechanical interaction (FCI) was indicated by the elongation sensors (EC) in the reference, annular and vipac test rods. FCI was not observed in the annular-coated rod. A preliminary estimate of the degree of FCI during the third cycle startup based on rod elongation data indicates that the interaction was 50% greater in the vipac rod than in the reference or annular rods.

An inspection of internal rod pressure readings in IFA-518 for reactor operation between November 28, 1979 and January 26, 1979 indicated all pressure instrumentation was operating satisfactorily. Two other observations are noteworthy:

- After the September 26, 1978 shield shift when the shield was lowered to cover the bottom cluster, the pressure transducers in the upper cluster rods indicated slowly increasing readings during steady state operation. There are two upper cluster rods with pressure transducers: one reference rod and one vipac rod. This behavior is attributed to increased fission gas release at the higher LHGR.
- The lower cluster rods exhibited constant pressure readings during this same period.

The above observations were obtained from rod pressures calculated using rod elongation and pressure data. The method for calculating rod pressures is being reviewed to either verify the approach or develop a revised method.

Several computer codes were written to handle Test Fuel Data Report (TFDR) and CALIB data - TFDR data is collected every 15 minutes and CALIB data is collected every 3-5 minutes. Program POW518 calculates average rod powers, upper and lower cluster powers, and total assembly power using the revised power equations. CHGTM reassigns time to CALIB data tapes so that all data is referenced to one starting time. CPYCAL compresses the CALIB disk file eliminating unnecessary data.

TFDR printouts were received for IFA-518 operation from January 26, 1979 to May 22, 1979. Magnetic tapes have not been received for this time period. The TFDR printouts showed erroneous readings in both the upper and lower cluster elongation sensors (EC). The sensors in the lower cluster appear to have operated satisfactorily at least during the April 5th power ascension, i.e., the first ramp after the shield was raised at the beginning of the third cycle. IFA believes that the EC errors are due to simultaneous activation of the sensing coils in the ECs and that this concurrent operation causes the magnetic fields to alter the individual elongation readings. The ECs were hard wired for the collection of FAST SCAN data during the last cycle of operation. The elongation sensors are no longer hard wired and corrections are being developed by IFA for the TFDR data affected by this problem. FAST SCAN data for both IFA-518 and IFA-517 were received from IFA and contain the same type of data as on TFDRs but data is sampled at rates up to 7 times per second. Programs for handling this data are being developed.

5.3.2 IFA-517 Irradiation

The HBWR Series H-4, consisting of one reference and three sphere-pac fuel rods, was loaded into IFA-517 and started irradiation on April 5, 1979. TFDR printouts were received for this initial cycle in IFA-517 and preliminary tape data has been placed on disk storage using the code RW517. The peak burnup on May 22, 1979 is estimated as 139 GJ/Kg (1.6 MWd/Kg) assuming a peak- to-average ratio of 1.034 which was based on the HBWR axial power profile.

The instrumentation in IFA-517 is operating satisfactorily except for the problem associated with the ECs. Preliminary temperature data is discussed in Section 7.2.

6.0 TASK 2C - IN-REACTOR DEMONSTRATIONS

R. K. Welty and P. G. Sarafian (Exxon Nuclear Company)

6.1 DESIGN AND ANALYSIS

The schedule for fabrication, licensing, and shipment of the three BRPR Series S-2 fuel assemblies was accelerated to meet the projected September-October startup of the Big Rock Point Reactor (BRPR). Plans are to substitute two BRPR Series S-2 assemblies (including sphere-pac rods) for two of the four BRPR Series S-1 assemblies (which contain some vipac rods). Results of the neutronics analysis show that the Series S-1 and S-2 assemblies are virtually identical with respect to power peaking. The two S-1 assemblies will be inserted later and they will receive the planned irradiation.

For purposes of economy, all of the rods required for BRPR Series S-2 and S-3 are being fabricated at the same time as the BRPR reload batch H-1 (cycle 16) fabrication. Details and bundle maps for Series S-2 assemblies were shown in the January-March 1979 Quarterly Progress Report. (1.10)

The parts list for BRPR Series S-2 and S-3 fuel rods, XN-NF-PL-15, was approved with one minor drawing change requested. The drawing for the annular-coated-pressurized segmented fuel rod assembly was revised by an interim document revision to add a vented segment connector between the middle and lower segments. A vented segment connector shown in Figure 6.1 is used where there is a small plenum area in an unpressurized rodlet. This situation exists in all lower rodlets and in the unpressurized middle reference pellet rodlet. The vent relieves fill gas pressurized by the heat of welding that would otherwise intrude the weld area and produce an unsatisfactory weld containing gas bubbles.

The Irradiation Test Authorization (XN-NF-ITA-1) for BRPR Series S-2 and S-3 demonstration fuel rods was approved, issued, and sent to CPC.

6.2 PROCESS DEVELOPMENT - BRPR SERIES S-2 AND S-3

6.2.1 BRPR Sphere-Pac Process Development

In arriving at an acceptable design for the sphere-pac fuel retainer disc shown in Figure 6.2, there was some speculation that thermal expansion and swelling of the fuel column could be impeded by bonding between the upper end fuel retainer and the cladding. The Halden rod design had a fuel retainer disc only at the top of the rod that was designed for an interference fit with about 2.3-4.5 Kg (5-10 lb) insertion force. A fuel retainer disc is to be inserted in BRPR Series S-2 and S-3 rods between the fuel column and the lower insulator disc as well. This is designed to prevent microspheres from intruding between the insulator disc and lower end cap. The retainer disc must fit snugly enough to prevent microsphere intrusion while not causing undue strain concentration in the cladding during creepdown. The design with an interference fit has been shown to prevent microsphere intrusion under the lower insulator disc and also to prevent microspheres from falling into the upper plenum during horizontal rod handling.

