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Semiannual Progress Report

April 1980 - September 1980

October 1980

Consumers
Power
Company

EXXON NUCLEAR
COMPANY, Inc.

Battelle

Pacific Northwest Laboratory

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Progress Report****April 1980 -
September 1980****October 1980****C.E. Crouthamel
Program Manager
Exxon Nuclear Company, Inc.
Richland, Washington 99352****M.D. Freshley
Project Manager
Pacific Northwest Laboratory
Richland, Washington 99352****DISCLAIMER**

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L.P. Bupp
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ABSTRACT

Progress on the Fuel Performance Improvement Program's fuel test and demonstration irradiations is reported for the period April-September, 1980. Included are results of out-of-reactor experiments with zircaloy cladding on the iodine assisted stress corrosion cracking mechanism. Preliminary results from the first eight ramp tests performed in the Halden Boiling Water Reactor are reported. The status of demonstration fuel irradiations in the Big Rock Point Reactor is described.

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SUMMARY

The Fuel Performance Improvement Program (FPIP) is being conducted for the U.S. Department of Energy by Consumers Power Company, Exxon Nuclear Company, Inc., and Pacific Northwest Laboratory^(a) to develop LWR fuels with improved performance. The program is designed to obtain data on and evaluate the irradiation and ramping behavior of improved fuel concepts, and to review the licensing requirements and update the technical bases for these concepts. An annular-coated-pressurized and a sphere-pac design are being evaluated in steady-state and ramping irradiations and their performance is being compared with a reference fuel design.

Results of recent out-of-reactor experiments to study zircaloy iodine-SCC failure using the PCI-simulator have shown high strain to failure values for susceptible zircaloy cladding which previously failed at low strains. Rigorous vacuum outgassing at 425°C and handling of the specimens in an inert atmosphere prior to testing resulted in consistently low strain-to-failure values for noncoated, susceptible zircaloy cladding specimens. The same pre-test treatment resulted in consistently high strain-to-failure values for noncoated cladding specimens that have been shown to be resistant to iodine-SCC. Graphite-coated, susceptible cladding specimens exposed to ambient air for very short periods after outgassing also produced high strain-to-failure results. However, the coated susceptible specimens failed at low strains when the test specimens were handled exclusively in an inert atmosphere box.

These results demonstrate the marked effect of oxygen potential on the iodine-SCC test results with zircaloy. Also, the graphite coating on the zircaloy does not appear to function as a barrier to the reaction with iodine under very dry, oxygen-free conditions.

The first series of power ramp tests on eight rods irradiated in the Halden reactor were completed. The test rods ramped included two reference pellet rods, two annular-coated rods, one annular rod, two sphere-pac rods, and one vipac rod. The rods were ramped to peak LHGRs of about 70 kW/m with

(a) PNL is operated by Battelle Memorial Institute.

the peak differential LHGRs of 40 kW/m. None of the test rods failed. However, the axial rod elongation behavior during the ramps clearly shows significant differences in the performance of the various fuel designs. Analysis of the ramp test data has not been completed; however, the following preliminary conclusions are made:

1. The PCMI behavior of the annular-coated fuel rod is superior to that of the reference fuel rods on the basis that the annular-coated Rod AC9 exhibited, (a) less axial cladding strain, (b) a higher LHGR for the onset of PCMI, and (c) a gradual onset of hard PCMI.
2. The PCMI behavior of the sphere-pac fuel rod is superior to that of the reference pellet rods on the basis that sphere-pac Rod S41 exhibited, (a) less axial cladding strain, (b) lower strain rate during moderate PCMI, and (c) essentially zero residual cladding strain after the ramp.
3. The PCMI behavior of the vipac fuel rod is superior to that of the reference pellet rods on the basis that vipac Rod V13 exhibited, (a) less axial strain, and (b) essentially zero residual strain after the ramp.

The average center fuel temperature in an instrumented sphere-pac rod was determined to be about 110°C lower than the center fuel temperature in a reference pellet rod at the beginning of irradiation in the HBWR. This is about 15°C less than the previously reported temperature difference with both types of rods operating at an LHGR of about 36 kW/m. At a burnup of 520 GJ/kgM (6 MWd/kgM), the temperature difference was reduced to about 65°C, probably the result of fission gas degradation of the helium fill-gas thermal conductivity.

A total of 367 demonstration rods have been fabricated for irradiation in the Big Rock Point Reactor (BRPR). The assemblies containing the Series S-2 and S-4 rods were shipped to BRPR for loading in the fall of 1980. The total of nine FPIP assemblies in the BRPR core contain all the 56 segmented rods in the program. The remaining 108 full-length rods have been fabricated and will be loaded in BRPR as Series S-3 in the fall of 1981.

The general scope of the technical bases being developed for FPIP fuel candidates has been expanded from design to include general performance and licensing considerations. The Electric Power Research Institute Fuel Performance Data Base System has been adopted and data entries have begun.

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CONTENTS

ACKNOWLEDGMENTS	iii
ABSTRACT	v
SUMMARY	vii
DESCRIPTION OF THE VARIOUS FUEL ROD TYPES	xv
LIST OF FPIP PUBLICATIONS	xvii
1.0 INTRODUCTION	1
1.1 PROGRAM OBJECTIVES	1
1.2 BACKGROUND	1
1.3 PROGRAM ORGANIZATION, OVERALL STATUS AND SCHEDULE	2
2.0 TASK 2A -- OUT-OF-REACTOR STUDIES	9
2.1 PCI SIMULATION TESTS	9
2.2 SPHERE-PAC MICROSPHERE OUTGASSING STUDIES	13
3.0 TASK 2B -- IN-REACTOR EXPERIMENTS	17
3.1 SUMMARY - HBWR IRRADIATION AND RAMPING PLAN	17
3.2 FABRICATION--HBWR SERIES H-2, H-3 AND H-4 RODS	17
3.3 HBWR STEADY-STATE IRRADIATION STATUS	17
3.4 HBWR POWER-RAMP TESTING	19
3.5 HBWR STEADY-STATE DATA ACQUISITION AND PROCESSING	21
3.6 PRIMARY RAMPING TEST STATUS	25
4.0 TASK 2C -- IN-REACTOR DEMONSTRATIONS	31
5.0 TASK 3 -- FUEL PERFORMANCE EVALUATIONS	41
5.1 SPHERE-PAC FUEL CENTERLINE TEMPERATURE MEASUREMENTS	41
5.2 LHGR ONSET FOR PCMI	41
5.3 POWER-RAMP RESULTS	45
5.4 THERMOCOUPLE DECALIBRATION	51
6.0 TASK 4 -- ESTABLISHMENT OF TECHNICAL BASES	53
6.1 TECHNICAL BASES	53
6.2 DATA AND REFERENCE FILE DEVELOPMENT	53
7.0 REFERENCES	55
APPENDIX A: CONVERSION TABLE FOR SI UNITS	A.1

FIGURES

1.1	Organization of the Fuel Performance Improvement Program	3
1.2	FPIP Schedule	6
1.3	FPIP Irradiation and Ramping Schedule	7
2.1	PCI Simulator Test Results for Nonsusceptible and Susceptible Zircaloy Cladding in the Noncoated Condition	10
2.2	PCI Simulator Test Results for Dag 4-Coated Nonsusceptible and Susceptible Zircaloy Cladding	11
2.3	Strain-to-Failure for Susceptible, Dag 4-Coated Zircaloy Cladding as Affected by Air Exposure After Outgassing	12
2.4	Water and Total Weight Loss of Fine UO ₂ Microspheres at Temperatures to 1000°C	15
3.1	Idealized Power Ramping Scheme	20
3.2	Rod Elongation During Initial Irradiation of Lower Cluster Rod ACP27 in IFA-518.2	23
3.3	Rod Elongation During Initial Irradiation of Lower Cluster Rod ACP29 in IFA-518.2	24
3.4	Rod Elongation During Initial Irradiation of Lower Cluster Rod R23 in IFA-518.2	25
3.5	Rod Elongation in Rod ACP27 Corrected for the Effect of Subcooling	26
3.6	Rod Elongation in IFA-518.2 Lower Cluster Annular-Coated-Pressurized Rods ACP27 and ACP29	27
3.7	Rod Elongation in IFA-518.2 Upper Cluster Rods Illustrating "Drift" in Fresh Rods	28
3.8	Elongation of Previously Irradiated Rods in IFA-518.2	29
3.9	Rod Elongation in Rod ACP27 from IFA-518.2 after Correction for Subcooling and "Drift" during the Initial Power Ascension	30
4.1	BRPR Series S-4 Fuel Assembly (H1-20) Map	38
4.2	BRPR Series S-4 Fuel Assembly (H1-21) Map	39

