Fuel Performance Improvement Program

Quarterly Progress Report
October - December, 1977

Consumers
Power
Company

EXXON NUCLEAR
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FUEL PERFORMANCE IMPROVEMENT PROGRAM

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October - December, 1977**

Compiled by C. E. Crouthamel

**Exxon Nuclear Company, Inc.
Richland, Washington 99352**

January 1978

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This report has been prepared for the U.S. Department of Energy under Contract No. EY-76-C-02-4066. Battelle, Pacific Northwest Laboratories work has been performed under Contract No. EY-76-C-06-1830.

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ABSTRACT

The Fuel Performance Improvement Program has as its objective the identification and demonstration of fuel concepts with improved power ramp performance. Improved fuels are being sought to allow reduction or elimination of fuel related operating guidelines on nuclear power plants such that the fuel may be power maneuvered within the rates allowed by the system technical specifications. The program contains a combination of out-of-reactor studies, in-reactor experiments and in-reactor demonstrations. Fuel concepts initially being considered include annular pellets, cladding internally coated with graphite and packed-particle fuels. The performance capability of each concept will be compared to a reference fuel of contemporary pellet design by irradiations in the Halden Boiling Water Reactor. Fuel design and process development is being completed and fuel rod fabrication will begin for the Halden test rods and for the first series of in-reactor experiments. The in-reactor demonstrations are being performed in the Big Rock Point reactor to show that the concepts pose no undue risk to commercial operation. Additional concepts may be considered as the result of a state-of-the-technology review of fuel-cladding interaction and assessment of fuel concepts and the out-of-reactor studies. The results of the program will be used to establish the technical bases for design of fuels with improved power ramp performance.

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1.0 INTRODUCTION

C. E. Crouthamel (Exxon Nuclear Company)

1.1 Program Objectives

The objective of the Fuel Performance Improvement Program (FPIP) is to identify and demonstrate light water reactor (LWR) fuel concepts which have the potential for improved power ramp performance.

These objectives are being pursued through a combination of out-of-reactor studies, in-reactor experiments, and in-reactor demonstrations. The program is seeking to establish the technical bases for design of improved fuel concepts which can be operated consistent with the maneuvering capability of nuclear steam supply systems.

1.2 Background

Experience has shown that fuel performance in light water reactors can be adversely affected by a rapid rate of power increase from partial to full power. This stems from the vulnerability of the fuel to fail as a result of interactions between the fuel and cladding (fuel-cladding interaction or FCI). Concern about the adverse effects of power increases (ramps) on current fuel designs has resulted in operating guidelines for both PWR's and BWR's which limit the rate at which reactor power level may be increased during startup. Following an extended shutdown, for example, system startup may require the fuel to not exceed 50 percent of rated power unless the ascension is extended out over several days. Similarly, there are restraints on startup from various intermittent plant shutdowns caused by malfunctions in other portions of the plant or control manipulations in the nuclear steam supply system.

While restrictions on the rate of power increase may improve fuel reliability, they impose an operating constraint on the nuclear utility industry. Typical restrictions on the allowable rate of power escalation in the general range of 10 percent to 30 percent per day can cost a plant perhaps 2.5 full power days on each startup with a corresponding reduction

in the plant capacity factor. Clearly there is a need and incentive for the demonstration of nuclear fuels which will allow less restrictive operation of nuclear power reactors.

The FPIP, therefore, addresses the opportunity to test and demonstrate nuclear fuel concepts which would remove nuclear fuel as a limitation to the operation of nuclear power reactors. Such an advance in technology would provide options allowing the nuclear power plant operator to produce more net electrical power per cycle.

1.3 Program Description

The Fuel Performance Improvement Program is a remedy oriented fuels program for light water nuclear reactors and is under the sponsorship of the U.S. Department of Energy (DOE). It has as its three principal participants Consumers Power Company, Exxon Nuclear Company, and Battelle-Northwest. The program was initiated in the fall of 1976 and became fully authorized in the spring of 1977. A summary description is presented here to identify the basic precepts and elements of the program.

The program was conceived on the premise that alternate fuel concepts which have the potential for improved power ramp performance could be screened and identified in a reactor testing and a reactor demonstration program. Improved ramp performance is being sought to allow less restrictive operation of light water nuclear reactors such that systems limitations, rather than fuel limitations, control the rate of power increase. This is being pursued through tasks which involve a state-of-the-technology review and assessment of fuel concepts, out-of-reactor studies, in-reactor screening tests, in-reactor fuel performance demonstrations, and fuel performance evaluations for the ultimate purpose of establishing the technical bases for design of commercially acceptable improved fuel designs.

The above general statement of the program contains several important precepts, or guidelines, which deserve further elaboration:

- 1) The program approaches FCI as a generic concern.

Fuel failure by power ramping is recognized to be a complex thermal-mechanical-chemical mechanism which is not fully understood. The mechanisms, within the context of average rates of core power increase, are also viewed as relevant to both PWR's and BWR's. While there are fuel design and system operational differences between these reactor types, the operative failure mechanisms and possible fuel design remedies are similar. The program approach also recognizes the use of up-to-date technology in modelling performance or predicting fuel behavior; however, the program is designed to ultimately answer the question of performance by irradiation tests and demonstrations rather than by intricate and complex modelling procedures alone. Stated another way, if one of the candidate fuels proves to be more durable, that candidate axiomatically must be an improvement in the state-of-the art. Those improvements would be of interest to both PWR and BWR nuclear steam system operation. Also, it is highly probable that the interest in operations would include both power rate changes and eventually extended burnup applications.

- 2) The program seeks fuel concepts which can operate unrestricted within the operational rates of change defined by the nuclear steam supply system.

There are three general considerations for operational fuel ramping in contemporary light water reactors: 1) fuel imposed ramp rate guidelines; 2) ramp rates imposed by the nuclear steam supply system technical specification limits; and 3) local ramp rates due to control rod movement (primarily BWR). This program is concerned with the elimination of fuel imposed ramp rate guidelines such that fuel can be power ramped within technical specifications of the reactor system. The program is not concerned with the high, local power ramps associated with control rod movement even though positive test

results may ameliorate, in part, control-rod/power-ramp related problems as well.

- 3) Any fuel concept selected for evaluation must have a substantial existing technical basis for use.

This program appreciates that any new or different nuclear fuel concepts must be selected with considerable attention to the associated risks which might ensue if new fuel concepts prove to be more prone to failure rather than less. The planning requirements for testing and demonstrating fuel, therefore, require that any fuel concepts selected must, in fact, have a substantial existing technical basis for use. Fuel concepts carried into testing and demonstration can only be justified where substantial laboratory and prior irradiations lead one to the conclusion that commercial use would not be detrimental.

- 4) Fuel performance improvements are to be established through test reactor experiments.

The irradiation program involves power ramp testing of new fuel concepts to failure. This will be done through preramp irradiations in both test and commercial power reactors. An initial screening phase irradiation and power ramping program is planned in the Halden test reactor. Preramp irradiation of fuel segments in the Big Rock Point (BRP) reactor followed by power ramp testing in a test reactor is planned later in the program.

- 5) Power reactor demonstrations are planned to show that selected fuel concepts pose no undue risk to commercial operation.

Any fuel concept tested will be demonstrated in the Big Rock Point (BRP) reactor to assure that pilot quantities can sustain a full use cycle of four years without operational failure. It is envisioned that the number of rods thus demonstrated will be large enough to certify performance results to a reasonable assurance. On the other hand, the number of rods demonstrated is restricted by programmatic

constraints and any certification of ultimate commercial use would have to be proved in more massive loadings of the best candidate fuel in a subsequent field demonstration.

The demonstration task envisions a series of four sequential pilot irradiations in the BRP reactor. Approximately 100 fuel rods will be inserted into reloads each year in the period between 1978 and 1981; these will include rods of the reference design and rods of the alternate concepts selected for evaluation.

Correspondingly, similar fuel will be irradiated and power ramp tested in test reactors in the same time frame. In the early years of the program, test results obtained from laboratory and test reactor data will be used to modify the definition of test fuel concepts fabricated for later years.