The original design of the sphere-pac fuel retainer disc was modified to open the center hole to 5.56 mm (0.219 inch). This should increase disc flexibility and decrease the contact with hot fuel. Development tests have shown that such a disc design prevents fuel from intruding into the upper plenum and significantly reduces the amount of fuel intruding between the lower insulator disc and the end cap. Process tests were also performed to determine the effect of these discs inside fuel rods. In the first test, a disc was sealed with fuel pellets and insulators inside a cladding tube and heated to 500°C for 72 hours. Results showed that normal internal cladding oxidation did not cause any galling or seizing between the retainer and the cladding. After the test, the retainer fell out of the cladding with no applied force. Because the oxide film was very thin in this test, a second test was performed in which the sphere-pac fuel retainer disc was retested by autoclaving two discs in an open tube for 72 hours at 400°C. The initial force to seat the two retainers, as measured by a Chatillon gauge, was

6.12 Kg (13.5 lbs) and 4.99 Kg (11.0 lbs). After autoclaving, the force to move the retainers dropped to 3.63 Kg (8 lbs) and 2.72 Kg (1 lb), respectively. It appears that the autoclave film is acting as a lubricant and actually lowers the friction between the disc and tube instead of locking the disc and inhibiting the fuel stack expansion.

Process development activities continued on decreasing the variability of axial density in the sphere-pac fuel columns. The previous problem of levitation of the lower insulator disc was solved by use of the zircaloy fuel retainer discs discussed above. None of the rods loaded with microspheres using the fuel retainers has shown any fuel under the insulator disc by X-ray analysis.

The previous method of filling a rod with microspheres was to pre-weigh the required amount of fuel and to load the tubes vibrationally until all the fuel was emptied from the feeders. This usually resulted in variable densities (70-80% TD) within the top 2.5 mm of the rod because the amounts of the three microsphere components tailed off at different rates when the very last of the fuel exited from the feeders. The low density regions could be caused by either an excess of fines or a lack of fines.

The solution for this problem is an overfill technique wherein an excess of fuel is added, filling the rod into the plenum area; then the rod is vibrated and finally the excess microspheres are removed by aspiration down to the correct plenum length. Rods loaded in this manner have 86.5-87.0% TD up to the very end of the fuel column. The spheres that are removed from the rod can easily be separated and recycled back to the feed stock.

This aspiration technique can also be used to rework a rod that has some discontinuity in the fuel columns (e.g., low density). The microspheres can be removed by aspiration down to a point just below the variable density region and the rod can then be refilled as described above. Two development rods were reworked by this method: regions with ~71% TD were removed and the rod was refabricated to meet acceptance specifications.

The 2.52% enriched microspheres for BRPR demonstration rods in Series S-2 and S-3 were received from ORNL. This shipment contained 29.6 Kg of coarse acceptable spheres and 8.5 Kg of coarse tentatively rejected but "usable" spheres. Because a total of 31.5 Kg of coarse microspheres are required to fabricate the BRPR Series S-2 and S-3 fuel rods, the coarse rejected "usable" spheres were analyzed to determine if they were indeed usable. SEM photos were made of the surface and compared to acceptable coarse microspheres from the same shipment. There were an estimated 50% more surface irregularities present on these microspheres, but nothing significant. Metallographic examination was also made on a representative sampling of the microspheres. Grain size was the same as acceptable fuel, and no porosity or gross cracking was present. Total gas measurements were made on vacuum outgassed (no heat) samples. Values were extremely low, 0.001 cc/gm, again indicating a solid crystal. Density measurements (by water immersion) were 99% TD (ORNL data results were 98.75% TD by Toluene immersion). Based on these analyses the coarse reject "usable" spheres were blended with the coarse feed material to be used for loading the Series S-2 and S-3 rods.

6.3 FABRICATION - BRPR SERIES S-2 AND S-3

6.3.1 BRPR Pellet Fabrication

Fabrication of the reference and annular pellets was completed. Very high yields (~98%) were obtained, largely attributable to the chamfered design of the pellets. Production data for the reference and annular pellets are listed in Tables 6.1 and 6.2.

One thousand reference and one thousand annular pellets were characterized (diameters measured) and laid out in fuel stacks on the outgas trays for loading into the middle rodlets of the segmented rods. Each pellet is identified by a scribed letter and number. To help keep track of each fuel stack layout, the pellets are segregated by similar letters for each rod. The last two pellets, in each case, can be adjusted if necessary using "short" pellets prepared in three different lengths, (7.11 mm {0.280"}, 6.6 mm {0.260"}, 6.1 mm {0.240"}) to permit adjusting to fuel column length requirements.

6.3.2 BRPR Graphite Coated Tube Fabrication

Graphite coating of the tubes for BRPR Series S-2 and S-3 is complete. The overall yield was nearly 100% following the initial adjustment of the Dag 4 solids content. The improvements that led to this high yield were continuous agitation of the Dag 4 solution and determination of the relationship between the drain time (after applying the graphite) and the Dag 4 solids content. Present practice is to fix the drain time at 10 minutes minimum and vary the solids content of the Dag 4 slightly between 14 and 16 wt% to meet required thickness.

During this production campaign, 201 full length-tubes and 12 segmented tubes were satisfactorily coated. In processing this number of tubes the Dag 4 solution was adjusted five times for solids content. A description of the coating experience gained during this task follows:

<u>Batch Number</u>	<u>Coating Sequence for Dag 4 Solution</u>
1	Coated tubes 20-59.
2	Coated tubes 60-79. Batch rejected due to thin coatings. Average film thickness from 0.004-0.005 mm (0.00016-0.00019 in.).
3	Coated tubes 80-15. Solids content increased to ~15 wt%.
4	Coated tubes 160-179. Dag diluted with 40 cc deionized water because Dag 4 solution had stood inactive for four days.
5	Recoated tubes 60-79 (from batch 2). Diluted Dag 4 with 20 cc deionized water. Previous experience has shown that when recoating tubes, a thinner solution is required. Coated tubes 180-220. Batch 5, after standing for two days had thickened to the point where additional thickening of the Dag 4 to coat the remaining new cladding was not necessary.