5.1	Corrected Measured Fuel Centerline Temperature During Initial Startup in IFA-517.1 Sphere-Pac and Reference Rods	42
5.2	Corrected Measured Fuel Centerline Temperature at 36 kW/m as a Function of Burnup in IFA-517.1 Sphere-Pac and Reference Rods	43
5.3	The Onset of PCMI During Reactor Startups	44
5.4	LHGR Required for the Onset of PCMI in Reference, Annular, and Annular-Coated Rods	45
5.5	Rod Elongations in Reference Rod R2, Annular-Coated Rod AC9, and Vipac Rod V13 from HBWR Series H-1 During Ramp Testing	47
5.6	Cladding Elongations in Reference Rod R1 and Sphere-pac Rod S41 from HBWR Series H-4 During Ramp Testing	48

TABLES

3.1	HBWR Irradiation Plan	18
3.2	Approximate Burnups of IFA-518.2 Rods	19
3.3	Preliminary Summary of Power-Ramping Condition	22
4.1	Big Rock Point Reactor Series S-1 Matrix (4 Assemblies: G-419, G-420, G-421, and G-422)	32
4.2	BRPR Series S-2 Matrix (3 Assemblies: H-101, H-102, and H-103)	32
4.3	BRPR Series S-3 Matrix (3 Assemblies: As yet undesignated)	33
4.4	BRPR Series S-4 Matrix (2 Assemblies: H-120 and H-121)	33
4.5	Summary of Demonstration Irradiations for the Big Rock Point Reactor	33
4.6	Design Parameters for BRPR Series S-4 Sphere-Pac Rods	34
4.7	BRPR Series S-4 Rods in Assembly H1-20	36
4.8	BRPR Series S-4 Rods in Assembly H1-21	36
4.9	Summary Status of Demonstration Assemblies for BRPR	37

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DESCRIPTION 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-density, 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 utilized 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.

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1. "Fuel Performance Improvement Program, Quarterly Progress Report, April-June 1977," compiled by C. E. Crouthamel, C00-4066-1, July 1977.
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.
3. W. J. Bailey and J. O. Barner, "Assessment of Fuel Concepts," C00-4066-3, PNL-2490, January 1978.
4. "Fuel Performance Improvement Program, Quarterly/Annual Progress Report, April-September 1977," compiled by C. E. Crouthamel, C00-4066-4, November 1977.
5. "Fuel Performance Improvement Program, Quarterly Progress Report, October-December 1977," compiled by C. E. Crouthamel, C00-4066-5, January 1978.
6. "Fuel Performance Improvement Program, Quarterly Progress Report, January-March 1978," compiled by C. E. Crouthamel, C00-4066-6, April 1978.
7. "Fuel Performance Improvement Program, Quarterly Progress Report, April-June 1978," compiled by C. E. Crouthamel, C00-4066-7, July 1978.
8. "Fuel Performance Improvement Program, Quarterly Progress Report, October 1977-September 1978," compiled by C. E. Crouthamel, C00-4066-8, October 1978.
9. "Fuel Performance Improvement Program, Quarterly Progress Report, October-December 1978," compiled by C. E. Crouthamel, C00-4066-9, January 1979.
10. "Fuel Performance Improvement Program, Quarterly Progress Report, January-March 1979," compiled by C. E. Crouthamel, C00-4066-10, April 1979.
11. F. W. Buckman, C. E. Crouthamel and M. D. Freshley, "FCI-Remedy Development for the Fuel Performance Improvement Program," Proceedings Light Water Reactor Fuel Performance, Portland, Oregon, April 29 to May 3, 1979, pp. 158-168.
12. 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.

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14. S. R. Wagoner, J. O. Barner and R. J. Guenther, "Comparative Performance of FCI-Remedy Fuel Designs," Trans. Am. Nucl. Soc. 33 (1979), pp. 271-272.
15. "Fuel Performance Improvement Program, Quarterly/Annual Progress Report, October 1978-September 1979," compiled by C. E. Crouthamel, COO-4066-13, October 1979.
16. "Fuel Performance Improvement Program Description and Characterization of HBWR Series H-2, H-3, and H-4 Test Rods," R. J. Guenther, J. O. Barner, and R. K. Welty, DOE/ET/32415-14, March 1980.
17. "Fuel Performance Improvement Program, the Development of Graphite-Coated Cladding for Improved PCI Performance of LWR Fuels," L. R. Bunnell and R. K. Welty, DOE/ET/32415-15, September 1980.
18. "Fuel Performance Improvement Program, Semiannual Progress Report, October 1979-September 1980," compiled by C. E. Crouthamel and M. D. Freshley, DOE/ET/34215-16, April 1980.

1.0 INTRODUCTION

1.1 PROGRAM OBJECTIVES

The objectives of the Fuel Performance Improvement Program (FPIP) are to identify and demonstrate fuel concepts with improved power ramping capabilities and to provide the supportive technical bases for developing commercial fuel designs derived from these concepts. As a result of their improved performance, the capability of the improved fuel concepts to achieve high burnup for better utilization of uranium will be enhanced.

1.2 BACKGROUND

The FPIP is focused on developing LWR fuel designs applicable to both BWRs and PWRs that exhibit improved fuel-cladding interaction behavior during rapid power changes or ramping. The primary fuel design being evaluated is the annular-coated-pressurized design. A sphere-pac fuel design is currently being evaluated at a more modest level. The performance of these designs is being determined by comparison with the performance of a reference pellet design. An important part of the integrated program is that it will not only demonstrate the improved performance of the advanced designs, but will provide the supportive technical bases for facilitating the commercialization of these designs.

One of the prime program requirements is the commercial applicability and acceptability of the results. Toward this end, the fuel concepts being evaluated were selected, in part, because of an existing technology and experience base to aid rapid commercialization and commercial practices were employed for fuel fabrication and quality assurance. Power ramping performance and performance limits are being evaluated in test reactors while full-term irradiations of a significant number of rods are being conducted in the Big Rock Point Reactor (BRPR) to demonstrate that the fuel concepts do not constitute any high inherent risks. The tests and demonstrations are designed to provide definitive data that are used to develop the technical bases in support of the design

and licensing of the fuel concepts on a lead test assembly, demonstration scale. The detailed program description and history have been presented in previous progress reports.^(1,2)

1.3 PROGRAM ORGANIZATION, OVERALL STATUS AND SCHEDULE

1.3.1 Program Organization

The program is being 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) and has subcontracted most of the work to ENC and PNL. The functional organization of the program, according to the task structure, is illustrated in Figure 1.1.

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 designated ENC to manage the 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 interpreting the results.

Task 1, Assess State-of-the-Technology and Potential Improved Concepts, which was completed early in the program, evaluated the state-of-the-technology with respect to fuel concepts with potential for improved PCI performance and provided the rationale for selecting those fuel concepts being evaluated.

Task 2A, Out-of-Reactor Experiments, activities provide data to support the technical design bases and aid in the interpretation of the irradiation experiments. Task 2B, In-Reactor Experiments, consists of a testing program conducted with instrumented fuel rods in the Halden Boiling Water Reactor (HBWR), followed by ramping of individual rods in the HBWR ramping rig, and tests that involve ramping of segmented rods after their irradiation to significant burnup levels in the Big Rock Point Reactor (BRPR). Task 2C, In-Reactor Demonstrations, involves the design, process development, fabrication,

FIGURE 1.1. Organization of the Fuel Performance Improvement Program

quality control, licensing, and demonstration irradiations in BRPR of a significant number of fuel rods comprising the primary fuel concepts.

The objective of Task 3, Performance Evaluation, is to analyze the data developed in the FPIP and evaluate the performance of the fuel concepts under study. These results are then utilized by the activities included in Task 4, Establishment of Technical Bases, to establish technical bases for design and the design criteria for the improved concepts. These technical bases for design and the design criteria form the basis of licensing submittals and facilitate the transfer of the technology to the industry.

1.3.2 Overall Program Status

Task 2A out-of-reactor experiments currently include PCI-simulator tests, which are being performed to define the function of the graphite coating to improve performance. Experiments were also performed to study the moisture

sorption/desorption characteristics of UO_2 microspheres used in sphere-pac fuel rods. These studies are described in Section 2.0.

Task 2B in-reactor experiments activities currently involve data acquisition associated with the continued steady-state irradiation and ramp testing of instrumented rods in HBWR. The current status of the in-reactor experiments is described in Section 3.0.

Eight additional segmented sphere-pac rods were fabricated for irradiation in BRPR as part of the Task 2C in-reactor demonstrations. Other activities are focused on following the progress of the irradiation of the 4 FPIP assemblies in BRPR which have achieved assembly-average burnups as high as 7500 MWd/kgM. The status of the in-reactor demonstrations is described in Section 4.0.

The results of the sphere-pac fuel centerline temperature measurements were re-evaluated as part of the Task 3 performance evaluations, Section 5.0. In addition, an evaluation was made to compare the LHGR for the onset of PCMI for the different fuel types, the evaluation of the ramp test results for the first 8 rods was continued, and the effects of in-reactor thermocouple decalibration were evaluated.