- 6) The degree of improvement of a new fuel concept is to be relatable to contemporary pellet fuel design.

The fuel concepts chosen for evaluation will be compared to a reference fuel of contemporary pellet design to give greater confidence in the degree of performance improvement. The reference pellet fuel will experience the same irradiation history and ramp testing as the selected concepts. This provides a direct basis for evaluating the degree of fuel performance improvement from the in-reactor experiments and in-reactor demonstration in addition to performance analysis by the best available mathematical models.

- 7) Commercial quality assurance practices are to be used in the design and fabrication of all fuel concepts.

The body of technology which constitutes the technical basis for the fuel concepts being tested will be applied in the same conservative discipline in fuel design and fabrication as that currently in use in the industry. Such a precept assures the utility user that any new fuel forms tested are not a product of laboratory research alone, but have embodied in their use more pragmatic principles which apply to an industrial manufacturer.

All test and demonstration fuel hardware will be designed and fabricated by the Exxon Nuclear Company under conditions representative of current commercial practice. This approach requires the same level of quality assurance and quality control that now abides in the nuclear fuel industry. This approach also means that comparative results between alternate fuel designs will have a relative basis of comparison to reference fuel manufactured from the same industrial source.

The program anticipates an expenditure of \$15.1 million through 1984. Expenditures in FY-1977 totaled slightly in excess of \$1 million and the level of work in FY-1978 is currently planned at \$2.3 million. The first series of test rods and demonstration fuel is under construction with testing in the Halden reactor to begin early in the summer of 1978 and the BRP reactor demonstration irradiations to begin late in the summer of 1978.

The program is being performed through specific tasks. The following paragraphs summarize the essential elements of each task and provide additional details of the program. A more detailed description of current activity is presented in subsequent section of this report.

Task 1 - Assess Fuel-Cladding Interaction (FCI) State-of-the-Technology and Potential Improved Concepts

Task 1 is providing a state-of-the-technology review of fuel-cladding interaction. It is drawing upon available literature and information from other ongoing fuels programs. This review is providing the basis for an assessment and selection of fuel concepts which have the potential for improved power ramping performance. The results of Task 1 are also helping to formulate the direction of the testing and demonstration program.

Task 2 - Testing and Demonstration Program

From the preliminary results of Task 1, the testing and demonstration program is currently planning to evaluate the following concepts: annular pellet fuel, cladding internally coated with graphite, and packed-particle

fuel. In some cases, combinations of these concepts will be evaluated. The basis for selecting these concepts is as follows: 1) they do not represent a significant departure from current commercial technology; 2) there is an adequate data base to permit ease of application; and 3) the data base indicates that the concepts have potential for improved FCI behavior. Other concepts under consideration for later inclusion in the program include packed spherical-particle fuel and barriers or liners on the inside surface of the cladding. The selected reference fuel is the same as the Exxon Nuclear Company standard design.

Task 2 consists of three major parts: out-of-reactor studies, in-reactor experiments, and in-reactor demonstrations.

Task 2A - Out-of-Reactor Studies

Task 2A is being performed to provide laboratory support for the implementation of fuel concepts. As an example, activity has been focused on methods for applying graphite coatings to the inside surface of cladding. This is being followed by the evaluation of outgassing behavior of the coatings to establish suitable design and process specifications. Evaluation of moisture desorption and resorption of packed particle fuels is being initiated to define the conditions required to remove moisture and to define the atmosphere control required to prevent moisture pickup. Additional laboratory work will be performed as needed in support of candidate fuel concepts.

Task 2A is also concerned with the evaluation and investigation of FCI phenomena.

Task 2B - In-Reactor Experiments

Task 2B is concerned with in-reactor experiments to determine the behavioral characteristics and performance limits of fuel concepts selected for testing. Two types of experiments are planned. The first is a series of screening tests in the Halden test reactor. These presently include 24 instrumented test and reference rods. The screening tests are designed to obtain data on the behavior of the fuel concepts during power changes and steady state operation. Selected rods will then be ramped to provide a relative ranking of the ramping performance of the reference design and

the improved concepts. The Halden reactor was selected for its highly developed instrumentation capability.

The second, or primary, phase experiments are designed to define more accurately the power ramp performance of the most promising concept(s) in a test reactor environment more typical of a commercial power reactor. The primary phase testing is presently planning to power ramp 15 rod segments which are to be preramp irradiated in the Big Rock Point reactor under normal commercial conditions. The power ramping of these rod segments in a test reactor will provide a clearer indication of the degree of improvement as compared to the companion reference fuel design.

Fuel rods being designed and fabricated for the in-reactor experiments are subjected to the commercial quality assurance practices of Exxon Nuclear Company. A full complement of design and characterization information is being prepared for the test rods. Plans are currently in progress for developing irradiation history, interim examination, and postirradiation examination data.

Task 2C - In-Reactor Demonstrations

Task 2C provides an in-reactor commercial demonstration of the selected concepts. The purpose of the demonstration is to show that the concepts pose no undue risk to commercial operation. Approximately 400 rods comprising the improved concepts and reference fuel will be irradiated in the Big Rock Point reactor. Fuel assemblies containing demonstration and reference fuel rods will be inserted in the reactor during each of four annual reloads through 1981. Each assembly is expected to remain in the reactor for four years and be subjected to power ramping cycles during normal operation. Segmented rods will be removed periodically for power ramping in the primary phase testing.

The fuel design, process definition, and fabrication are being performed under the commercial quality assurance practices of Exxon Nuclear Company.

A full data package involving design, process development, fuel rod characterization, irradiation history, interim examination and post-irradiation examination is being developed.

Task 3 - Fuel Performance Evaluation

Task 3 serves as a focal point for the collection and evaluation of data developed in this program and other FCI-related programs. The activity provides the overall coordination needed to establish data requirements for characterization, postirradiation evaluations and data analysis. Data collected in Task 3 will be maintained in a report file and a computerized data file system for use in evaluating program results and developing technical bases for commercial fuel design and analysis.

Task 4 - Establishment of Technical Bases

Task 4 is concerned with the transfer of program results to commercial application through the establishment of technical bases for design. They provide the foundation for product design and include technology established through research and development, operating experience and from licensing requirements. They also include the analytical methods and computer codes used in design. In this program technical bases will be established by selecting, from Task 3, appropriate data and analysis procedures which can be used in the design and assessment of fuels with improved power ramp performance.

The functional arrangement of the program tasks is shown in Figure 1.1. The Task 1 review and assessment is intended to provide the basis starting point for the program and establishes the initial direction for the testing and demonstration program. The activities of Task 2 work in parallel to develop additional technology and performance data related to the selected fuel concepts. Task 3 integrates the program results and information from all sources. Task 4 draws upon the results assembled by Task 3, introduces commercial factors, and develops the technical bases for design of improved fuels. Tasks 3 and 4 provide continuous input to the technical assessment to guide the testing and demonstration tasks.

1.4 Program Organization

The performance of the program involves the direct participation of Consumers Power Company (CPC), Exxon Nuclear Company (ENC), and Battelle-Northwest (BNW). The program is under prime contract to CPC from the U.S. Department of Energy (DOE). The majority of the work is under subcontract to ENC and to BNW. The functional arrangement of the program organization by task is shown in Figure 1.2.

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.

BNW is primarily responsible for the performance of the research and development portion of the program and the scientific interpretation of results.

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

FIGURE 1.1: Fuel Performance Improvement Program Task Relationships

FIGURE 1.2: Fuel Performance Improvement Program Organization

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

W. J. Bailey (Battelle-Northwest)

The objective of the Task 1 activity is to assess the state of the technology of fuel-cladding interaction and to identify fuel concepts with potential for improved power ramping capabilities in order to provide direction and technical support for the program scope. The activities will result in the publication of two reports. The state of the technology review and the resulting report presents a summarization of postulated fuel-cladding-interaction mechanisms as reported in the open literature. Utilizing the results of the state of the technology review, an assessment of the fuel concepts with potential for improved power ramping capabilities is being made and will be described in a second report. Those concepts that will be investigated as part of this program and the technical bases which support their selection are also presented in the assessment report. In particular, the assessment report will provide the supportive information and rationale for selection of the primary fuel concepts to be evaluated in this program. Progress on the two activities is described below.