The average film thickness for the full length tubes was nominally 6 μm (0.00024 in.). The film thickness on the rodlets for the segmented fuel was controlled tightly at 6.29 μm (0.000248 in.); σ for the rodlet coating thickness = ± 0.14 mm.

TABLE 6.1. Production Summary - BRPR Series S-2 and S-3 Reference Pellets

<u>Lot Number</u>	<u>Tooling Used</u>	<u>Lubricant and Pore Former, wt%</u>	<u>Press Density gm/cm³</u>	<u>Sintering Temp., °C (actual)</u>	<u>Sintering Time, hr.</u>	<u>Sintered Density, %TD</u>
SR 67-2 Rev. 1 2.52%	11.8 mm (0.465") dished and chamfered tooling design; green density ~5.25 gm/cm ³	0.1 wt% zinc stearate and 0.4 wt% Sterotex	5.25	1740	2.82	93.45 $\sigma=0.27$

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Resinter test (1700°C for 24 hours)

Density before resinter = 93.30% TD $\sigma = 0.31$

Density after resinter = 93.74% TD $\sigma = 0.27$

- 1) Powder was milled at 8000 RPM with an 0.686 mm (0.27") herring bone screen.
- 2) Roll compacted at 6.99 mm (2.75"); granulated through a 10 mesh screen to minimize protrusions.
- 3) Lube and pore former added to prepped powder, tumbled for 12 minutes.

TABLE 6.2. Production Summary - BRPR Series S-2 and S-3 Annular Pellets

<u>Lot Number</u>	<u>Tooling Used</u>	<u>Lubricant and Pore Former, wt%</u>	<u>Press Density gm/cm³</u>	<u>Sintering Temp., °C (actual)</u>	<u>Sintering Time, hr.</u>	<u>Sintered Density, % TD</u>
SR-48-1 Rev. 1 2.69%	11.8 mm (0.465") flat faced, annular and chamfered tooling design; green density ~5.25 gm/cm ³ core rod diameter = 3.7 mm (0.1459")	0.1 wt% zinc stearate and 0.4 wt% Sterotex	5.25	1780	3.4	94.35 $\sigma=0.24$

⌘ Resinter test (1700°C for 24 hours)

Density before resintering = 93.57% TD $\sigma = 0.17$

Density after resintering = 93.89% TD $\sigma = 0.10$

- 1) Powder was milled at 8000 RPM with an 0.686 mm (0.027") herring bone screen.
- 2) Roll compacted at 50.8 mm (2.00"); granulated through a 10 mesh screen. Re-roll compacted at 76.2 mm (3.00") granulated through a 10 mesh screen to minimize protrusions.
- 3) Lube and pore former added to prepped powder; tumbled for 12 minutes.

FIGURE 6.1

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REFERENCES

- 4.1 D. G. Hardy, A. S. Bain, and R. R. Meadowcroft, "Performance of CANDU Development Fuel in the NRU Reactor Loops," Proc. Conf. American Nuclear Society, Water Reactor Fuel Performance, p 198 (1977).
- 4.2 A. J. Bain, J. C. Wood, C. E. Coleman, "Fuel Designs to Eliminate Defects on Power Increase," Proc. Conf. British Nuclear Soc., p. 56.1 (1973).

7.0 TASK 3 - FUEL PERFORMANCE EVALUATIONS

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This task serves as a focal point for the integration and evaluation of data developed from the Fuel Performance Improvement Program (FPIP) and evaluates the results of other FCI-related programs as they may pertain to the FPIP. The task activities are intended to provide the basic resources for establishing technical bases for fuel design. At the present time, most of the activities in this task are directed towards determining the validity of the IFA-518 power equations, estimating the iodine produced in the HBWR Series H-4 reference rod, and monitoring the fuel rod behavior in IFA-518 and IFA-517.

7.1 HBWR SERIES H-1 PERFORMANCE EVALUATION

The Series H-1 fuel rods under irradiation in the Halden Boiling Water Reactor (HBWR) assembly IFA-518 include 12 fuel rods with six of these rods hexagonally arrayed in an upper cluster and the remaining six rods in the lower cluster. Four fuel forms are included in HBWR Series H-1: reference, annular, annular-coated, and vipac. A moveable neutron absorbing shield covers one of the clusters at any one time allowing the power of the shielded cluster to be reduced approximately 40%. The IFA-518 assembly is instrumented with neutron detectors, fuel centerline thermocouples, cladding elongation sensors and internal pressure transducers.

As part of the in-reactor experiments, calculations of rod-average and local linear heat generation rates (LHGR) are made to compare performance of the various fuel designs. Errors in these calculations were noted by both the Institutt for Atomenergi (IFA) and Pacific Northwest Laboratories (PNL) after the first shield shift on September 26, 1978 when the silver shield was lowered from the upper to the bottom cluster. The shield down position caused the LHGRs at the thermocouple (LHRTF) to be lower for a given temperature than when the shield was up. The error in reference fuel was estimated to be

~18%. A slightly smaller error in LHGR was noted for annular-coated fuel and may be compared with reference fuel using Figures 7.1 and 7.2. A concerted effort was made to determine the cause of this error and any other errors in the power equations so that an accurate comparison can be made between the performance in each fuel rod.

The methods used to treat the corrections for calculated upper and lower cluster powers, individual rod- average powers, and the power at the thermocouple tips in HBWR Series H-1 fuel rods are described in the following sections.

FIGURE 7.1. Comparison of the Temperature Changes in a Reference Rod During Several Power Ascensions.