The scope of the technical bases reports under Task 4, Establishment of Technical Bases, has been expanded from only design to include performance and licensing considerations. Draft technical bases reports which address coated cladding and annular pellets were completed. The EPRI Fuel Performance Data Base System has been adopted for FPIP and the software for this system has been implemented on the ENC PRIME computer. Characterization data for the HBWR series test rods have been entered into this system. These activities are described in Section 6.0.

1.3.3 Program Schedule

The FPIP is scoped and activity milestones are established to meet the objective of developing definitive results needed to proceed with the design and licensing of lead test assembly demonstrations of the primary fuel concept in commercial BWRs and PWRs before the contract date of December 31, 1984.

This is being accomplished by performing the development activities according to a logical progression, although some activities are conducted concurrently (Figure 1.2). The activities and their general progression include supportive out-of-reactor experiments, instrumented test reactor irradiations, severe ramping tests, demonstration irradiations in a commercial BWR, ramping tests of the primary concept under prototype conditions, concurrent performance evaluations, and development of the technical bases. The schedule interactions between the test reactor irradiations and ramp tests and the demonstration irradiations and ramp tests are illustrated in Figure 1.3.

A detailed discussion of the progress according to each task activity is presented in the following sections.

FIGURE 1.2. FPIP Schedule

CYCLING OR STEADY-STATE IRRADIATION

RAMPING TEST

PIE

----- RAMPING INFO FLOW

FIGURE 1.3. FPIP Irradiation and Ramping Schedule

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2.0 TASK 2A OUT-OF-REACTOR STUDIES

2.1 PCI SIMULATION STUDIES

The PCI simulator is a load application system that uses a fractured alumina annulus to simulate the loading conditions that occur in the cladding at the pellet-pellet interfaces. All the tests have been performed with non-irradiated cladding specimens.^(3,4) As a matter of background, 12 identical tests of FPIP cladding (Lot 3775A) produced widely scattered values for average hoop strain-to-failure (Figure 2.1). It is hypothesized that the reason for the wide scatter is that this cladding, which has been shown to be highly resistant to iodine stress-corrosion cracking (iodine-SCC),⁽⁵⁾ has a low population of crack initiating defects, and the small area under load in the PCI-simulator results in a low probability of encountering such a defect. On the other hand, when cladding that was known to be highly susceptible to iodine-SCC failure (Lot 6719) was tested under the same conditions, the failure strain was low (average 1.0%) and very reproducible ($\sigma = 0.18\%$, 12 tests) (Figure 2.1). When the inside surfaces of both types of cladding were coated with a 6 μm -thick layer of Dag 4, outgassed, and tested, neither failed at average hoop strains of up to 8% (Figure 2.2). The beneficial effects of graphite to improve PCI-behavior seemed amply demonstrated by these results.

During subsequent testing of noncoated, susceptible cladding in the PCI-simulator, low-strain failures did not occur as they had previously, even though the test parameters and procedures had been kept constant. Several possible explanations, all of which involved an increase of water or oxygen concentration in the test section from an unknown source, were evaluated. Evaluating the effect of reducing the water/oxygen concentration depended upon having cladding that was known to be susceptible to iodine-SCC failure. However, virtually all of the tubing from Lot 6719 had been consumed in the earlier tests. As a result, it was thus necessary to use other cladding from the same vendor as Lot 6719 (Lots 5484 and 8182) without assurance they were highly susceptible to iodine-SCC. For other tests, the ends from previously tested Lot 6719 cladding specimens were welded between other cladding segments so that the susceptible cladding was in the central loaded zone.

FIGURE 2.1. PCI Simulator Test Results for Nonsusceptible (Lot 3775A) and Susceptible (Lot 6719) Zircaloy Cladding in the Noncoated Condition. Test conditions were 300°C, 2% (hr)⁻¹ strain rate, 6 mg I₂/cm² cladding.

Over 40 tests were performed using longer purge times for flushing air from the test section and using freshly dried iodine, as well as other measures to reduce the water/oxygen concentration. None of these measures resulted in low-strain failures. However, when the test section components were assembled and sealed in an inert-atmosphere glove box, noncoated specimens from Lots 6719, 8182, and 5484 all exhibited low-strain (ca. 1%) failures. FPIP cladding from Lot 3775A continued to generally fail at high strain, if at all, which was in agreement with the previous results. Hence, it is concluded that the strain-to-failure in susceptible cladding when tested in iodine, is highly sensitive to the partial pressure of oxygen/water.

During previous tests performed with coated cladding specimens (Figure 2.2), the coated specimens were exposed to air for a short time prior to testing with no redrying step. Thus, some unknown amount of water/oxygen

FIGURE 2.2. PCI Simulator Test Results for Dag 4-Coated Nonsusceptible (Lot 3775A) and Susceptible (Lot 6719) Zircaloy Cladding. Test conditions were 300°C, 2% (hr)⁻¹ strain rate, 5 mg I₂/cm² cladding.

was sorbed by the coating and was present in the test section. When coated, susceptible cladding specimens were vacuum outgassed at 425°C and transferred to the inert-atmosphere drybox with no air exposure prior to testing, low-strain failures resulted (Figure 2.3). The same effect was noted when, in an attempt to delineate the physical and chemical effects of graphite to improve PCI behavior, the coating was applied to the Al₂O₃ annulus which loads the cladding instead of to the inside surface of the cladding (Figure 2.3). If the coated Al₂O₃ was outgassed and exposed to air prior to testing, the failure strain of susceptible cladding was high. On the other hand, if no air exposure was permitted, the failure strain was low. Therefore,

FIGURE 2.3. Strain-to-Failure for Susceptible, Dag 4-Coated Zircaloy Cladding as Affected by Air Exposure After Outgassing. Test conditions were 300°C, 2% (hr)⁻¹ strain rate, 6 mg I₂cm² cladding.

it was shown that for low oxygen/water concentrations, graphite coating does not produce a dramatic increase in the strain-to-failure under these test conditions. This conclusion supports the hypothesis that the efficacy of graphite coatings to improve PCI behavior is related to its effect on internal rod chemistry as well as reducing the coefficient of friction between the fuel pellets and the cladding.

The pronounced dependency of iodine-SCC on oxygen partial pressure is explainable by thermodynamics; the reaction of zircaloy with oxygen is preferred over that with iodine. The reaction kinetics are also very fast, as

can be demonstrated by opening to air an evacuated ampoule at 250°C containing a zircaloy specimen that has been reacted with iodine. Dense purple vapor fills the ampoule within seconds as a result of the rapid decomposition of ZrI_4 on the surface of the zircaloy by oxygen to release free iodine. These results suggest that in order to prevent reaction with iodine, it might be beneficial to provide a controlled oxygen source within a fuel rod in order to maintain a protective zirconium-oxide film on the inside surface of the cladding. Depending on the treatment it has received, a graphite coating might function as such an oxygen source. However, in order to meet fuel rod hydrogen specifications, measures are taken to keep the graphite coatings very dry; hence, they also contain little sorbed oxygen. A few additional experiments will be conducted to determine if oxygen purposely incorporated into a coating could retard iodine-SCC.

The results of these out-of-reactor PCI-simulator tests have shown that low-strain failures can be produced in coated susceptible zircaloy cladding when the graphite coating is outgassed and dry. On the other hand, instrumented ramp tests performed in the Halden Boiling Water Reactor have shown that graphite coating reduces hard pellet-to-cladding interaction. During ramp testing, the limit of average hoop strain resulting from pellet thermal expansion and cracking (after gap closure by pellet cracking) is estimated to be no more than ca. 1%. Under these limited strain conditions, the graphite coating reduces pellet-cladding mechanical interaction. However, it appears that dry coating does not afford iodine-SCC protection to susceptible cladding when tested under more severe sustained strain conditions such as those introduced by the PCI simulator.

2.2 SPHERE-PAC MICROSPHERE OUTGASSING STUDIES

During the fabrication of sphere-pac fuel for the FPIP, the UO_2 microspheres were vacuum outgassed prior to loading to reduce moisture/hydrogen concentrations. Supportive analytical studies are being performed to evaluate the sorption/desorption characteristics of the fuel particles.

For most of the analyses of UO_2 microsphere samples, a thermogravimetric analyzer was used to measure total weight loss at temperatures up to 1000°C,

and an electrolytic moisture analyzer was used to measure the moisture release up to 1000°C even though the normal the temperature limit of the P_2O_5 cell is about 500°C. The moisture analyzer uses a P_2O_5 electrolytic cell that changes conductivity in response to the amount of moisture present in a gas stream flowing over the sample.

Depleted UO_2 gel-derived microspheres, made approximately two years ago by ORNL, were used for these studies. The microspheres had been stored during the interim in sealed plastic bags contained in sealed cans, but the condition of their ambient atmosphere was otherwise unknown. Fine ($\sim 44 \mu m$ dia) microspheres were used for most of these tests in order to provide a detectable weight loss. While they constitute only 22% of the fuel weight in the normal three size-fraction mix, they contribute 75% of the surface area.