2.1 State-of-the-Technology Evaluation

Comments and suggestions on the draft of the report entitled "State-of-the-Technology of Fuel-Cladding Interaction" (C00-4066-2/PNL-2488) have been received. The draft has been restructured and the comments and suggestions received were incorporated where possible. The final version of the report is being typed and prepared for submittal for clearance. It is planned to have the revised draft ready for clearance submittal during the first week in January. The report will be duplicated and issued as soon as the required approvals for publication have been obtained.

2.2 Fuel Concept Assessment

All comments and suggestions expected on the draft of the report entitled "Assessment of Fuel Concepts" (C00-4066-3/PNL-2490) have been received.

Based on the comments and suggested changes, a major revision of the draft is required and work is continuing to accomplish that end. Included among the modifications is the addition of the experience bases for the primary fuel concepts selected for evaluation. It is planned to submit the revised draft to the Steering Committee for review by the end of February.

3.0 TASK 2 - TESTING AND DEMONSTRATION PROGRAM

M. D. Freshley and D. W. Brite (Battelle-Northwest)

3.1 Test Program Overview

The objective of the testing and demonstration program, which forms the basis for the Fuel Performance Improvement Program, is to conduct a comprehensive, well balanced testing program to provide the data needed for establishing technical design bases for those fuel concepts with improved power ramping capabilities. Activities associated with the coordination of the general testing and demonstration programs have focused on identifying critical needs and activities within each task and assuring that the activities are performed in an integrated and mutually complementary manner. This includes Task 2A, Out-of-Reactor Studies; Task 2B, In-Reactor Experiments; and Task 2C, In-Reactor Demonstrations.

Task 2A activities are involved with providing information needed to support the initial Task 2B irradiation experiments in the Halden Boiling Water Reactor (HBWR) and the initial Task 2C demonstration irradiations in the Big Rock Point (BRP) reactor. Task 2B activities have been involved with developing an optimized in-reactor testing program that will consist of screening tests to be conducted in the HBWR followed by ramping of individual rods in the HBWR ramping rig and primary tests that consist of irradiations to significant burnups in BRP followed by ramping in a suitable test reactor. Task 2C activities involve the design, process development, and fabrication of the screening test rods for irradiation in HBWR and the primary fuel concepts that will comprise the first reload demonstration irradiations in BRP.

Annular pellet, coated-cladding, and packed-particle or vipac fuel are the primary fuel concepts deemed to have potential for improved power-ramping capability that have been selected for initial investigation as part of the Fuel Performance Improvement Program. Selection of these concepts for the first phase of the evaluations is based on: 1) the preliminary results from the Task 1 state of the technology evaluation;

2) the concepts do not represent a significant departure from current commercial technology; and 3) there is an adequate data base to facilitate application.

The combination of the primary concepts to be included in fuel designs for initial screening tests in HBWR are as follows:

- reference fuel
- annular fuel
- coated-cladding/annular fuel
- packed-particle fuel

The initial 100-rod group of rods to be irradiated in BRP will also include about 15 segmented, full-length rods. The fuel segments, or rodlets, which are to be irradiated in BRP and subsequently ramped in a test reactor, will be comprised of:

- reference fuel
- coated-cladding/reference fuel
- annular fuel
- coated-cladding/annular fuel
- packed-particle fuel

Rods with the reference fuel design are included in all the irradiations to provide a basis for evaluating the performance of the fuel concepts with potential for improved power-ramping capability. The coating used in the initial irradiations will be graphite applied to the inside surface of the cladding. Vipac will be in the form of irregularly-shaped, high-density particles produced by high-energy rate, pneumatic impaction (Dynapak) and vibrated into the cladding tubes. As UO_2 sol-gel spheres become available, subsequent irradiations will utilize sphere-pac fuel rods in the place of vipac.

Coordination of sub-task activities is continuing in order to maintain scheduled progress. Critical items are identified and attention is being focused on resolving the problem areas. Evaluation of the moisture and outgas characteristics of graphite-coated cladding in order to establish

an acceptable product specification, vipac process development, installation of the fuel rod loading station and its associated inert atmosphere box, and procurement of the enriched UO_2 are currently among the more important items that have been identified as being critical with respect to the Halden irradiation schedule. Progress in addressing the critical items is being made.

Additional depleted UO_2 was dynapaked to investigate the effect of different processing parameters on particle shape. These included different techniques to remove the compacted material from the cans, preslugging and granulation prior to compaction, and a modified can loading technique. These techniques have resulted in the production of more regular shaped particles with less lamination and a particle size distribution that is more suitable for vibrational compaction.

Results of the initial experiments with depleted UO_2 showed that the size distribution of the pneumatically impacted (Dynapak) particles after removal from the impaction container was approximately 15% + 6 mesh, 35% - 6 + 10, and 50% - 10 mesh. Removal of the UO_2 without the usual pressing action to deform the container was attempted in an effort to obtain larger UO_2 particles. Repeated tapping and pounding with a hammer proved to be a satisfactory technique for removing the UO_2 in view of the relatively small number of impaction billets to be processed. In both techniques, one end of the impacted billet is first removed by machining in a lathe. The size distribution of ORNL-supplied depleted UO_2 removed from the Dynapaked billet by this technique was 80% + 6 mesh, 8% - 6 + 10 mesh, and 12% - 10 mesh.

4.0 TASK 2A - OUT-OF-REACTOR STUDIES

P. E. Hart and L. R. Bunnell (Battelle-Northwest)

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 minimize fuel-cladding-interaction (FCI) failures during reactor ramping. During the first quarter of fiscal year-1978, task activities focused on the development of the graphite-coated cladding fuel concept. This coating technology has been transferred to ENC. In the following section, BNW activities in this area are summarized.

4.1 Offgas Studies

As reported previously,^(4.1) the graphite dags contain significant quantities of H_2 and H_2O following baking at $450^\circ C$. Based on offgas measurements made on bulk dag specimens and on coated cladding, graphite coatings made from Dag 154* contain significantly lower moisture and H_2 levels than Dag RD-5* coatings. This difference in behavior is attributed to the presence of a sodium silicate binder in the RD-5 material. Depending on baking conditions, the moisture levels in the Dag 154 may be as much as a factor of 10 less than in the Dag RD-5. In order to maintain moisture levels in the finished fuel pins within the specified limits, the Dag 154 material will be used as the coating material instead of the RD-5 dag. To further minimize moisture and other impurity concentrations, the specified coating thickness has been reduced from $12.7 \mu m$ (0.5 mil) to $6.4 \mu m$ (0.25 mil).

Based on an ENC-performed evaluation of the effect of the baking cycle temperature on mechanical properties of zircaloy, the baking cycle has been defined as 24 hr at $425 \pm 25^\circ C$. Typical results of offgas measurements performed on bulk dag specimens following baking at $450^\circ C$ for 24 hr at a vacuum of $1.3 \times 10^{-3} Pa$ (10^{-5} torr) or better are summarized in Table 4.1. Although there is a larger than desirable variation in results for specimens from any one baking cycle, moisture and hydrogen contents are within specifications for the coatings. These measurements were made

*Acheson Colloids, Port Huron, Michigan

on bulk dag specimens with thicknesses of 25 μm or less rather than on coated Zircaloy cladding. This is because of the possible reaction of released oxidants with the cladding and solution of hydrogen in the cladding when coated cladding is heated during measurements at temperatures greater than 450°C.

TABLE 4.1. Hydrogen and Moisture Release from Bulk Dag 154 Specimens previously Outgassed at 450°C

Total Gas Released, cc/gm of Graphite (1)	H ₂ , ppm of Graphite	H ₂ , ppm of Fuel (3)	H ₂ O, ppm of Graphite (2)	H ₂ O, ppm of Fuel (3)
12.0	604	0.29	3433	1.63

(1) Measured at 1600°C, 56 mole% H₂, 27% CO₂, 13% N₂, 1.6% CO₂.

(2) Measured at 860°C, average of three runs.