FIGURE 7.2. Comparison of the Temperature Changes in a Annular-Coated Rod During Several Power Ascensions

7.1.1 HBWR Series H-1 Power Equations

The original power equations were supplied by IFA and included the effects of gamma heating and hardening of the neutron flux spectrum by the silver absorber shield. These equations also included terms to correct for flux tilt, depletion of fissile fuel, and the difference in the fissile fuel mass in the upper and lower cluster. The total IFA-518 assembly power was given by the following equation:

$$QA = K_g \cdot CD_L \cdot SP_L \cdot \left(\frac{I_1 + I_4}{2I_1} \right) \cdot \left(\frac{I_1 + I_2 + I_3}{3} \right) + K_g \cdot CD_U \cdot SP_U \cdot MR \cdot \left(\frac{I_5 + I_8}{2I_5} \right) \cdot \left(\frac{I_5 + I_6 + I_7}{3} \right) \quad (1)$$

where,

QA = total assembly fission power, kW

K_g = calibration constant corrected for gamma heating, kW/nanoamps
(1.128 for shield up, 1.120 for shield down)

CD_L, CD_U = depletion factors for the lower and upper clusters,
respectively

SP_L, SP_U = spectral factor to account for spectrum hardening (1.0 for
nonshielded cluster, 1.1 for shielded cluster)

MR = fissile fuel mass ratio between upper and lower cluster (1.02)

I_j = current from the jth neutron detector, nanoamps.

Equation (1) is composed of two components. The first component contains I_1, I_2, I_3 and I_4 and represents the lower cluster power. The second component involves I_5, I_6, I_7 and I_8 and is the upper cluster power.

Once the cluster powers are determined, an appropriate power split between the six rods in a cluster will give the total power for each rod. These rod powers may be divided by the active fuel length to determine the average linear heat generation rate (ALHR) in each rod. The original ALHRs for rods in the lower and upper clusters are given by Equations (2) and (3), respectively.

$$\begin{aligned}
 \text{ALHR}_i = & K_g \cdot \text{CD}_L \cdot \text{SP}_L \cdot \left(\frac{M_i}{M} \right) \cdot \left(\frac{1}{L_i} \right) \cdot \left(\frac{I_1 + I_4}{2I_1} \right) \\
 & \cdot (0.757 I_n + 0.333 I_m - 0.091 I_f)
 \end{aligned} \tag{2}$$

for $i = 1, 6$

$$\begin{aligned}
 \text{ALHR}_i = & K_g \cdot \text{CD}_U \cdot \text{SP}_U \cdot \text{MR} \cdot \left(\frac{M_i}{M} \right) \cdot \left(\frac{1}{L_i} \right) \cdot \left(\frac{I_5 + I_8}{2I_5} \right) \\
 & \cdot (0.757 I_n + 0.333 I_m - 0.091 I_f)
 \end{aligned} \tag{3}$$

for $i = 7, 12$

where,

M_i = fissile mass in the i th rod, g

M = total fissile mass in the appropriate cluster, g

L_i = length of the 11% enriched fuel region in the i th rod, m

I_n = current of the nearest neutron detector in the 3-detector array, nanoamps

I_m = current of the next nearest detector in the 3-detector array, nanoamps

I_f = current of the farthest detector in the 3-detector array, nanoamps.

The terms involving I_n , I_m and I_f in Equations (2) and (3) result from the inclusion of a radial flux tilt factor. The correct detectors to use for I_n , I_m and I_f are different for each rod and are listed in Table 7.1.

The linear heat generation rate of the fuel rod at the thermocouple (LHRTF) is determined by multiplying Equation (2) by the ratio $2I_4/(I_1 + I_4)$. This ratio is an

TABLE 7.1. Neutron Detectors Used to Calculate Rod Powers

<u>Rod Position</u>	<u>Detector Number</u>		
	<u>I_n</u>	<u>I_m</u>	<u>I_f</u>
1	1	2	3
2	2	1	3
3	2	3	1
4	3	2	1
5	3	1	2
6	1	3	2
7	5	6	7
8	6	5	7
9	6	7	5
10	7	6	5
11	7	5	6
12	5	7	6

approximation for the power at the thermocouple tip compared with the average rod power. The LHRTF equations were given by the following:

$$LHRTF_j = ALHR_{j+1} \cdot \left(\frac{CDTF}{CD_L} \right) \cdot \left(\frac{2I_4}{I_1 + I_4} \right) \quad (4)$$

or

$$LHRTF_j = Kg \cdot CDTF \cdot SP_L \cdot \left(\frac{M_{j+1}}{M} \right) \cdot \left(\frac{1}{L_{j+1}} \right) \cdot \left(\frac{I_4}{I_1} \right) \cdot (0.757 I_n + 0.333 I_m - 0.091 I_f) \quad (5)$$

for $j = 1, 5$ (Thermocouples are in rod positions 2 through 6.)

where,

CDTF = depletion factor at the thermocouple tips in the lower cluster.

7.1.2 Power Equation Corrections

The IFA-518 power equations were analyzed to improve the accuracy of the calculated LHGRs for the rods. In conjunction with these analyses, an in-reactor experiment was performed with IFA-518 and the resulting data were evaluated to determine the effect of neutron streaming at the ends of the shields. A summary of the variables and constants affected by the analyses is shown in Table 7.2. Some of these corrections contribute towards the elimination of the observed 18% error in LHGR. Other corrections are required for shielded and nonshielded operation and do not reduce the error caused by the shield movement.

Fuel Rod Axial Profile

One of the inaccuracies noted in the LHRTF calculations was based on the assumption that the thermocouple tips and the midplane of neutron detector No. 4 (ND4) were located at the same axial position. In actuality, the midplane of ND4 is located 67 mm below the thermocouple tip and a power based on the ND4 midplane flux underpredicts the actual fission rate at the thermocouple tip location.

An axial displacement factor (AX) may be approximated by use of the HBWR axial flux profile. Using this method, a 2.5% power increase between the ND4 midplane and the thermocouple tips was estimated with the HBWR at full power. An AX of 1.025 is considered to be the best available correction for the axial power increase between ND4 and the thermocouple tip and should be multiplied times Equation (5) to correct the LHRTF for the flux profile. This correction does not reduce the observed 18% error because AX is required for both shielded and nonshielded operation.