Sintered pellets are normally considered to be adequately dried by vacuum baking at temperatures of $\sim 600^\circ C$. Likewise, it was expected that $600^\circ C$ would be sufficient for drying the near-theoretical-density UO_2 microspheres because of their lack of surface-connected porosity. It was indicated that essentially all the moisture had been desorbed at about $500^\circ C$ (Figure 2.4); however, the thermogravimetric results indicated that the specimens were still apparently losing weight at temperatures to $1000^\circ C$. Work by other investigators⁽⁶⁾ suggests that temperatures as high as $1000^\circ C$ may be required to release all moisture from sintered UO_2 pellets. There was very little dependence of weight loss on isothermal hold time in any of these tests; holding at any specific temperature for one hour was sufficient to achieve the equilibrium state.

The weight lost at $500^\circ C$ was approximately half water (Figure 2.4). The balance was about 40% hydrogen, along with small amounts of N_2 , CO , and CO_2 ; at least these were the gases collected at $1000^\circ C$. It is possible that over the temperature range of 500 to $1000^\circ C$, any water released is being decomposed by the UO_2 and the hydrogen liberated is not detected as water. The hydrogen present in the $1000^\circ C$ offgas sample would add the

FIGURE 2.4. Water and Total Weight Loss of Fine UO_2 Microspheres at Temperatures to 1000°C

equivalent of about 63 ppm water, which is significant. Additional experiments are being performed to determine the time/temperature offgassing conditions required to achieve the total hydrogen specification limit of 1.5 ppm for the UO_2 microspheres.

For the moisture resorption studies, it was originally intended to dry the microspheres to a standard state (600°C for 4 hr in vacuum) and then measure resorption of moisture from a carrier gas such as helium. However, the medium (280-320 μm dia) and coarse (1150-1410 μm dia) sized microspheres did not absorb measurable amounts of moisture even when exposed to helium saturated with H_2O at 26°C.

Fine microspheres that were dried and then exposed to 26°C H_2O -saturated helium for 20 hours, showed moisture release of 58 ppm when vacuum-baked immediately after their exposure. In contrast, if dry helium was flowed over these microspheres for 20 hours before baking, their apparent weight returned to its original value and vacuum-baking showed no detectable moisture release.

This indicates that the microspheres can be exposed to air for short periods of time and can then possibly be redried by storing in a flowing dry atmosphere. Apparently formation of the strong H- and OH- bonds between UO_2 and H_2O is time-dependent.

3.0 TASK 2B -- IN-REACTOR EXPERIMENTS

J. O. Barner, M. E. Cunningham, R. J. Guenther, K. A. Hsieh, and M. S. Smith
(Pacific Northwest Laboratory)

R. K. Welty
(Exxon Nuclear Company)

3.1 SUMMARY - HBWR IRRADIATION AND RAMPING PLAN

Details of the general testing criteria, selection of the irradiation facilities, and the detailed descriptions of irradiations currently in progress were presented in previous progress reports.^(7,4)

The HBWR irradiation and ramping plan has been described previously⁽³⁾ and is summarized in Table 3.1. As indicated in the table, the steady-state irradiation of test rigs IFA-518.1 and IFA-517.1, as well as power-ramp testing of the initial eight rods, has been completed.

3.2 FABRICATION--HBWR SERIES H-2, H-3 AND H-4 RODS

The HBWR Series H-2, H-3 and H-4 characterization report, DOE/ET/34215-14 (PNL-3371), was completed.⁽⁸⁾ This report is a working document and provides as-built data describing the rods. The major topics in the report are:

- experiment description
- process development
- fabrication and characterization procedures and methods
- physical, chemical, and ceramographic characterization data.

3.3 HBWR STEADY-STATE IRRADIATION STATUS

The steady-state irradiation of IFA-518.2, which contains 12 instrumented test rods, was initiated in mid-March 1980 (Table 3.1). Approximate burnups in the rods as of the end of September 1980 are listed in Table 3.2. Operation to date has been satisfactory.

TABLE 3.1. HBWR Irradiation Plan

Rod No.	Rod Type (a)	Original Series Number	Charging Date	Rig (b)	Cluster (c)	Peak Goal Burnup, GJ/kgM (d)	Ramp Date
H1LCR002	R	H-1	7-78	518.1	LC	840 (9.7)	8-80
H1LCA006	A	H-1	7-78	518.1	LC	920 (10.7)	7-80
H1LCG009	AC	H-1	7-78	518.1	LC	940 (10.9)	5-80
H1LCG010	AC	H-1	7-78	518.1	LC	920 (10.6)	7-80
H1LCP013	VP	H-1	7-78	518.1	LC	850 (9.8)	5-80
H1LCP014	VP	H-1	7-78	518.1	LC	830 (9.6)	NP ^(e)
H1UCR003	R	H-1	7-78	518.1&.2	UC	1400 (16)*	8-81
H1LCR005	R	H-1	7-78	518.1	UC	680 (7.9)	8-81
H1LCA007	A	H-1	7-78	518.1	UC	760 (8.8)	8-81
H1LCG011	AC	H-1	7-78	518.1&.2	UC	1400 (16)*	8-81
H1UCP017	VP	H-1	7-78	518.1	UC	680 (7.9)	NP
H1UCP018	VP	H-1	7-78	518.1	UC	670 (7.8)	NP
H1LCR001	R	H-4	4-79	517.1	NA ^(f)	570 (6.6)	9-80
H2SPS040	SP	H-4	4-79	517.1	NA	580 (6.7)	9-80
H2XXS041	SP	H-4	4-79	517.1	NA	590 (6.8)	5-80
H2XXS042	SP	H-4	4-79	517.1	NA	580 (6.7)	NP
H2UCR023	R	H-2	2-80	518.2&.3	LC	1400 (16)*	11-82
H2UCR024	R	H-2	2-80	518.2&.3	LC	1400 (16)*	2-83
H2UCG027	ACP	H-2	2-80	518.2&.3	LC	1400 (16)*	11-82
H2UCG028	ACP	H-2	2-80	518.2&.3	LC	1400 (16)*	11-82
H2UCG029	ACP	H-2	2-80	518.2&.3	LC	1400 (16)*	11-82
H2UCS033	SP	H-2	2-80	518.2&.3	LC	1400 (16)*	2-83
H3UCR025	R	H-3	2-80	518.2&.3	UC	1400 (16)*	5-83
H3UCR026	R	H-3	7-81	518.3	UC	700 (8)*	5-83
H3UCG030	ACP	H-3	2-80	518.2&.3	UC	1400 (16)*	2-83
H3UCG031	ACP	H-3	2-80	518.2&.3	UC	1400 (16)*	2-83
H3UCG032	ACP	H-3	7-81	518.3	UC	700 (8)*	5-83
H3UCS034	SP	H-3	2-80	518.2&.3	UC	1400 (16)*	5-83

(a) R = Reference

A = Annular

AC = Annular-coated (Dag 154)

VP = Vipac

SP = Sphere-pac

ACP = Annular-Coated (Dag 4)-Pressurized.

(b) 518 = IFA-518 (2 six-rod clusters)

517 = IFA-517 (1 four-rod cluster).

(c) LC = Lower Cluster

UC = Upper Cluster

(d) Rod average burnup. Peak rod burnups are approximately 1.14 and 1.05 times the average values for IFA-518 and IFA-517, respectively. (* = projected value).

(e) NP = Not Planned.

(f) NA = Not Applicable.

TABLE 3.2. Approximate Burnups of IFA-518.2 Rods

Rod No.	Rod Type ^(a)	Burnup as of September 30, 1980			
		Rod Average, GJ/kgM (MWd/kgM)		Rod Peak, GJ/kgM (MWd/kgM)	
H2UCR023	R	420	(4.8)	480	(5.5)
H2UCR024	R	430	(5.0)	490	(5.6)
H2UCG027	ACP	470	(5.4)	540	(6.2)
H2UCG028	ACP	460	(5.3)	520	(5.9)
H2UCG029	ACP	460	(5.3)	520	(5.9)
H2UCS033	SP	420	(4.8)	480	(5.5)
H1UCR003	R	1120	(12.9)	1280	(14.7)
H1LCG011	AC	1120	(12.9)	1280	(14.7)
H3UCR025	R	420	(4.8)	480	(5.5)
H3UCG030	ACP	440	(5.1)	500	(5.7)
H3UCG031	ACP	460	(5.3)	520	(5.9)
H3UCS034	SP	420	(4.8)	480	(5.5)

(a) R = Reference
AC = Annular-coated (Dag 154)
SP = Sphere-pac
ACP = Annular-Coated (Dag 4)-Pressurized.