(3) Based on a 2100:1 fuel to graphite weight ratio in Halden fuel pins. On a fuel weight basis, the H₂ and H₂O specifications for the coating are 1.5 and 5 ppm, respectively.

Care must be exercised when interpreting the results of hydrogen and moisture measurements made on graphite. Moisture and hydrogen can react with graphite at temperatures above 400°C. These reactions would affect the measured inventories of these species; however, such reactions are judged not to have a significant impact on the results listed in Table 4.1. Although it would be desirable to make both measurements in the same system at the same temperature and time, such a capability is not available.

4.2 Coating Application and Processing

With the flood coating technique described previously, (4.1) application of the coating to the inside surface of the cladding in more than one layer is desirable in order to assure defect-free coverage. The coating buildup rate of Dag 154, diluted 1:1 by volume with isopropanol to about

11.3% solids, was determined by air gauging, using maximum/minimum diameter measurements to account for cladding ovality. Figure 4.1 is typical of results obtained on a BRP-sized tube 190 cm long. From the figure, it is shown that (1) a two coat application of the dag produces a coating approaching the desired 6.4 μm thickness; (2) the coating tends to be thicker at the ends of the cladding than in the center region, presumably as a result of flow during vertical draining (draining direction reversed after each coat cycle); and (3) the rate of coating buildup is higher near the ends of the cladding tubes, probably because the previous coating "wicks" solvent away from the liquid dag, thereby immobilizing the graphite solids.

The solids content of the dag is of prime importance in determining coating buildup rate. The solids content can be determined by weighing a sample before and after drying to constant weight overnight at 95-100°C. In order to determine the influence of solids content on coating buildup rate, and also to infer the optimum solids content required to produce the specified coating thickness, the solids content was varied on each side of the 1:1 volumetric concentration. Coatings were applied to 51 cm Halden-sized cladding, and thicknesses were measured by air gauging after the application of two coats as shown in Figure 4.2. Data points are an average of four measurements made along the cladding length; the bars indicate the maxima and minima. Coating buildup rate is roughly linear with solids content in the range evaluated, and it is indicated that ~11 wt% solids will produce coatings in the center of the acceptance thickness range. The time allowed for the dag to drain after flood coating, the forced-air drying time, and drying temperature were standardized at 15 minutes each and 80°C, respectively. Adherence to a fixed time-temperature schedule should materially aid in achieving reproducible coating thicknesses.

SEM examination was used to evaluate the effect of the bake cycle on coating integrity. BRP-sized cladding pieces 2 cm long were coated with Dag 154 to a thickness of about 12 μm for examination (Figure 4.3). The

FIGURE 4.1. Coating Thickness Profiles for BRP-Sized Tubing Obtained Using Dag 154 Containing 11.3 wt% Solids. Measurements made in 7.6 cm increments.

FIGURE 4.2. Average Coating Thickness Obtained in Two Coating Cycles as a Function of Weight Solids in Dag 154.

a. As-applied

1 μ m

b. Buffed

c. Buffed and baked 24 hrs at 425°C

FIGURE 4.3. SEM Photographs of Dag 154 Coatings in Three Conditions.

buffed coating is visibly smoother than the non-buffed. After baking 24 hr at 425°C in a 1.3×10^{-3} Pa (10^{-5} torr) vacuum, no cracking is evident. The only visible difference between as-buffed and baked coatings is a slight roughening of the surface during baking.

When initial baking studies were conducted in a rough-pump vacuum (about 50 μ m), cladding surfaces (both coated and uncoated) turned black due to the formation of a ZrO_2 layer. However, the thickness of the observed layer was small, 0.15 μ m (0.006 mil), and is judged to be of no concern relative to cladding performance. Recent baking cycles have been performed at pressures in the 1.3×10^{-3} Pa range without any cladding discoloration. In spite of the water and other oxidants which are driven off the coatings, the cladding is shiny, even under the dag. It appears that the water is being released at a lower temperature in the harder vacuum, thus preventing oxidation of the cladding.

4.3 Friction Testing

In order to obtain quantitative data on the reduction in fuel-to-cladding frictional forces resulting from the use of dag coatings, a pellet push-through test was devised. As shown in Figure 4.4, the test consists of forcing a 25-38 μ m (1 to 1.5 mil) oversized UO_2 pellet through a piece of cladding provided with an entrance taper, and measuring the force required. After the test, the force and the cladding inner surface condition are compared for coated and uncoated cladding. The force is heavily dimension-dependent and considerable scatter in the data is expected, but the test does duplicate the geometry of the real case.

Figure 4.5 shows room temperature data obtained using Dag RD-5. The data indicate a factor of 3 to 4 reduction in frictional force for the coated-cladding case relative to the uncoated case. Examination of the tubing inner surface (Figure 4.6) shows severe galling in the uncoated case while the coated cladding is merely scratched at a few points. The UO_2 pellet was metallized in the case of the uncoated cladding, and was coated with graphite in the coated-cladding tests. The apparatus was modified to test at 400°C in a dry inert gas; work will be initiated shortly using Dag 154.

FIGURE 4.4. Schematic Diagram of Friction Testing System

FIGURE 4.5. Frictional Force as f(Sliding Distance) Coated and Uncoated Tubes

a. Uncoated, severe galling evident.

b. Coated, only damage is occasional scratches.

1 cm

c. Coating removed, scratches show in tubing inner surface.

FIGURE 4.6. Inner Surface of the Tubing After Test.

Advanced Coatings

A literature search was initiated to identify other materials which might be useful as coatings and/or lubricants to alleviate FCI. Criteria used include:

1. Compatible with the nuclear environment; no undesirable irradiation products.
2. Applicable to cladding inner surface.
3. Capable of functioning as lubricant in a dry atmosphere at 400°C and under high loads.
4. Obtainable in high purity.
5. Compatible with other components of the fuel element.

MoS₂ is indicated as a possible coating material other than the materials (graphite and siloxane) presently in use by AECL.^(4.2) However, MoS₂ may not be acceptable for use in a nuclear reactor environment because of the possible introduction of sulfur-containing compounds into the reactor coolant system and reaction with Inconel and stainless steel. According to the literature, MoS₂ performs well as a lubricant in dry atmospheres at high loads, and is available as a colloidal suspension. The irradiation products of MoS₂, as determined by the computer code ORIGEN,^(4.3) are judged to be innocuous at 2590 Gj/kgm. The same is true of graphite. The most abundant irradiation products are listed in Table 4.2.

Note:

Figure 4.6 in Reference 4.1 contains a graphics error (mislabeling of the units on the ordinate). The corrected figure appears as Figure 4.7, on page 28 of this report.

TABLE 4.2 Radiation Products Produced by 2590 GJ/kgm
Irradiation*

1. MoS₂, 6.14g/fuel pin

<u>Element</u>	<u>Grams Discharged</u>
H	1.29×10^{-4}
He	4.85×10^{-4}
Na	4.54×10^{-15}
Mg	1.50×10^{-9}
Al	5.61×10^{-10}
Si	3.64×10^{-3}
P	1.21×10^{-4}
Cl	9.28×10^{-5}
Ar	3.90×10^{-10}
Sr	1.94×10^{-12}
Y	5.37×10^{-14}
Zr	9.36×10^{-6}
Nb	1.07×10^{-6}
Tc	9.05×10^{-3}
Ru	2.83×10^{-3}
Σ Others	$< 10^{-16}$

2. Graphite, 1.96 g/fuel pin

<u>Element</u>	<u>Grams Discharged</u>
H	9.36×10^{-11}
He	1.03×10^{-6}
Li	1.24×10^{-10}
Be	2.32×10^{-6}
B	8.11×10^{-15}
N	1.38×10^{-11}
Σ Others	$< 10^{-16}$

*Assumptions: GE BWR/6, 8 x 8 assembly, with following specifications:

Fuel Pellet O.D.	= 1.06 cm	Pitch	= 1.63 cm
Clad I.D.	= 1.08 cm	Active Height	= 376 cm
Clad O.D.	= 1.25 cm

FIGURE 4.7. Thermal Conductivity of Dag 154 as a Function of Temperature.