During power ascensions, the HBWR flux profile changes as shown in Figure 7.3 and the ratio ND4/ND1 indicates that the flux profile also changes depending on reactor power and the shield positions of IFA-518 and the adjacent IFA-507. Although the change in flux profile appears significant

TABLE 7.2. Corrections in IFA-518 Power Equations

Variable or Constant Affected	Variable or Constant Name	Cause of Discrepancy	Effect of Correction	Reduction In Observed 18% Error
MR	Mass Ratio (Cluster)	Power split between the upper and lower cluster is based on a mass ratio. The percentage of mass removed is higher than the percentage of power lost for annular rods and thermocouple holes.	Calculated power in the lower cluster will be an increased proportion of the total assembly power.	None
Mi/M	Mass Ratio (Rod)	Power to each rod is based on the rod's contribution to the total cluster fuel mass. This underpredicts the power in annular fuel regions.	Average LHGR and LHGR at the thermocouple will be higher for annular rods and lower for both vipac and reference rods.	None
AX	Axial Displacement Factor	The thermocouple tips are 67 mm above the ND4 midplane which underpredicts the LHGR at the thermocouples.	Will cause the LHGR at the thermocouple to increase about 2.5% for both shielded and nonshielded conditions.	None
Kg	Calibration Constant	Kg was calculated from the original power equations and subcooled power data. New equations require a new kg.	The current value of 1:128 will decrease to 1.085.	None
LHGR	Transient LHGR	The vanadium neutron detector half life of 3.755 minutes causes the LHGR readings to lag the temperature response by about 5.4 minutes.	Proper correction will eliminate the underprediction of LHGR during power ascensions and the overprediction of LHGR during power decreases.	None
CDTF	Burnup Depletion Factor at the Thermocouple	The CDTF was 1.0 prior to September 26, 1978 but the CDTF was 0.981 immediately after shield movement and acted like a step function. The burnups just before and after the September 26, 1978 shield shift are essentially the same.	Increases the LHGR values at the thermocouple for the shielded condition in comparison with the nonshielded LHGR values at the thermocouple.	1.9%
TE	Thermocouple End Effect Factor	Neutron streaming at the shield ends allows higher flux at the thermocouple tips than at the neutron detector (ND4).	LHGR values at the thermocouple will be higher than previously calculated for the shielded cluster.	5.3%
SE	Integral End Effect Factor	Neutron streaming at both shield ends causes higher power in the fuel rods than indicated by the neutron detectors.	Distributes a greater portion of the total assembly power to the shielded cluster.	3.7%
R	Radial Flux Factor	Thermal flux depression at ND4 caused by absorptions in the close proximity of silver shield.	Will bring the shielded and nonshielded plots of temperature versus LHGR at the thermocouple closer together.	6.0%
HF	Hole Factor	Power at the thermocouple is over-predicted by the axial profile since the thermocouple tip is in an annular fuel region.	Decreases the LHGR at the thermocouple.	None

FIGURE 7.3. Observed Changes in the Ratio Between the Upper and Lower Neutron Detectors in the Lower Cluster of IFA-518

during power ascensions, the actual shape of the flux profile between the locations of ND4 and the thermocouples is uncertain. The correction to AX as a function of reactor power level may only be 1 to 2 percent which could be within the error of graphically obtaining AX from an HBWR flux profile. Thus, this correction is not applied.

Neutron Streaming

While investigating the relative position of ND4 and the thermocouple tips, it was noted that the upper end of the silver shield extends 63 mm above the thermocouple tips and only 9 mm above the 11% enriched fuel region. This suggested that neutron streaming at the end of the shield may cause a higher power at the thermocouple tip than the calculated LHRTF.

The importance of neutron streaming was estimated analytically. Assuming as isotropic neutron source was attenuated 60% when the neutrons passed through the shield, the increase in power at the thermocouple tip due to neutron streaming was estimated to be 2.3%. This was obtained from a triple integral for volume in spherical coordinates where the position of the thermocouple tip and the ND4 midplane were the origins for the spheres. The simplifying assumptions used in this analysis were expected to underpredict the effect of neutron streaming.

An in-reactor experiment was performed to determine the importance of neutron streaming. This experiment involved raising the IFA-518 silver shield approximately 200 mm while changes in ND4 and thermocouple readings were monitored. The measured fuel temperatures decreased as the shield was raised while the ND readings remained constant. No temperature changes were seen after the shield was raised 100 mm. This experiment demonstrated that neutron streaming was definitely affecting the power at the thermocouple tips but not at the neutron detector. Thus, neutron streaming accounts for some of the observed discrepancy in LHGR because the power at the thermocouple was higher with the shield down than indicated by the neutron detectors alone.

The temperature increase at the thermocouple indicated that approximately 5.3% more power resulted from shielded operation than calculated from neutron detector readings. Therefore, Equation (5) for the LHRTF should be

multiplied by a thermocouple end effect factor (TE) of 1.053. This assumes that the percent change in temperature is directly proportional to the power change.

The calculated rod power also requires correction because neutron streaming affects the ALHR. Neutron streaming occurs at both ends of the shielded rods. However, the rods in the upper cluster should have a smaller correction for neutron streaming: the upper cluster has 30 mm of shield below the 11% enriched fuel while the lower cluster rods are shielded only 9 mm above the 11% fuel. Because the region of highest flux is in the top of the lower cluster rods and in the bottom of the upper cluster rods, the total power increase in an upper cluster rod should be less than in a lower cluster rod. However, there is no direct method for determining such a power correction in the upper cluster. An approximation could be made from the shield movement experiment but use of different power corrections for the upper and lower cluster rods would require a different K_g for each cluster.

Finally, the lower cluster rods with thermocouple holes in the fuel have lower powers in the hole region than indicated by the flux profile alone. Because of the difficulties in determining two SE values, an estimated total rod power increase from the streaming effect of 3.7% is being used in the revised power equations. The SE correction factor for ALHRs has a value of 1.037. If each rod contributes similarly to the cluster's neutron streaming, then the cluster power may also be multiplied by SE. Because the SE correction is used to calculate the average power of the rods, the SE factor must also appear in Equation (5) for LHRTFs.