3.4 HBWR POWER-RAMP TESTING

The initial 8 power-ramp tests performed on rods irradiated in HBWR were completed. The idealized ramping scheme for the tests (Figure 3.1) consisted of a preconditioning period (a) of ≥ 24 hours at a linear heat generation rate (LHGR) of ~ 28 kW/m, which was slightly below the previous steady-state LHGR; an LHGR increase (b) at a rate of ~ 0.1 kW/m-min by increasing reactor power; a rapid (~ 3 kW/m-min) LHGR increase (c) by depressurizing the ^3He from coils surrounding the test rod; and a steady-state holding period (d) at or near the peak LHGR achieved at the end of the ^3He depressurization. Axial peak-to-average LHGR ratios were in the range 1.02 to 1.05 and the peak LHGR during the ramp tests occurred near or above the midplane of the ~ 0.46 m (18 in.)-long fuel stack. The design peak maximum LHGR of > 70 kW/m was achieved in most

FIGURE 3.1. Idealized Power Ramping Scheme

instances and, within reasonable limits, the specified test conditions were achieved during most of the tests. Exceptions were 1) an inadvertent reactor control rod drop immediately after the ^3He ramp during the testing of Rod H1LCG010 (AC10) resulting in a short-term decrease in the peak LHGR that was recovered within 10 min, and 2) achieving a peak maximum value of ~ 78.5 kW/m in Rod H1LCR002 (R2) versus the design value of 73 kW/m.

During the non-critical portion of the ramp tests, i.e., during the pre-conditioning period (a) and the last 7 hours of the peak power hold period (d), the CALIB system was used for data acquisition at intervals of 3 to 5 min. During the critical portions of the tests, i.e., during the reactor ramp (b), the ^3He ramp (c), and the first hour of the peak power hold (d), the FAST-SCAN system was used for data acquisition with an interval of 2 sec. Data recorded on the CALIB system consisted of time, reactor power, coolant temperature, vanadium neutron detector signals, cobalt neutron detector signal,

failure detector signal, rig differential coolant temperature, ^3He pressure, and the rod elongation (or pressure) signal. The same data were recorded on the FAST-SCAN system except that the vanadium neutron detector signals were not recorded because the response of the detectors is slow when compared to the ^3He -induced ramp rate. The data from the two systems will now be combined into a single file.

A preliminary summary of the ramping conditions for each rod is listed in Table 3.3. Final values will be determined after the data sets have been combined and qualified. None of the rods failed as a result of the ramp tests. A discussion of the preliminary results, based upon the FAST-SCAN data, is presented in Section 5.1.

3.5 HBWR STEADY-STATE DATA ACQUISITION AND PROCESSING

A cursory evaluation of rod elongation behavior for IFA-518.2 suggested that some pellet-cladding mechanical interaction (PCMI) occurred in the fresh pellet rods throughout the first power ascension (points a to f in Figure 3.2), as evidenced by the general rate of elongation of approximately 0.0046 mm (kW/m). This is about twice the elongation rate expected for free thermal expansion of the cladding which would normally be interpreted as being indicative of no PCMI. During the subsequent IFA-518.2 shutdown (points j to k) that occurred several days later and the second power ascension (points l to n), the rate of cladding elongation was representative of free thermal expansion. There was evidence of a small amount of PCMI in the IFA-518.2 lower cluster pellets rods during operation at the maximum LHGR condition, as evidenced by a cyclic elongation signal in phase with the normal cyclic variation in reactor power of $\sim \pm 3\%$. However, there was no indication of strong PCMI because cladding elongation did not increase when the peak LHGR that was previously achieved in the pellet rods (points m to n in Figure 3.2 and similar regions in Figures 3.3 and 3.4) was exceeded. Therefore, it is concluded that significant PCMI did not occur in the pellet rods during the first power ascension of IFA-518.2.

Two factors not associated with PCMI were determined to have affected cladding elongation during the initial startup of IFA-518.2, i.e., changes in

TABLE 3.3. Preliminary Summary of Power-Ramping Condition

<u>Rod Number</u>	<u>Average Burnup, GJ/kg (MWd/kgM)</u>		<u>Peak Ramp LHGR (a), kW/m</u>	<u>Peak Differential LHGR, kW/m</u>	<u>Ramp Rate, kW/m-min</u>	<u>Peak Reactor Power, MW</u>
H1LCG009 (AC9)	940	(10.9)	71	43	3.2	15.0
H1LCG010 (AC10)	920	(10.6)	73	45	~3.5	15.2
H1LCR002 (R2)	840	(9.7)	78	51	~3.5	15.5
H2XXS041 (S41)	590	(6.8)	72	44	3.7	14.5
H2SPS040 (S40)	580	(6.7)	75	47	~3.5	~14.5
H1LCR001 (R1)	570	(6.6)	72	44	~3.5	15.0
H1LCA006 (A6)	930	(10.7)	68	40	~3.5	~14.5
H1LCP013 (V13)	850	(9.8)	71	43	3.4	14.0

(a) Preliminary number based upon estimates.

coolant temperature and a "drift" of cladding elongation that occurs during the initial startup of fresh rods. During the period from points a to h (Figure 3.2), subcooled water was being injected into the test section and, in general, the amount of subcooling increased with increasing LHGR. From points c to e and f to h, where heat balance calibrations were being performed, subcooled flow was varied over a wide range. During the calibrations, the increased amount of subcooling caused an apparent elongation of the test rods. (Note: This is opposite of what would normally be expected, i.e., an increase in subcooling while maintaining a constant outlet temperature would be expected to result in a decrease in the average cladding temperature and hence, a contraction. However, what is actually being measured is the differential elongation between the test rod and a stay rod upon which the instrumentation is mounted. The apparent cladding elongation is caused by less contraction of the stay rod compared with the fuel rod as a function of temperature.) Using data obtained during the two calibration periods for the condition of decreasing subcooling and nearly constant LHGR, a subcooling factor of $0.0035 \text{ mm}/^{\circ}\text{C}$ was determined for Rod ACP27. Normalizing the data

FIGURE 3.2. Rod Elongation During Initial Irradiation of Lower Cluster Rod ACP27 in IFA-518.2

according to the subcooling factor, significantly reduced the rate of cladding elongation during the initial startup of IFA-518.2 (Figure 3.5).

A "drift" in the cladding elongation signal with time or burnup at a constant LHGR occurs during the initial irradiation of fresh rods. This "drift" in elongation signal for the lower cluster ACP rods in IFA-518.2, is illustrated in Figure 3.6 between points o and p and is also represented by the difference in elongation between points i and j in Figure 3.2. This "drift" only occurs in fresh fuel rods and the rate of "drift" decreases with time/burnup. Figure 3.7 shows a similar "drift" in fresh upper cluster rods and Figure 3.8 illustrates the absence of a "drift" in previously-irradiated

FIGURE 3.3. Rod Elongation During Initial Irradiation of Lower Cluster Rod ACP29 in IFA-518.2

rods during the same period. A similar drift also occurred during the initial irradiation of all IFA 518.1 rods. The cause of the "drift" is not currently known but may be related to the relaxation of internal fabrication stresses/strains in the cladding.

The elongation versus LHGR curve for data between points o and p in Figure 3.6 (points i to j in Figure 3.2) was fit to a third order polynomial and the equation was extrapolated back to zero time. By combining the effects of subcooling and the "drift", the apparent anomalies in the cladding elongation behavior of the IFA-518.2 lower cluster rods was eliminated (Figure 3.9).

FIGURE 3.4. Rod Elongation During Initial Irradiation of Lower Cluster Rod R23 in IFA-518.2

Use of the subcooling correction and the "drift" correlation will permit meaningful comparisons of elongation data obtained from different rods in the same, as well as, other test rigs.

3.6 PRIMARY RAMPING TEST STATUS

Central segments from the segmented rods in BRPR Series S-1 and S-2 will be utilized for the primary ramp tests after completion of the steady-state irradiations. Forty-eight segmented rods are currently undergoing irradiation in BRPR. Status of the BRPR irradiations is described in Section 4.0. Initial primary power-ramp tests on BRPR segments are tentatively planned for April 1982.

FIGURE 3.5. Rod Elongation in Rod ACP27 Corrected for the Effect of Subcooling

FIGURE 3.6. Rod Elongation in IFA-518.2 Lower Cluster Annular-Coated-Pressurized Rods ACP27 and ACP29. Note "drift" between points o and p.

FIGURE 3.7. Rod Elongation in IFA-518.2 Upper Cluster Rods
Illustrating "Drift" in Fresh Rods

FIGURE 3.8. Elongation of Previously Irradiated Rods in IFA-518.2. Note lack of "drift."

FIGURE 3.9. Rod Elongation in Rod ACP27 from IFA-518.2 after Correction for Subcooling and "Drift" during the Initial Power Ascension

4.0 TASK 2C -- IN-REACTOR DEMONSTRATIONS

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

A total of 367 rods were fabricated for the Big Rock Point Reactor demonstration irradiations and are divided into four series, as summarized in Tables 4.1 through 4.5. The BRPR Series S-1 rods were placed into the BRPR core in the spring of 1979, 4 assemblies containing segmented Series S-1 and S-2 rods were loaded into BRPR in the fall of 1979, and the remaining Series S-1, S-2, and S-4 rods were shipped to BRPR for loading in the fall of 1980. The total of 9 assemblies in the BRPR core contain all the segmented rods in the demonstration series. The rod loading maps for the 2 Series S-4 assemblies (H1-20 and H1-21) are shown in Figures 4.1 and 4.2. The Series S-4 segmented rods are similar to the Series S-2 segmented sphere-pac rods except that the bottom segments of the S-4 rods are also pressurized. This was made possible by a modification to the high pressure welding chamber which permitted pressurization of the bottom segment. Table 4.6 lists the design parameters of the S-4 sphere-pac rods. Tables 4.7 and 4.8 list the serial number of each rod and a description of the rodlets.