References

- 4.1 C00-4066-4, FPIP Program Quarterly/Annual Progress Report, April-September 1977, C. E. Crouthamel.
- 4.2 A. S. Bain, J. C. Wood, C. E. Coleman, "Fuel Designs to Eliminate Defects on Power Increases," Proc. Conference of the British Nuclear Society, p. 56.1 (1973).
- 4.3 ORNL-4628: ORIGEN, The ORNL Isotope Generation and Depletion Code, M. J. Bell, 1973.

5.0 TASK 2B - IN-REACTOR EXPERIMENTS

J. O. Barner and S. R. Wagoner (Battelle-Northwest)

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 rod concepts which have potential for improved operating behavior in LWRs. Of particular importance with respect to the behavior of current designs, is the occurrence of cladding failure which is attributed to fuel-cladding mechanical and chemical interactions (FCI) during power increases. Although fuel rod designs (rod diameters and internal pressures) and reactor operating conditions (coolant pressures and temperatures) are different for PWR and BWR plants, the potential for FCI is generic to both LWR types.

Comparative parametric experiments will be performed with selected improved fuel concepts in instrumented test vehicles in test reactors in a power cycling-ramping mode. The power cycles and ramping conditions will be of sufficient duration, differential power, and ramp rate to provide significant behavioral differences. Testing under conditions that will provide information needed to eliminate currently imposed operating restrictions that are intended to prevent fuel-cladding interactions will be emphasized.

5.1 Summary of Testing Plan

Details of the general testing criteria, the selection of the irradiation facilities, and the detailed test descriptions were presented previously. (5.1)

The four fuel concepts selected for the evaluation are:

- 1) a reference design utilizing a standard solid dished-pellet,
- 2) an annular pellet design with a hole volume equivalent to 10% of a solid, undished pellet,
- 3) the annular pellet design combined with graphite-coated cladding, and,
- 4) a packed-particle fuel design.

The performance characteristics of the four fuel concepts will be evaluated in three sequential groups of screening irradiations to be conducted in the Halden Boiling Water Reactor (HBWR). The results of these comparative irradiations will be utilized to select the most promising fuel concept for evaluation in the primary phase. The primary phase tests will utilize fuel rod segments that were preirradiated in BRP as part of the Task 2C in-reactor demonstrations. Figure 5.1 illustrates the scheduler inter-relationship for the screening phase, the primary phase, and the demonstration irradiations.

5.1.1 Summary of Screening Phase Plan

The HBWR test rig (IFA-518), which will consist of two six-rod clusters, will be alternately power-cycled between a peak conditioning power of 44.5 kW/m and a peak reconditioning power of 29.0 kW/m. After two power cycles and accumulation of a peak burnup of 700 to 850 GJ/kg (8 to 10 MWd/kg), the upper cluster will be moved to the lower position and six fresh rods inserted into the upper position. In another two power cycles and 700 to 850 GJ/kg burnup accumulated, the lower cluster rods will be replaced by the upper rods, and six fresh rods inserted in the upper cluster. This sequence will be repeated one more time. The rods will be individually power-ramped in an IFA ramping rig in HBWR. Thus, six rods will be ramped from the original lower cluster after 700 to 850 GJ/kg burnup, twelve rods will be ramped after cycling irradiation in both the upper and lower cluster positions after a peak burnup of about 1200 to 1400 GJ/kg, and six rods will be ramped from the last upper cluster after 700 to 850 GJ/kg burnup.

Power control during cycling irradiation will be by movement of a silver absorber shield which alternates position between the two six-rod clusters. Power control during ramping will be by a combination of adjusting the ^3He pressure in coils surrounding the test rod and reactor power. The tests have been designed to achieve peak ramping powers as high as 65 kW/m in rods that have accumulated 1700 GJ/kg burnup.

FIGURE 5.1. Task 2B In-Reactor Experiments-Schedule and Information Flow

The IFA-518 testing instrumentation includes inlet and outlet flowmeters and thermocouples to determine total rig power, self-powered axial and radial neutron detectors to determine axial power variations and radial flux tilts, and a failure detector to measure the presence of fission products in the coolant.

The test rod instrumentation includes LVDT combination rod-elongation/failure monitors in the bottom of all rods except those with internal pressure monitors, fuel centerline thermocouples in the upper portion of all six rods in the initial lower cluster, and LVDT internal rod pressure monitors in the bottom of several rods from the first and second loadings of the rig. The combination elongation/failure monitors and the internal pressure monitors will be operative during both the cycling and ramping portions of the tests. The thermocouples will be operative during only the cycling portion.

The test matrix for the HBWR screening tests in IFA-518 is illustrated in Table 5.1. The table illustrates the makeup for each sequential loading of the testing rig and the individual instrumentation included on each test rod.

Detailed design for test rods and the test rig have been completed.

5.1.2 Summary of Primary Phase Plan

Based upon the results of the screening phase tests in HBWR, the fuel concept with the most potential for improved FCI behavior will be selected from the segmented rods available from the Task 2C in-reactor demonstrations for subsequent ramp-testing in the primary testing phase. The test reactor for the primary phase has not been selected. The central rodlet from segmented Task 2C in-reactor demonstration rods will be utilized for the primary phase tests.

TABLE 5.1. Task 2B: Screening Phase Test Matrix

ROD TYPE	H1LC	H1UC			TOTAL TESTED	TOTAL TESTED BURNUP, MWd/Kg
		H2LC	H2UC			
			H3LC	H3UC		
REFERENCE					6	3/10 3/20
ANNULAR					6	3/10 3/20
ANNULAR - COATED					6	3/10 3/20
VIPAC					6	3/10 3/20
TOTALS (DATE REQ'D)	6 (3-78)	6 (3-78)	6 (6-79)	6 (9-80)	24	12/10 12/20

-TEST IDENTIFICATION

H = TEST NUMBER
LC = LOWER CLUSTER
UC = UPPER CLUSTER

- INSTRUMENTATION IDENTIFICATION

E = ELONGATION SENSOR
P = PRESSURE TRANSDUCER
T = FUEL THERMOCOUPLE

= A ROD WITH LISTED INSTRUMENTATION

All four of the potential improved fuel concepts, and a reference design with graphite-coated cladding, will be irradiated as rod-lets in the BRP demonstrations. The graphite-coated-cladding reference design was included in order to separate the individual effect of the graphite coating, if desired.

5.2 Performance Evaluation

Performance evaluation calculations are being run in two areas, i.e., data processing and performance prediction, in order to provide a basis for evaluating the test results.

The presence of the thermocouple in the fueled region of selected test rods will cause localized perturbations in power and peak fuel temperature when compared with rods without thermocouples. Prediction of the

perturbations will allow a better understanding of the in-reactor performance data. The largest perturbations are expected to occur in the reference and the packed-particle designs because fuel is replaced by thermocouple components in these designs, whereas this is not the case in the annular pellet design.

Preliminary calculational results indicate that, for equivalent powers and for equivalent fuel-cladding gaps, insertion of a thermocouple in a pellet reduces the peak fuel temperature by about 150 to 200 C. Axial heat conduction up the thermocouple does not cause any significant temperature perturbation at the thermocouple junction. The calculated temperature decrease caused by the presence of the thermocouple is due solely to the effect of removal of fuel by the thermocouple hole. The effect of gamma heating in the thermocouple on indicated fuel temperature was calculated to be insignificant. Further calculations will be performed to evaluate the effects of the presence of the thermocouple and gamma heating in the thermocouple on packed-particle fuel temperatures.

Calculations were made to compare the relative performance of solid and annular pellet fuel rods operating in a BWR and illustrate how the differences in operating conditions may relate to FCI behavior. The results of GAPCON-THERMAL-3 calculations show that early in life at a heat rating of 42.6 kW/m, the peak fuel temperature in annular pellet fuel is predicted to be about 200 C lower than for solid pellet fuel (Figure 5.2). Although the volumetric average temperature for the two fuel types is not significantly different at startup, the difference increases with burnup, with the fuel temperature being less for annular pellet fuel. The calculations also show that as burnup proceeds, the difference in peak fuel temperature between the two fuel types increases; becoming a constant at about 400 C less for annular pellets than for solid pellets after a burnup of about 8.6 GJ/kg which is the burnup range for the onset of PCI effects (Figure 5.3).