Burnup Corrections

Corrections are made to the calibration constant, K_g , for fissile fuel depletion at the thermocouple tip by the use of CDTF. At the beginning of life, CDTF is 1.0 because no fissile fuel has been consumed. The ALHRs and LHRTFs were calculated by using CD_L , CD_U , and CDTF values of 1.0 up to the first shield shift on September 26, 1978. At that time, the depletion factors were corrected for burnup and CDTF was given a value of 0.981. Because the burnup immediately before and after a shield shift is essentially the same,

the introduction of a new CDTF value caused a 1.9% error between LHRTFs calculated just prior to and after the first shield shift. The CDTF has been reset to 1.0 for present calculations but will be redetermined on a more frequent basis to provide a true comparison of power data.

Radial Flux Depression

Inclusion of SE and TE multipliers in Equation (5) for LHRTFs and equilization of CDTF before and after the shield shift remove 11.7 of the 18.0% error in the reference fuel LHRTFs. The close proximity of the neutron detectors to the silver shield may cause the neutron detector to indicate a depressed power compared with the actual power at the fuel rods in the assembly interior. Correction for this situation is made by multiplying Equation (5) by the radial flux factor (R). A value of 1.063 has been assigned to R for all fuel types. It should be noted that R as well as SE and TE are unity in power equations for the nonshielded condition.

Power Split Between Clusters and Fuel Rods

Although the observed 18% error in LHRTFs is essentially eliminated by SE, TE and R, the power distribution for each cluster and the power among the individual rods must be addressed. Assigning power for each rod strictly on a mass basis results in overestimating the power in rods with no central hole and underestimating the power in annular fuel. This results from the shape of the radial power profile in the fuel which maximizes in the outer region.

Estimates have been made of the actual power reduction resulting from fuel removal at the center of the rods. Calculations using the BRT-I code determined the radial power profile in the reference, annular, and vipac fuels with and without central holes. The effect of fuel removal on rod power is designated as the hole factor (HF) and is shown in Table 7.3 for each fuel type.

An approximation to the power distribution between rods in a cluster may be made by using the HF factors. If the amount of fuel removed from the central hole is added back to the mass of the fuel rods, then a corrected mass $(M_i)^C$ can be calculated to determine the actual power distribution between rods. The use of M_i^C is actually a mathematical expedience because M_i^C does not really exist.

TABLE 7.3. Data Used to Calculate M^C

Rod Positions, i	Rod Type	HF	Pc	M_i , g	M_i^C , g	L_i , m	M_i^C/M^C	$\left(\frac{M_i^C}{M^C}\right) \cdot \left(\frac{1}{L_i}\right)$
1	Annular	0.9596	0.8997	342.2	365.0	0.44893	0.1662	0.3702
2	Reference	0.9856	0.9605	371.9	373.6	0.44486	0.1701	0.3823
3	Vipac	0.9802	0.9496	360.2	362.2	0.44737	0.1649	0.3686
4	Vipac	0.9802	0.9496	363.7	365.7	0.44865	0.1665	0.3712
5	Annular	0.9596	0.8997	341.8	364.6	0.44907	0.1660	0.3696
6	Annular	0.9596	0.8997	342.2	365.0	0.44836	0.1662	0.3707
7	Annular	0.9596	0.8997	343.5	366.4	0.44934	0.1656	0.3686
8	Reference	1.0	1.0	373.9	373.9	0.44478	0.1690	0.3799
9	Vipac	1.0	1.0	367.4	367.4	0.44676	0.1661	0.3718
10	Vipac	1.0	1.0	369.1	369.1	0.44740	0.1668	0.3728
11	Reference	1.0	1.0	374.1	374.1	0.44450	0.1691	0.3804
12	Annular	0.9596	0.8997	339.1	361.7	0.44819	0.1635	0.3648

If a flat axial profile is assumed then M_i may be calculated from the following:

$$M_i^C = M_i \cdot (1 - VR) + VR \cdot M_i \cdot HF/P_c \quad (6)$$

where,

VR = fuel volume in the region with a central hole divided by the total volume of the fuel

$$= \frac{1}{1 + \frac{L_s}{P_c \cdot L_h}}$$

L_s = length of the fuel rod containing solid fuel

L_h = length of the fuel rod containing a central hole

P_c = fraction of power left in a fuel rod as calculated by the fraction of mass removed.

$$= 1 - \frac{D_{id}^2}{D_{od}^2}$$

D_{id} = inside diameter of the fuel with a central hole, mm

D_{od} = outside diameter of the fuel, mm

Equation (6) should be corrected for the dish at both ends of the reference pellet, but the dishes are not geometrically similar, they cover about twice the diameter of the central hole, and the total dish volume is only 1.45% of the HBWR Series H-1 reference pellet. Thus, this correction is probably not significant for any of the corrected masses.

The M_i^C values calculated from Equation (6) are listed in Table 7.3 along with the corrected rod-to-cluster mass ratios (M_i^C/M^C). These ratios were determined by adding the M_i for each rod to obtain the corrected

cluster mass (M^C). The corrected masses cause the calculated LHRTFs of vipac and reference rods with thermocouples to decrease 1 to 2% while the corrected LHRTFs of annular rods increase 2 to 3%. The corrected masses also result in a MR of 1.01.

An additional correction resulting from the central hole is the reduction of the LHRTFs calculated for vipac and reference fuel with thermocouples. The original power equations did not account for the power increase between the ND4 midplane and the thermocouple tip when based on the flux profile; however, the power profile will show a discontinuity at the bottom of the thermocouple hole in solid fuel rods. The axial flux profile will then cause the power to continue to rise according to the change in flux. The HF may be multiplied times Equation (5) for reference and vipac rods to account for the power loss in the thermocouple region. Annular rods have an axial power profile that is the same shape as the axial flux profile and do not need this correction at the thermocouple tip.