The remaining 108 full-length rods (Series S-3) will be assembled in BRPR reload batch H-2 in mid-1981.

Fuel rod pre-characterization data for HBWR Series H-2 through H-4 (Halden) have been assembled into the EPRI Fuel Performance Data Base format. Series H-1 data are scheduled to be entered in November 1980. Data files are being assembled for the BRPR Series S-1, S-2, and S-3 fuel rods. Due to the number of rods involved (367), this task will continue into early 1981.

Work continued on the topical report which describes the sphere-pac process development and fabrication and the characterization report for the BRPR Series S-1, S-2, and S-3 rods.

Table 4.9 summarizes the status, as of the end of the last cycle (Cycle 16) of the BRPR Series S-1, S-2, S-3, and S-4 fuel assemblies that are being or will be irradiated in the Big Rock Point Reactor.

TABLE 4.1. Big Rock Point Reactor Series S-1 Matrix
(4 Assemblies: G-419, G-420, G-421, and G-422)

<u>Fuel Form</u>	<u>Parameters</u>		<u>Full-Length Rods</u>	<u>Segmented Rods^(a)</u>	
	<u>Coated</u>	<u>Pressurized</u>			
Vipac	--	Yes	15	4	
	--	No	--	--	(4)
Annular	Yes	Yes	24	4	(4)
	Yes	No	20	4	(4)
	No	No	20	4	(4)
Reference	Yes	No	--	--	(4)
	No	No	<u>40^(b)</u>	<u>8</u>	<u>(4)</u>
			119	24	(24)

- (a) Number of segmented rods (each consisting of three rodlets) and the number of middle segment rodlets for ramp testing. Numbers in () are lower segment rodlets and are considered to be spares. Top segment rodlets not listed all contain reference pellets and primarily function as non-test fuel.
- (b) Sixteen "product-line", i.e., nonchamfered pellet rods, received equivalent characterization. All other pellets are chamfered.

TABLE 4.2. BRPR Series S-2 Matrix (3 Assemblies:
H-101, H-102, and H-103)

<u>Fuel Form</u>	<u>Parameters</u>		<u>Full-Length Rods</u>	<u>Segmented Rods^(a)</u>	
	<u>Coated</u>	<u>Pressurized</u>			
Annular	Yes	Yes	56	8	--
Sphere-Pac	--	Yes	14	8	--
Reference	No	No	<u>14</u>	<u>8</u>	<u>(24)</u>
			84	24	(24)

- (a) Number of segmented rods (each consisting of three rodlets) and the number of middle segment rodlets for ramp testing. Numbers in () are lower segment rodlets and are considered to be spares. Top segment rodlets all contain reference pellets and primarily function as spacers.

TABLE 4.3. BRPR Series S-3 Matrix (3 Assemblies:
As yet undesignated)

<u>Fuel Form</u>	<u>Parameters</u>		<u>Full-Length Rods</u>	<u>Segmented Rods</u>
	<u>Coated</u>	<u>Pressurized</u>		
Annular	Yes	Yes	66	--
Sphere-Pac	--	Yes	24	--
Reference	No	No	<u>18</u>	<u>--</u>
			108	--

TABLE 4.4. BRPR Series S-4 Matrix (2 Assemblies: H-120, H-121)

<u>Fuel Form</u>	<u>Parameters</u>		<u>Full-Length Rods</u>	<u>Segmented Rods</u>
	<u>Coated</u>	<u>Pressurized</u>		
Sphere-Pac	--	Yes	--	8 (8) ^(a)

(a) Lower segments in parenthesis and in this case, the middle and lower segments are both helium pressurized.

TABLE 4.5. Summary of Demonstration Irradiations
for the Big Rock Point Reactor

<u>Fuel Form</u>	<u>Parameters</u>		<u>Full-Length Rods</u>	<u>Segmented Rods^(a)</u>	
	<u>Coated</u>	<u>Pressurized</u>			
Annular	Yes	Yes	146	12	--
	Yes	No	20	4	(8)
	No	No	20	4	(4)
Sphere-Pac	--	Yes	38	16	(8)
Vipac	--	Yes	15	4	--
	--	No	--	--	(4)
Reference	Yes	No	--	--	(4)
	No	No	<u>72^(b)</u>	<u>16</u>	<u>(28)</u>
			311	56	(56)

(a) Lower segment rodlets in parenthesis.

(b) Includes 16 nonchamfered "product-line" rods that received equivalent characterization. All other pellets are chamfered.

TABLE 4.6. Design Parameters for BRPR Series S-4 Sphere-Pac Rods

<u>Fuel</u>	
Type	Sphere-Pac
Material	UO ₂ Microspheres
Density (microspheres); %TD	>96.5 (target >98% TD)
Enrichment, wt% ²³⁵ U	2.52
Smear Density, %TD	87.0
<u>Insulator Disks (each segment)</u>	
Material	Alumina
Density, g/cm ³	3.89
Diameter, mm (in.)	9.915 (0.362)
Length, mm (in.)	5.1 (0.200)
Number, location	2, one at each end of fuel column
<u>Plenum Spring (upper segment)</u>	
Material	Inconel X-750
Spring force, (nominal), N (lb)	49 (11)
Spring constant, (nominal), N/m (lb/in)	2100 (12)
Spring volume, mm ³ (in ³)	628 (0.0383)
<u>Plenum Spring (mid and lower segment)</u>	
Material	Inconel X-750
Wire diameter, mm (in.)	0.805 (0.0317)
Coil O.D. (at working length), mm (in.)	8.64 ± 0.38 (0.340 ± 0.015)
Free length, mm (in.)	56.90 ± 0.76 (2.240 ± 0.030)
Spring force, N (lb)	9.34 (2.10)
Spring constant, N/m (lb/in.)	394 (2.23)
Spring volume, mm (in ³)	304 (0.0185)
<u>Fill Gas</u>	
Material	He
Purity, %	99.9
Pressurization, MPa (psi)	0.462 ± 0.048 (67 ± 7)

Fuel Retainer

Material	Zircaloy-2 or -4
O.D., mm (in.)	9.703 ± 0.0000 (0.3820 ± 0.0000) 0.0127 0.0005
I.D., mm (in.)	5.563 ± 0.127 (0.219 ± 0.005)
Thickness, mm (in.)	0.102 ± 0.0254 (0.004 ± 0.001)
Number per sphere-pac rod	2

Cladding

Material	Zircaloy-2
Condition	Cold Worked-Stress Relieved
O.D., before etching, mm (in.)	11.455 ± 0.038 (0.451 ± 0.0015) ^(a)
I.D., mm (in.)	9.677 ± 0.038 (0.381 ± 0.0015)
Wall thickness, minimum, mm (in.)	0.831 (0.0327)
Length (upper segment), mm (in.)	666.24 ± 0.13 (26.230 ± 0.005)
Length (mid and lower segment), mm (in.)	600.18 ± 0.13 (23.629 ± 0.0005)
Allowable bow, mm/m (in./ft)	1.25 (0.015)
Coating (present in some rods with annular fuel)	Graphite
Coating layer thickness, mm (in.)	0.0064 ± 0.0038 (0.00025 ± 0.00015)

Fuel Rod

Number of segments	3
O.D., mm (in.)	11.405 ± 0.038 (0.449 ± 0.0015)
Allowable bow, mm/m (in./ft)	1.25 (0.015)
Plenum length (upper plenum), mm (in.)	99.09 (3.901)
Plenum length (mid and lower plenum), mm (in.)	32.82 (1.292)
Fuel column length (upper segment), mm (in.)	544 ± 4.06 (21.435 ± 0.16)
Fuel column length (middle segment), mm (in.)	546.84 ± 2.03 (21.529 ± 0.08)
Fuel column length (lower segment), mm (in.)	544.45 ± 2.03 (21.435 ± 0.08)
Axial distance between fuel columns, mm (in.)	71.12 (2.800)
Fuel configuration (upper segment)	Reference
Fuel configuration (middle and lower segments)	Sphere-pac

(a) Cladding selected to fall within a range of ± 0.025 mm (± 0.001 in.).