As a result of the lower temperatures in annular pellet fuel, the release of fission products, which are thought to play an important role in PCI

FIGURE 5.2. Predicted Radial Fuel Temperature Distributions in Reference and Annular Pellet Fuel Rod Designs

FIGURE 5.3. Predicted Peak Fuel Temperature in Reference and Annular Pellet Fuel Rod Designs as a Function of Burnup

behavior, should be less because less fuel volume is operating in the high temperature, high release center. In fact, the calculations indicate that the fission gas release in the annular pellet fuel is only 60% of the release expected in the solid pellet fuel at the same linear heat rating.

Another important effect of the lower fuel temperature in annular pellets is to reduce significantly the fuel-to-cladding interfacial pressure during steady-state (Figure 5.4) and ramping conditions (Figure 5.5). Hence, cladding strain, which is one of the requisites for FCI failure by a stress-corrosion-cracking mechanism, should also be significantly lower in annular pellet type fuel. During steady-state operation, the onset of significant interfacial pressure is delayed for annular pellet fuel (Figure 5.4). Hence, at a given burnup, interfacial pressure and cladding stresses are lower for annular pellet fuel than for the reference fuel design. These calculated predictions for annular pellet fuel are based on average pellet properties and do not take into account the added benefit of reduced pellet hourglassing associated with annular pellets compared with solid pellets. Also, for any given power increase, the resulting increases in peak and volumetric average fuel temperature are 20-30% lower for annular pellet fuel than for the reference fuel design (Figure 5.5). Therefore, fuel-cladding differential expansion and the resultant cladding hoop stresses are expected to be 20-30% lower for annular pellet fuel than for solid pellet fuel. In addition, the cladding stress reduction for annular pellet fuel is expected to be operative for both general differential-expansion-induced stresses and localized stresses, i.e., at pellet-pellet interfaces and over radial fuel cracks. The comparisons illustrated in Figure 5.5 represent calculational results from annular and reference fuel rods after fuel-cladding gap closure occurred in both rod types, i.e. after approximately 1300 GJ/kg burnup.

REFERENCES

- 5.1. C.E. Crouthamel, "Fuel Performance Improvement Program," Quarterly-Annual Progress Report, April-September, 1977, C00-4066-4, November 1977.

FIGURE 5.4. Predicted Fuel-Cladding Interfacial Pressure
in Reference and Annular Pellet Fuel Rod Designs

FIGURE 5.5. Predicted Fuel Differential Temperatures in Reference and Annular Pellet Fuel Rod Designs During Power Ramps

6.0 TASK 2C - IN-REACTOR DEMONSTRATIONS

R. K. Welty (Exxon Nuclear Company)

The objective of this task is to demonstrate commercial acceptability of fuel concepts which have the greatest potential for improved power cycling capability. This will be accomplished through irradiation of a statistically significant sample of rods fabricated by commercial processes involving full industrial quality assurance.

This task is structured as a series of fuel irradiations followed by examination at selected fuel burnups. The irradiation of selected fuel concepts will be performed in the Consumers Power Company's Rig Rock Point reactor. A total of approximately 400 fuel rods will be irradiated.

Fuel assemblies containing demonstration fuel rods will be inserted in the reactor with each of four annual fuel reloads through 1981. Each assembly is expected to remain in the reactor for four years and will be subjected to the maximum power levels and power ramping cycles that are consistent with normal operations. A limited number of fuel rods will be removed periodically (at intermediate burnup levels) from each series for destructive examination or power ramping in a test reactor.

6.1 Series 1 Design and Analysis

6.1.1 Halden

Drawings have been completed for Halden test fuel rod assemblies and components. A draft parts list is prepared and specifications are being completed. A specification for the graphite-coated cladding is still being developed. Moisture and hydrogen impurity levels in Dag 154 are still being evaluated. It appears that impurity levels in this coating material may be low enough to be within acceptable limits.

A partial parts list revision for Task 2B, Halden test fuel rods was approved and issued. The parts list revision includes UO₂ powder and long lead time procurement items.

6.1.2 Big Rock Point

A partial parts list for Task 2C, Big Rock Point demonstration fuel rods, was approved and issued by ENC for CPC approval. All specifications and drawings are completed with the exception of the coated cladding product specification.

Four rod designs have been chosen for the Series 1 full length demonstration rods to go into Big Rock Point Reload Batch G-4. These design characteristics are listed in Table 6.1.

Four similar rod designs have been chosen for Series 1 segmented rods to go into Big Rock Point Reload Batch G-4. These design characteristics are listed in Table 6.2.

Four Series 1 fuel assemblies are to be fabricated for Big Rock Point Reload Batch G-4. Bundle maps are shown in Figures 6.1 and 6.2. A total of 80 full length demonstration rods and 16 segmented rods are to be included.

An Irradiation Test Authorization, ITA-77-1, was issued in November. The ITA describes design and fabrication requirements for Series 1 demonstration fuel rods.

Design analyses are continuing to support licensing of FPIP fuel rods in the G-4 reload of Big Rock Point.

The quality assurance plan for Task 2C was revised and issued. The revision includes program nomenclature changes and procedural changes.

6.2 Fuel Process Development

6.2.1 Annular Pellet Process Development

The annular pellet tooling was tested by pressing and sintering UO_2 from various sources, including the depleted material produced at ORNL, with acceptable results. Green pellet and sintered data are listed in Table 6.3.

6.2.2 Packed-Particle Fuel Process Development

Packed-particle fuel vibratory compaction tests were started in the non-atmospheric controlled equipment. The variables tested to date were particle size, percentage of particle sizes, techniques in fuel loading and vibratory compaction time and air pressure on the vibrator. Results are listed in Table 6.4.

6.2.3 Graphite-Coated Cladding Process Development

An evaluation of the effect of graphite coating moisture levels on the susceptibility of the zircaloy cladding to hydride failure is underway. Halden tubes were graphite-coated with Dag 154. Fuel pellets and glass ampules containing various quantities of water have also been prepared. Plans are to seal the pellets and ampules inside the cladding, break the ampules and then heat the capsules for 24 hours at 450°C to react the contained moisture with the coated and uncoated zircaloy. The capsules will then be pinched to rupture the oxide and then reheated for 100 hours. This should provide a means of inducing hydrogen reaction at the cracks in the oxide. The relationship between initial moisture levels and hydriding of the cladding will be evaluated.

To investigate the effects on the zircaloy when outgassing coated cladding, a test was run--six samples from the same tube were heated at 450° ± 10°C for 0, 2, 4, 18, 27, and 51 hours and then tensile tested.

The results are as follows:

<u>Heat Time (Hrs.)</u>	<u>0.2% Yield (KSI)</u>	<u>Ult. (KSI)</u>	<u>Elong. (%)</u>
0	75.8	103.3	22.2
2	76.6	103.3	22.2
4	76.3	103.6	21.7
18	76.6	102.5	22.7
27	75.8	101.6	23.2
51	74.7	100.2	23.7
Specs Require	>60.0	>70.0	>16.0

After 51 hours, there was only a few percent drop in both yield and ultimate strength, and an increase in elongation. Ovality measurements indicate no measurable changes in ovality of any heated tubes. From these results, it is shown that a cladding outgassing treatment of 24 hours at $425^{\circ} \pm 25^{\circ}\text{C}$ will have no harmful effect on its mechanical properties. Similar tests will be run at $500^{\circ} \pm 10^{\circ}\text{C}$.

A processing area has been devoted in the Process Engineering Laboratory to apply graphite to the cladding tubes. Several tests have been conducted to adapt the coating techniques to the production facility. An air gauging system will also be set up in the same area to measure the graphite film thickness.

6.2.4 Process Equipment

The structural steel platform to support the packed-particle fuel and pellet loading gloveboxes has been erected. The gloveboxes are in place. The DRI-TRAIN air purification units are being installed. Installation is expected to be complete by mid-January.

6.2.5 Fuel Pin Components (Halden)

Insulator discs, plenum springs, cladding and lower end plugs have been received for the Halden test pins. The enriched UO_2 and instrumented end caps are expected by the end of January.