Vanadium Neutron Detector Delay Time

The half life of ^{52}V which is used in the neutron detectors, is 3.755 min which results in a mean life of 5.42 min. This causes the calculated power to lag the fuel temperature response during power ascensions and power decreases. One method of correcting the neutron detector readings involves interpolating approximately 5.4 min beyond a specific time of interest and approximating the correct neutron detector reading. FIX 1 has been written to perform this task for IFA-518 data now on disk storage. The net effect of these neutron detector corrections is to increase rod powers during power ascensions and decrease the rod powers during reactor power decreases.

Calibration Constant

Because the calibration constant was determined from Equation (1) using subcooled flow data, any alterations to this equation will require recalculation of K_g to satisfy the original subcooled power. One method of determining the new K_g is to substitute values into a new Equation (1) and obtain the new K_g by ratios. If the nonshielded cluster provides

approximately 60% of the total rig power and the shield up and shield down gamma heating contributions are 4.0% and 4.7%, respectively, then the new K_g is 1.085 averaged for both shield positions based on an original average K_g of 1.175 kW/nanoamps uncorrected for gamma heating.

7.1.3 Revised Power Equation Results

The corrections made to the rod powers in IFA-518 have significantly reduced the observed 18% error in LHRTFs calculated with the shield up and down. Figure 7.4 shows the results of incorporating all of the corrections. There is a slight disagreement between LHGRs calculated for the shield up and down but this error is comparable to the range of LHGRs calculated for several power ascensions with the same shield position. Although improved corrections for fissile fuel depletion, axial power profile changes, and neutron detector delay time may be possible and even warranted in the future, the current power equations offer a distinctly improved capability in fuel performance evaluations.

7.1.4 Thermocouple Reliability

The reliability of IFA-518 thermocouple measurements was checked using the code MWRAM (7.1,7.2,7.3). This program predicts fuel temperatures during power decreases, compares the predicted and actual temperatures, and provides a measure of thermocouple reliability. MWRAM was developed for previous HBWR experiments with pellet fuel rods. Using transient power data from IFA-518 operation up to January 1979, MWRAM showed that thermocouple No. 3 (TF3) in vipac rod No. 13 is unreliable as was suggested previously. TF1 and TF2 in reference rod No. 2 and vipac rod No. 14, respectively, demonstrated reliable operation up to January 1979.

7.2 HBWR SERIES H-4 PERFORMANCE EVALUATION

The first cycle of irradiation for the reference and sphere-pac fuel rods in HBWR Series H-4 began in April and ended in late May. Analyses of these IFA-517 fuel rods included a preliminary comparison of rod elongation and fuel temperatures as well as the calculation of the iodine inventory in the reference rod after a projected 432 GJ/Kg (5 MWd/Kg) burnup.

FIGURE 7.4. Comparison of Corrected and Uncorrected Power Data Prior To and After the September 28, 1978 Shield Shift.

An initial evaluation of rod elongation data for the sphere-pac and reference rods was made, but the evaluation is complicated by the hard wiring of the elongation sensors in both IFA-517 and IFA-518. This problem was discussed in Section 5.3.1. While the absolute values of the elongation readings are questionable, the relative change in rod elongations should be correct. Further analysis of the elongation data is in progress.

The measured fuel temperature in the sphere-pac rod is significantly lower than the temperature for vipac rod no. 14 in IFA-518. At 40 kW/m, vipac rod No. 14 operated $\sim 200^{\circ}\text{C}$ higher than the reference rod in IFA-518, but both sphere-pac rods operated $\sim 145^{\circ}\text{C}$ below the reference rod in IFA-517. The fuel temperatures for the two sphere-pac rods are within 2% of one another as shown in Figure 7.5. Sphere-pac 1 and Sphere-pac 2 in Figure 7.5 are rod Nos. 41 and 42, respectively. The reference rods in IFA-517 and IFA-518 are operating within 10°C of one another at 40 kW/m using IFA-supplied LHGRs for IFA-517 and PNL-corrected LHGRs for IFA-518. Sphere-pac fuel temperatures in IFA-517 are $\sim 105^{\circ}\text{C}$ higher than the temperatures for the annular rods in IFA-518.

The sphere-pac rods are pressurized at 354 kPa helium pressure whereas the vipac rods are not pressurized. Although the helium pressurization of the sphere-pac rods has a significant effect in reducing fuel temperatures, the sphere-pac thermal conductivity may also be better than for the vipac fuel. GAPCON-THERMAL-3 calculations indicate that the absence of thermocouple holes in vipac and reference fuel rods would increase fuel temperatures by 175°C and 80°C , respectively. It is expected that this effect would be similar for sphere-pac fuel without thermocouple holes.

Using a typical IFA-517 LHGR of 40 kW/m, the iodine concentration over the active fuel length in the reference rod was estimated for a burnup of 432 GJ/kg (5 MWd/Kg). Only ^{131}I and ^{129}I production were considered because most of the iodine inventory is composed of these two isotopes. Assuming equilibrium conditions and 100% release, the cumulative yield of $8.04\text{d } ^{131}\text{I}$ is 2.77% resulting in $22.1\text{ }\mu\text{g}/\text{cm}^2$ of ^{131}I . This ^{131}I concentration

FIGURE 7.5. Comparison of Reference and Sphere-Pac Fuel Temperature Behavior in IFA-517

would decay to $\sim 10 \mu\text{g}/\text{cm}^2$ within 10 days after reactor shutdown. The essentially stable ^{129}I is formed from both 74.4 min ^{129}Te and 33d $^{129\text{m}}\text{Te}$. Assuming the ^{129}I results from only ^{129}Te decay, $50.4 \mu\text{g}/\text{cm}^2$ of iodine-129 accumulates after 92 days at 40 kW/m. What portion of iodine is released and what fraction remains as free iodine is not known; however, based on a reported (7.4) threshold for stress corrosion cracking of $\sim 1 \mu\text{g}/\text{cm}^2$, even a 2% release as free iodine-129 would provide sufficient iodine for failure under the necessary stress/strain conditions. However, at a burnup of only 432 GJ/kg, cladding stress due to fuel relocation and fuel-cladding contact is probably insufficient to cause stress-corrosion cracking.