TABLE 4.7. BRPR Series S-4 Rods in Assembly H1-20

<u>Fuel Rod Location</u>	<u>Rod Type</u>	<u>Serial No.</u>
1F	Segmented; sphere-pac (pressurized) middle & lower segments	LU310016
		LU320016
		LU330016
6A	Segmented; sphere-pac (pressurized) middle & lower segments	LU310018
		LU320018
		LU330018
6K	Segmented; sphere-pac (pressurized) middle & lower segments	LU310015
		LU320015
		LU330015
11F	Segmented; sphere-pac (pressurized) middle & lower segments	LU310017
		LU320017
		LU330017

TABLE 4.8. BRPR Series S-4 Rods in Assembly H1-21

<u>Fuel Rod Location</u>	<u>Rod Type</u>	<u>Serial No.</u>
1F	Segmented; sphere-pac (pressurized) middle & lower segments	LU310020
		LU320020
		LU330020
6A	Segmented; sphere-pac (pressurized) middle & lower segments	LU310019
		LU320019
		LU330025(a)
6K	Segmented; sphere-pac (pressurized) middle & lower segments	LU310026
		LU320026
		LU330026
11F	Segmented; sphere-pac (pressurized) middle & lower segments	LU310025
		LU320025
		LU330019(a)

(a) Serial number out of sequence in lower segment.

TABLE 4.9. Summary Status of Demonstration Assemblies for BRPR

<u>Series</u>	<u>Assembly Number</u>	<u>Assembly Type</u>	<u>Assy. Avg. Burnup, MWd/kgM</u>	<u>Avg. Rod LHGR, kW/m(kW/ft)</u>	<u>Max. Rod LHGR, kW/m(kW/ft)</u>
S-1	G419	Full-length rods	~ 0	-	-
	G420	Full-length rods	~ 0	-	-
	G421	Full-length and segmented rods	7.3	14.1 (4.3)	20.7 (6.3)
	G422	Full-length and segmented rods	7.4	14.1 (4.3)	20.7 (6.3)
S-2	H101	Full-length rods	0	-	-
	H102	Full-length and segmented rods	7.5	14.3 (4.4)	21.3 (6.5)
	H103	Full-length and segmented rods	7.5	14.4 (4.4)	21.3 (6.5)
S-3 (Full-length rods to be loaded in December 1981; Assemblies No. H2XX, etc.)					
S-4	H120	Segmented sphere-pac rods	0	-	-
	H121	Segmented sphere-pac rods	0	-	-

(Serial Nos. are for lower rodlets)

FIGURE 4.1. BRPR Series S-4 Fuel Assembly (H1-20) Map

(Serial Nos. are for lower rodlets)

FIGURE 4.2. BRPR Series S-4 Fuel Assembly (H1-21) Map

Blank

5.0 TASK 3 -- PERFORMANCE EVALUATION

J. O. Barner, M. E. Cunningham, R. J. Guenther,
K. A. Hsieh, and M. S. Smith
(Pacific Northwest Laboratory)

5.1 SPHERE-PAC FUEL CENTERLINE TEMPERATURE MEASUREMENTS

As previously reported,⁽³⁾ during the irradiation of the HBWR Series H-4 rods in the Halden Boiling Water Reactor (HBWR) in IFA-517.1 to a burnup of 575 GJ/kgM (6.6 MWd/kgM), 1) the 2 sphere-pac rods with thermocouples operated at essentially the same central fuel temperature at equivalent linear heat generation rates (LHGRs), and 2) the sphere-pac central fuel temperatures were lower than the temperatures in the reference pellet rod by a constant differential at equivalent LHGRs as a function of burnup. Correction of two errors in the data reduction codes have modified these observations as follows:

1. At an LHGR of 36 kW/m, the central fuel temperatures in the 2 sphere-pac rods differed by about 40°C at the beginning of the irradiation (Figure 5.1). In addition, at 36 kW/m, the average of the central fuel temperatures at the beginning-of-life in the 2 sphere-pac rods was about ~110°C lower than the fuel temperature in the reference rod instead of ~125°C.
2. The difference in central fuel temperature between the sphere-pac and the reference pellet rods at equivalent LHGRs decreased as a function of burnup. This is probably a result of degradation of the fill-gas conductivity due to higher fission gas release from the sphere-pac fuel. At zero burnup and at 36 kW/m, the sphere-pac rods operated at a central fuel temperature about 110°C lower than the reference fuel, while at about 520 GJ/kgM (6 MWd/kgM), the difference was reduced to about 65°C (Figure 5.2).

5.2 LHGR ONSET FOR PCMI

The onset for pellet-cladding mechanical interaction (PCMI), i.e., the LHGR where rod elongation deviates from the elongation due to axial thermal expansion of the cladding, was determined for 3 types of rods during power

FIGURE 5.1. Corrected Measured Fuel Centerline Temperature During Initial Startup in IFA-517.1 Sphere-Pac and Reference Rods

FIGURE 5.2. Corrected Measured Fuel Centerline Temperatures at 36 kW/m as a Function of Burnup in IFA-517.1 Sphere-Pac and Reference Rods

ascensions at different burnup levels. The rods were reference Rod R2, annular Rod A6, and annular-coated Rod AC9, all from the lower cluster of IFA-518.1. The method of analysis involved determining the point of inflection in the cladding elongation versus LHGR curves, as illustrated in Figure 5.3, during power ascensions up to rod average LHGRs of 35 to 37 kW/m at different burnup levels. Both the reference (R2) and annular (A6) pellet rods showed evidence of PCMI (Figure 5.4) starting at a burnup of about 260 GJ/kgM (3 MWd/kgM), while the annular-coated Rod AC9 did not indicate any PCMI until a burnup of about 690 GJ/kgM (7.5 MWd/kgM) had been achieved. The reference rod appears to have a greater dependence of PCMI onset on burnup than the annular rod while over the small burnup range that PCMI was monitored for the annular-coated rod, no burnup dependence was indicated. The higher burnup threshold and the

FIGURE 5.3. The Onset of PCMI During Reactor Startups

FIGURE 5.4. LHGR Required for the Onset of PCMI in Reference, Annular, and Annular-Coated Rods

absence of a burnup dependence for the onset of PCMI in the annular-coated rod, when compared to the reference and annular rods, illustrates the beneficial effects of the graphite coating to reduce cladding strain.

5.3 POWER-RAMP RESULTS

As mentioned in Section 3.4, none of the 8 HBWR Series H-1/H-4 rods that were power-ramped to the conditions listed in Table 3.3 failed as a result of the test. Actually, cladding failure was considered unlikely at these relatively low burnup levels achieved in the HBWR. Because none of the rods failed, a performance comparison for the different fuel types based on a fail/no-fail criterion was not possible. However, based on rod elongation behavior during the ramps, it was indicated that the annular-coated and packed-particle

designs (both sphere-pac and vipac) exhibited superior ramping performance when compared to the reference design. The bases for these observations and preliminary conclusions are described below.

As noted in Section 3.4, FAST-SCAN data for only a portion of the first 6 ramp tests has been processed at this time. Five of the rods were equipped with elongation sensors while the 6th rod incorporated an internal pressure transducer. The FAST-SCAN data covers only that portion of the test involving the reactor ramp starting at an LHGR of 28 kW/m, the ^3He -induced portion of the ramp, and about 1 hour of the subsequent holding period at the maximum LHGR.

The 5 rods equipped with elongation sensors for which FAST-SCAN data are currently available are the 2 reference rods (R1 and R2), 1 annular-coated rod (AC9), 1 sphere-pac rod (S41), and 1 vipac rod (V13). The 5 rods were power-ramp tested according to the scheme shown in Figure 3.1. Cladding elongation behavior during the ramp tests as a function of LHGR for the HBWR Series H-1 rods is shown in Figure 5.5. Similar data for the Series H-4 rods are shown in Figure 5.6. Behavior comparisons between the different fuel types are based upon the ramp-induced, axial mechanical strain in the cladding, i.e., the total measured axial cladding strain less the axial thermal expansion of the cladding. The axial expansion data during the non-PCMI-portion of the tests, i.e., at low power, are required to accurately deduce the thermal expansion component of the measured axial strain and are not currently available. However, the thermal expansion component is expected to be equivalent to the slope shown for Rod AC9 between ~30 kW/m and ~50 kW/m (Figure 5.5), i.e., 0.0035 mm/(kW/m) and the following discussion is based on the assumption that the thermal expansion component is equivalent to this value.

The elongation behavior of the 2 reference rods was similar (Figures 5.5 and 5.6). During the reactor portion of the ramps up to ~34 kW/m and during the initial portion of the ^3He -induced ramps, both rods elongated at a rate approximately 3.5 times the free thermal expansion of the cladding. This is interpreted as being indicative of a moderate amount of PCMI. Subsequently, both rods exhibited a nearly instantaneous change to hard PCMI at LHGR levels of 44 kW/m for Rod R2 and 54 kW/m for Rod R1. The difference in LHGR for the

FIGURE 5.5. Rod Elongations in Reference Rod R2, Annular-Coated Rod AC9, and Vipac Rod V13 from HBWR Series H-1 During Ramp Testing (Base irradiation was in IFA-518.1.)