6.2.6 Documentation

The PTA, No. 230 (Process Test Authorization) has been completed in draft form. This document will be the controlling information and data package for fabrication of the Halden test pins. A separate PTA will be written for the Big Rock demonstration fuel rods.

TABLE 6.1
CHARACTERISTICS OF SERIES 1
FULL LENGTH DEMONSTRATION FUEL RODS FOR BRP

Reference Fuel

Type	Solid Pellet
Material	UO ₂
Density (% theoretical)	93.5 $\pm$ 1.5
Diameter, mm (in.)	9.436 $\pm$ 0.013 (0.3715 $\pm$ 0.0005)
Length, mm (in.)	7.62 $\pm$ 0.64 (0.300 $\pm$ 0.025)
Dish	Both ends
Dish Volume (%)	2.0 $\pm$ 0.5
Enrichment (w/o U-235)	2.52 $\pm$ 0.05

Annular Fuel

Type	Annular pellet
Material	UO ₂
Density (% theoretical)	93.5 $\pm$ 1.5
Diameter, O.D., mm (in.)	9.436 $\pm$ 0.013 (0.3715 $\pm$ 0.0005)*
I.D., mm (in.)	2.984 $\pm$ 0.127 (0.1175 $\pm$ 0.005)
Length, mm (in.)	7.62 $\pm$ 0.64 (0.300 $\pm$ 0.025)
Dish	None
Enrichment	2.69 $\pm$ 0.05

Packed Powder Fuel

Type	Vibratory compacted particles
Material	UO ₂
Density (particles, % theoretical)	>98
Diameter (clad I.D., mm (in.))	9.677 $\pm$ 0.038 (0.3810 $\pm$ 0.0015**)
Enrichment (w/o U-235)	2.52 $\pm$ 0.05
Smear Density (% theoretical)	87.0 $\pm$ 1.0

TABLE 6.1 (Continued)

Insulator Disks

Material	Alumina
Density (kg/m ³)	3890
Diameter, mm (in.)	9.195 (0.362)
Length, mm (in.)	5.1 (0.200)
Number, Location	2, One at each end of fuel column

Plenum Spring

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)

Fill Gas

Material	He
Purity (%)	99.9
Fill pressure, MPa (atm)	0.10 (1)

Cladding

Material	Zircaloy-2
Condition	CW-Stress relieved
O.D., mm (in.)	11.455 $\pm$ 0.038 (0.451 $\pm$ 0.0015**)
I.D., mm (in.)	9.677 $\pm$ 0.038 (0.381 $\pm$ 0.0015**)
Wall thickness, minimum, mm (in.)	0.831 (0.0327)
Length, mm (in.)	1902.18 $\pm$ 0.38 (74.889 $\pm$ 0.015)
Allowable bow, mm/m (in/ft)	1.25 (0.015)
Coating (present in some rods with annular fuel)	Dag 154
Coating layer thickness, μ m (in.)	6.4 $\pm$ 3.8 (0.00025 $\pm$ 0.00015)

TABLE 6.1 (Continued)

Fuel Rod

O.D., mm (in.)	11.405 ± 0.051 (0.449 ± 0.002)***
Plenum length, mm (in.)	99.08 (3.901)
Fuel column length, mm (in.)	1778.0 ± 6.4 (70.00 ± 0.25)
Allowable bow, mm/m (in/ft)	1.25 (0.015)

* Pellet O.D. = 9.411 ± 0.013 mm (0.3705 ± 0.0005 in.) in coated cladding rods.

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

*** Rods to be selectively etched to fall within a range of ± 0.038 mm (± 0.0015 in.)

TABLE 6.2
CHARACTERISTICS OF SERIES 1
SEGMENTED FUEL RODS FOR BRP

Reference Fuel

Same as demonstration rods.*

Annular Fuel

Same as demonstration rods.*

Packed Powder Fuel

Same as demonstration rods.

Insulator Disks (each segment)

Same as demonstration rods.

Upper Plenum Spring (upper segment only)

Same as demonstration rods.

Fill Gas

Same as demonstration rods.

Mid and Lower Segment Plenum Spring

Material	Inconel X-750
Wire Diameter, mm (in.)	0.805 (0.0317)
Coil O.D., at working length, mm (in.)	8.64 $\pm$ 0.38 (0.340 $\pm$ 0.015)
Free length, mm (in.)	56.90 $\pm$ 0.76 (2.240 $\pm$ 0.030)
Spring force, N (lb)	9.34 (2.10)
Spring constant, N/m (lb/in)	394 (2.23)
Spring volume, mm ³ (in ³)	304 (0.0185)

TABLE 6.2 (Continued)

Cladding

Material	Same as demonstration rods.
Condition	Same as demonstration rods.
O.D.	Same as demonstration rods.
I.D.	Same as demonstration rods.
Wall Thickness	Same as demonstration rods.
Coating (present in some segments with reference or annular fuel)	Same as demonstration rods.
Coating thickness (mm)	Same as demonstration rods.
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 ± 0.005)

Fuel Rod

Number of segments	3
O.D.	Same as demonstration rods.
Allowable bow	Same as demonstration rods.
Plenum length, upper plenum	Same as demonstration rods.
Plenum length, mid and lower plenum, mm (in.)	33 (1.300)
Fuel column length, upper segment, mm (in.)	544.45 ± 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 material, upper segment	Reference fuel
Fuel material, mid and lower segment	Reference fuel Annular fuel Packed Powder fuel

*Pellet O.D. = 9.411 ± 0.013 mm (0.3705 ± 0.0005 in.) in coated cladding rods.

TABLE 6.3
ANNULAR PELLET TEST RESULTS
(10 pellets per test)

Test #1

Depleted UO₂ slugged to a 4.65 g/cc slug density
and sintered at 1700°C for four hours.

Green				Sintered ^(a)		
Length,cm	Diameter,cm	Density g/cc	End Cap	Length,cm	Diameter,cm	Density %TD
1.019	1.279	5.10	slight	0.8047	1.001	95.06
1.021	1.279	5.19	slight	0.7798	1.006	95.05
1.002	1.280	5.38	slight	0.8133	1.017	95.27
0.9936	1.280	5.43	major	0.8103	1.021	95.25
1.034	1.283	5.62	gross	0.8407	1.033	95.37

Test #2

Depleted UO₂ slugged to a 4.65 g/cc slug density remilled
50% of the +40 particles and sintered at 1700°C for four hours.

Green				Sintered ^(a)		
Length,cm	Diameter,cm	Density g/cc	End Cap	Length,cm	Diameter,cm	Density %TD
1.042	1.283	5.70	slight	0.8562	1.039	95.23

Test #3

Depleted UO₂ as received and sintered at 1700°C for
four hours.

Green				Sintered ^(a)		
Length,cm	Diameter,cm	Density g/cc	End Cap	Length,cm	Diameter,cm	Density %TD
1.019	1.283	5.68	none	0.8562	1.039	95.23

TABLE 6.3 (Continued)

Test #4

Depleted UO₂ slugged to a 4.65 g/cc slug density and sintered at 1650°C for four hours.

Green					Sintered (a)			
Length cm	Dia. cm.	Density g/cc	Hole Dia.,cm	End Cap	Length cm	Dia. cm.	Density %TD	Hole Dia.,cm
1.008	1.281	5.32	0.399	slight	0.8067	1.014	93.47	0.315
1.018	1.282	5.42	0.399	slight	0.8202	1.020	93.62	0.318
1.018	1.283	5.50	0.399	slight	0.8392	1.025	93.63	0.318
1.020	1.283	5.59	0.399	major	0.8405	1.030	93.98	0.320
1.016	1.284	5.73	0.399	slight	0.8306	1.037	94.18	0.323
1.025	1.285	6.04	0.400	slight	0.8306	1.052	95.00	0.325

Test #5

Depleted UO₂ and milled lot 90-10 and sintered at 1650°C for four hours.