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DEFINITION OF THE VARIOUS FUEL ROD TYPES^(a)

<u>Fuel Type</u>	<u>Description</u>
Standard	Standard ENC "product-line." Solid, cylindrical, dished-end-fuel pellet.
Reference	Solid, cylindrical, dished-end, chamferred-corner (except HBWR Series H-1) fuel pellet.
Annular	Cylindrical, flat-ended, chamferred-corner (except HBWR Series H-1) fuel pellet with a central hole equivalent to 10 vol% of a solid, undished pellet.
Vipac	Packed-particle fuel composed of high-sensity, angular, fuel shards, produced by high-energy pneumatic compaction (Dynapak). The rods are pressurized with 400-500 kPa of helium, except in the HBWR Series H-1, and the lower segments of the BRPR segmented rods.
Sphere-pac	Packed-particle fuel composed of high-density, spherical particles produced by the sol-gel process. The rods are pressurized with 400-500 kPa of helium.
Coated-cladding	Cladding coated with Dag 4 ^(b) graphite, except HBWR Series H-1 which utilizes Dag 154 ^(b) graphite.
Pressurized	Rods pressurized with 400-500 kPa of helium.
Reference-coated	Reference fuel pellets combined with coated cladding.
Annular-coated	Annular fuel pellets combined with coated cladding.
Reference-pressurized	Reference fuel pellets in a pressurized rod.
Annular-coated-pressurized	Annular fuel pellets combined with coated cladding in a pressurized rod.

(a) All rods are clad with cold-worked and stress-relieved Zircaloy-2 cladding.

(b) Product of Acheson Colloids Corp., Port Huron, MI.

8.0 TASK 4 - ESTABLISHMENT OF TECHNICAL BASES

K. R. Merckx (Exxon Nuclear Company)

The current major activities of Task 4 are:

- Establishing a detailed technical bases for the thermal conductivity of sphere-pac uranium fuel,
- Transferring the applicable data bases from the FRDA program into the FPIP technical bases,
- Updating the history generation capability for the performance files of FPIP supported irradiations, and
- Revising the existing technical bases for the FPIP.

A DOE supported task of the Fuel Refabrication and Development Program (FRAD) evaluated fuel properties for several fuel compositions and forms. An extensive review and evaluation of vipac and sphere-pac fuel properties were included as part of this task. The reference file structure for this work was compatible with that used in FPIP. The transfer of sphere-pac and vipac references and editing of the resultant file structure is partially completed. Improvements have also been incorporated into the reference file reading program which include editing functions of keyword and format checks, and/or search-routines with various output formats and reference output suitable for direct use in reports.

Theoretical results for evaluating the thermal conductivity of sphere-pac fuel have been obtained that outline the different aspects of the dependence of the thermal conductivity on key parameters. For instance thermal conductivity has been considered with respect to fill gas pressure, temperature, content of fission gas in fill gas, and extent of restructuring. These preliminary results will be checked with experimental data to assess the accuracy.

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APPENDIX A

CONVERSION TABLE FOR SI UNITS

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CONVERSION TABLE FOR SI UNITS^(a)

	To convert from	to	Use equation or multiply by
Temperature	degree Fahrenheit (°F)	degree Celsius (°C)	$^{\circ}\text{C} = (^{\circ}\text{F} - 32)/1.8$
	degree Fahrenheit (°F)	kelvin (K)	$\text{K} = (^{\circ}\text{F} + 459.67)/1.8$
	degree Celsius (°C)	kelvin (K)	$\text{K} = ^{\circ}\text{C} + 273.15$
Time	hour (hr)	second (s)	3.600×10^3
	day (d or day)	second (s)	8.640×10^4
Length	inch (in.)	metre (m)	2.540×10^{-2}
Area	inch ² (in. ²)	metre ² (m ²)	6.452×10^{-4}
Volume	inch ³ (in. ³)	metre ³ (m ³)	1.639×10^{-5}
Pound-Mass	lbm	kilogram (kg)	4.536×10^{-1}
Pound-Force	lbf	newton (N)	4.448
Pressure or stress	psi (lbf/in. ²)	pascal (Pa)	6.895×10^3
Energy ^(b)	British thermal unit (Btu)	joule (J)	1.055×10^3
	calorie (cal)	joule (J)	4.187
	watt-second (W-s)	joule (J)	1.000
Linear Heat Rating	kilowatts/ft (kW/ft)	kilowatts/metre (kW/m)	3.281
Burnup	megawatt-day/metric ton of	gigajoule/kg of heavy	8.640×10^{-2}
	heavy metal ^(c) (MWd/MTM)	metal ^(c) (GJ/kgM)	

(a) Conversions are based on ASTM Metric Practice Guide E380-74. Four of the prefixes (*) are to be avoided where possible.

Prefix	SI Symbol	Multiplication Factors
tera	T	$1\,000\,000\,000\,000 = 10^{12}$
giga	G	$1\,000\,000\,000 = 10^9$
mega	M	$1\,000\,000 = 10^6$
kilo	k	$1\,000 = 10^3$
hecto*	h	$100 = 10^2$
deka*	da	$10 = 10^1$
deci*	d	$0.1 = 10^{-1}$
centi*	c	$0.01 = 10^{-2}$
milli	m	$0.001 = 10^{-3}$
micro	μ	$0.000\,001 = 10^{-6}$
nano	n	$0.000\,000\,001 = 10^{-9}$
pico	p	$0.000\,000\,000\,001 = 10^{-12}$
femto	f	$0.000\,000\,000\,000\,001 = 10^{-15}$
atto	a	$0.000\,000\,000\,000\,000\,001 = 10^{-18}$

(b) 10^6 Btu = 1.055 GJ = 0.17 barrels of oil = 100 kW-hr, assuming one barrel of oil = 42 gallons and the conversion efficiency from heat to electricity is 0.33.

(c) M is normally the SI symbol for mega; an exception in this report is the additional use of M in burnup units to denote heavy metal.

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