FIGURE 5.6. Cladding Elongations in Reference Rod R1 and Sphere-pac Rod S41 from HBWR Series H-4 During Ramp Testing (Base irradiation was in IFA-517.1.)

onset of hard PCMI in the 2 rods probably reflects differences in their burnup levels. Rod R2 had accumulated a burnup of ~840 GJ/kgM (9.7 MWd/kgM) while Rod R1 had a burnup of ~570 GJ/kgM (6.6 MWd/kgM). In addition, the peak LHGR and peak burnup in Rod R2 occurred at the top of the fuel stack while in Rod R1, the peak conditions occurred near the midplane of the fuel stack. This condition would tend to promote more fuel cracking, fuel fragment misalignment, and fuel cladding mechanical interaction in the top of Rod R2 than in the top of Rod R1. As a result, the gauge length over which the interaction was operative would be greater in Rod R2 than Rod R1 and the slope of the elongation versus LHGR curve, or the maximum elongation, would be greater in Rod R2 than in Rod R1 as is observed. An apparent abrupt slippage between the fuel column and the cladding occurred in Rod R2 at an LHGR of ~49 kW/m. Both reference rods exhibited cladding contraction during the latter stages of the power ramp and during the first hour of the hold period at the maximum LHGR. This contraction is probably associated with creep in the hotter portions of the fuel to relieve the interaction stresses between the fuel and cladding. Neglecting the axial thermal expansion component, both reference rods exhibited about the same maximum cladding elongation (strain), which was the largest amount that occurred in any of the 5 rods evaluated.

The elongation behavior of annular-coated Rod AC9 was characterized by free thermal expansion of the cladding up to an LHGR of ~53 kW/m, a gradual onset of PCMI between 53 and 63 kW/m, hard interaction above 63 kW/m, and rod contraction during the steady-state hold period at the maximum LHGR. The delay in the onset of PCMI and the gradual onset of hard PCMI is attributed to the beneficial effects of the annular pellets and/or the graphite coating. When hard interaction occurred, it was similar to that which occurred in the reference rods. Cladding contraction that occurred at the peak LHGR is attributed to fuel deformation and elastic strain recovery in the cladding. A direct behavior comparison can be made between Rod AC9 and reference Rod R2 because these rods had the same irradiation history, the same axial power distribution during the pre-ramping irradiation, and similar burnup levels.

During the reactor ramp and the first part of the ³He-induced ramps, the behavior of sphere-pac Rod S41 was characterized by cladding elongation at a

rate 2 to 3.5 times the rate for free axial thermal expansion. This is interpreted as being indicative of a moderate amount of fuel-cladding mechanical interaction and is about the same as the most severe cladding elongation that occurred during the initial startup with fresh sphere-pac fuel rods. At about 58 kW/m, the axial elongation decreased as evidenced by the inversion in the slope of the cladding elongation versus LHGR curve. Axial contraction of the rod during the latter part of the power ramp and during the peak power hold period may be indicative of fuel deformation and the relief of fuel and cladding stresses. When the axial thermal expansion of the cladding is accounted for, it appears that there is essentially zero residual axial cladding strain in the sphere-pac rod after 1 hour at the peak LHGR.

The elongation behavior of the vipac fuel rod, Rod V13, was very similar to that of the sphere-pac fuel rod. The initial interaction, which is similar to the moderate interaction that occurred in the reference rods, was greater than for the sphere-pac rod. Perhaps the initially greater interaction compared to the sphere-pac rod, can be attributed to the angular nature of the vipac fuel fragments and their interaction with the cladding. The lower LHGR for the onset of cladding contraction for the vipac rod compared with the sphere-pac rod (~48 kW/m versus ~58 kW/m) may be associated with the lower effective thermal conductivity and higher fuel temperatures of vipac fuel when compared to the pressurized sphere-pac fuel. The higher burnup and expected higher fission gas release in the vipac rod compared with the sphere-pac rod could also be factors. Again, when factoring out the axial thermal expansion of the cladding, it appears that there is essentially zero residual axial cladding strain in the vipac fuel rod after one hour at the peak power.

Based upon the above observations from the power-ramp tests, the following preliminary conclusions are made:

1. The PCMI behavior of the annular-coated fuel rod is superior to the reference fuel rods on the basis that the annular-coated Rod AC9 exhibited, (a) less maximum axial cladding strain, (b) a higher LHGR for the onset of PCMI, and (c) a gradual onset of hard PCMI.

2. The PCMI behavior of the sphere-pac fuel rod is superior to the reference pellet rods on the basis that the sphere-pac Rod S41 exhibited, (a) less maximum axial cladding strain, (b) a lower strain rate during moderate PCMI, and (c) essentially zero residual cladding strain after the ramp.
3. The PCMI behavior of the vipac fuel rod is superior to reference pellet rods on the basis that the vipac Rod V13 exhibited, (a) less maximum axial strain, and (b) essentially zero residual strain after the ramp.

5.4 THERMOCOUPLE DECALIBRATION

The W-3% Re/W-25% Re fuel centerline thermocouples used in the test rods in IFA-517.1 and IFA-518.1 were subject to decalibration resulting from transmutation of the thermocouple constituents, primarily tungsten to rhenium, in the neutron flux. The decalibration effect results in indicated temperatures that are lower than the actual temperatures. Obviously, qualitative comparisons between the temperature measurements in the rods can be made using the as-measured signals. However, the technical bases in support of licensing of the fuel concepts require best-estimate temperatures.

The Halden Project has estimated the decalibration effect based on theoretical considerations and has attempted recalibration of irradiated thermocouples.⁽⁹⁾ Based upon theoretical considerations, a decalibration factor in the range from 2.5 to 4.0% per 10^{24} n/m² was estimated. Recalibration data on reliable thermocouples indicates that the decalibration factor is in the range from 1.0 to 3.0% per 10^{24} n/m² and may be dependent upon the operating temperature and/or recalibration temperature. The Halden Project recommends a value of 1.7% per 10^{24} n/m².

Another estimate for the thermocouple decalibration factor was determined from transient data taken during a reactor scram as part of the Experimental Support and Development of Single-Rod Fuel Codes Program (NRC).⁽¹⁰⁾ This decalibration measurement was based on data from IFA-432 rods irradiated to a

burnup of approximately 1900 GJ/kgM (22 MWd/kgM) and amounted to a total decalibration of ~20%. On this basis, the estimated decalibration factor was 1.75% per 10^{24} n/m² at the thermocouple tip.

These data show considerable scatter and the operating temperature, recalibration temperature and fluence dependencies are not accurately known. However, thermocouple decalibration can be expected to have a 3 to 8% effect on the measured temperatures which is considered to be significant for the HBWR experiments. The value of 1.75% per 10^{24} n/m² appears reasonable and will be used. For data published in topical reports as part of the evaluation of the HBWR data for FPIP, both the measured temperatures and the "corrected" temperatures will be reported. This will permit additional corrections to be made if subsequent determinations of the decalibration factor improve the data.

6.0 TASK 4 -- ESTABLISHMENT OF TECHNICAL BASES

K. R. Merckx
(Exxon Nuclear Company)

6.1 TECHNICAL BASES

The general scope of the technical bases reports being developed for the FPIP fuel candidates has been expanded from design oriented technical bases to include general performance and licensing considerations. The general format of: 1) criteria, 2) methodology, and 3) data and theoretical models has been retained. Comment drafts of graphite coated cladding and annular fuel pellets technical bases reports have been completed. The new versions of these reports are structured so that FPIP test results in terms of fuel rod performance and the results from specialized tests are factored into the methodology and data-theory sections.

The general safety issues have been reviewed and comments have been factored into the criteria and methodology sections of the reports. Detailed analyses for the annular-coated-pressurized fuel design are in progress.

Use of some of the FPIP-developed data to support the criteria and methodology requires further expansion. For example, a more detailed gap conductance model is needed to explain the measured temperature drop across the graphite coating. Also, data needed to evaluate the deformation mechanism model for annular pellet fuel is being collected and archived on the EPRI Fuel Performance Data Base System.

6.2 DATA AND REFERENCE FILE DEVELOPMENT

As the means for accumulating, archiving, transmitting, and manipulating the fuel performance data and technical bases references collected during the FPIP, computer data base filing systems are being used. The contents of the reference files have been updated to include the new references that have been used in developing the technical bases. Only minor changes in the search and editing procedures were implemented to make the system more compatible with the PRIME computer interactive data handling procedures.

The EPRI Fuel Performance Data Base System has been adopted for the fuel characterization and performance data accumulation and archiving. The actual software for this system was just recently received and implemented on the ENC PRIME computer and rod characterization data from the Halden characterization reports has been entered into this system. No data base system limitations have been encountered, and a working knowledge of this system is being developed.

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APPENDIX A

CONVERSION TABLE FOR SI UNITS

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 heavy metal ^(c) (MWd/MTM)	gigajoule/kg of heavy metal ^(c) (GJ/kgM)	8.640×10^{-2}

(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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