Green					Sintered (a)			
Length cm	Dia. cm	Density g/cc	Hole Dia.,cm	End Cap	Length cm	Dia. cm	Density %TD	Hole Dia.,cm
1.013	1.280	5.35	0.399	none	0.8189	1.034	89.44	0.320
1.006	1.281	5.46	0.399	none	0.8158	1.040	89.87	0.323
1.013	1.281	5.93	0.399	none	0.8755	1.046	90.72	0.325

Test #6

Depleted UO₂ reslugged to a 4.6 g/cc slug density sintered at 1650°C for four hours.

Green					Sintered (a)			
Length cm	Dia. cm	Density g/cc	Hole Dia.,cm	End Cap	Length cm	Dia. cm	Density %TD	Hole Dia.,cm
1.059	1.284	5.42	0.399	none	0.8600	1.042	90.81	0.323
1.053	1.282	5.54	0.399	none	0.8593	1.047	91.40	0.325
1.039	1.283	5.64	0.399	none	0.8509	1.051	92.25	0.328
0.965	1.283	5.74	0.399	none	0.7945	1.057	92.43	0.328
1.011	1.284	5.83	0.399	none	0.8369	1.060	92.84	0.330

TABLE 6.3 (Continued).

Test #7

Depleted UO₂ slugged to a 4.65 g/cc slug density and sintered at 1650°C for four hours.

Green					Sintered ^(a)			
Length cm	Dia. cm	Density g/cc	Hole Dia.,cm	End Cap	Length cm	Dia. cm	Density %TD	Hole Dia.,cm
1.011	1.283	5.44	0.399	none	0.8204	1.028	94.15	0.318
1.033	1.284	5.72	0.399	slight	0.8519	1.043	94.85	0.323

Test #8

Depleted UO₂ as received and sintered at 1550°C for four hours.

Green					Sintered ^(a)			
Length cm	Dia. cm	Density g/cc	Hole Dia.,cm	End Cap	Length cm	Dia. cm	Density %TD	Hole Dia.,cm
1.036	1.283	5.27	0.399	none	0.8189	1.031	92.94	0.320
1.004	1.283	5.48	0.399	slight	0.8024	1.041	93.85	0.325
1.023	1.283	5.79	0.399	gross	0.8303	1.058	94.71	0.330

(a)

Task 2B pellet density %TD target $94.00 \pm 1.5\%$
 diameter - 1.026 ± 0.0013 cm
 length - 0.8128 ± 0.064 cm
 hole diameter - 0.325 ± 0.013 cm

TABLE 6.4

VIBRATORY COMPACTION TEST RESULTS

<u>Test No.</u>	<u>Particle Size Distribution</u>	<u>Smear Density (%TD)</u>	<u>Conditions</u>
1	55% -6+10 mesh 25% -20+60 mesh 20% -200 mesh	82	Single loading at 30 secs and 4.14×10^5 Pa (60 psi), finished with 90 secs and 5.52×10^5 Pa (80 psi)
2	60% -6+10 mesh 15% -32+100 mesh 25% -200 mesh	82	Same conditions as Test 1
3	55% -6+10 mesh 25% -20+60 mesh 20% -200 mesh	82.7	Loaded in 2 equal parts - 30 secs and 4.14×10^5 Pa (60 psi), finished with 90 secs at 5.52×10^5 Pa (80 psi)
4	55% -6+10 mesh 25% -20+60 mesh 20% -200 mesh	81.4	Loaded in 5 equal parts - same conditions as Test 3
5	60% -6+10 mesh 25% -10+60 mesh 15% -200 mesh	77.4	Loaded for 60 sec at 4.83×10^5 Pa (70 psi)
6	60% -6+10 mesh 25% -10+60 mesh 15% -200 mesh	82.2	Loaded for 60 sec at 6.90×10^5 Pa (100 psi)
7	60% -6+10 mesh 25% -16+60 mesh 15% -200 mesh	85.3	Loaded for 30 sec at 1.38×10^5 Pa (20 psi), finished at 75 secs and $2.76-5.52 \times 10^5$ Pa (40 to 80 psi)
8	60% -6+10 mesh 25% -16+60 mesh 15% -200 mesh	86.4	Loaded for 30 sec at 1.38×10^5 Pa (20 psi), finished at 180 secs and $2.76-6.90 \times 10^5$ Pa (40 to 100 psi)
9	60% -6+10 mesh 25% -16+60 mesh 15% -200 mesh	85.6	Same conditions as Test 8
10	60% -6+10 mesh 25% -16+60 mesh 15% -200 mesh	83.8	Same conditions as Test 8 - except fuel loaded in 2 equal parts

FIGURE 6.1

BRP SERIES 1 DEMONSTRATION ASSEMBLIES NO. 1 AND 2

<p>(H) — 3.82 w/o U-235, UO₂</p> <p>(HG) — 1.2 w/o Gd₂O₃ in
3.82 w/o U-235, UO₂</p> <p>(1) (M) — 2.52 w/o U-235, UO₂</p> <p>(L) — 1.50 w/o U-235, UO₂</p> <p>(2) — 2.69 w/o U-235, UO₂
Annular Pellet</p> <p>(3) — 2.69 w/o U-235, UO₂
Annular Pellet/Coated Cladding</p> <p>(4) — 2.52 w/o U-235, UO₂
Packed-Particle</p> | <p>(5) — 2.52 w/o U-235, UO₂ and Coated-
Cladding Segmented Rod</p> <p>(6) — 2.52 w/o U-235, UO₂ and 2.69 w/o
U-235 Annular Pellet Segmented Rod</p> <p>(7) — 2.52 w/o U-235, UO₂ and 2.69 w/o
U-235 Annular Pellet/Coated
Cladding Segmented Rod</p> <p>(8) — 2.52 w/o U-235, UO₂ and 2.52 w/o
U-235 Packed-Particle Segmented Rod</p> <p>(X) — Inert Rod</p> <p>(H) — Tie Rod</p> <p>(H) — Spacer Capture Rod</p> |
|---|---|

FIGURE 6.2

BRP SERIES 1 DEMONSTRATION ASSEMBLIES NO. 3 AND 4

(H) — 3.82 w/o U-235, UO ₂	(5) — 2.52 w/o U-235, UO ₂ and Coated Cladding Segmented Rod
(HG) — 1.2 w/o Gd ₂ O ₃ in 3.82 w/o U-235, UO ₂	(6) — 2.52 w/o U-235, UO ₂ and 2.69 w/o U-235 Annular Pellet Segmented Rod
(1) (M) — 2.52 w/o U-235, UO ₂	(7) — 2.52 w/o U-235, UO ₂ and 2.69 w/o U-235 Annular Pellet/Coated Cladding Segmented Rod
(L) — 1.50 w/o U-235, UO ₂	(8) — 2.52 w/o U-235, UO ₂ and 2.52 w/o U-235 Packed-Particle Segmented Rod
(2) — 2.69 w/o U-235, UO ₂ Annular Pellet	(X) — Inert Rod
(3) — 2.69 w/o U-235, UO ₂ Annular Pellet/Coated Cladding	(Tie Rod) — Tie Rod
(4) — 2.52 w/o U-235, UO ₂ Packed-Particle	(Spacer Capture Rod) — Spacer Capture Rod

7.0 TASK 3 - FUEL PERFORMANCE EVALUATION

M. D. Freshley and R. E. Schreiber (Battelle-Northwest)

The Fuel Performance Evaluation Task serves as a focal point for the integration and evaluation of data developed from the FPIP and evaluates the results of other FCI-related programs. The task activities are intended to provide the basic resources from which technical bases for fuel design can be established.

Performance evaluation involves establishing the data requirements as they relate to preirradiation characterization, postirradiation evaluations, and data analysis. Evaluation of the fuel concepts under study will utilize the information developed in the various testing phases of this program as well as results obtained from ongoing FCI-related programs supported by DOE, EPRI, and other laboratories. Performance evaluation necessitates thorough characterization of the fuel rods prior to irradiation in order to permit detailed preirradiation and postirradiation data comparisons. The master list of preirradiation information requirements for the Task 2B screening tests (Table 7.1) was finalized. This listing involves extensive data gathering to fully characterize the rods in order to form the basis for performance evaluation and data analysis.

Careful attention to an evaluation of the conditions to which the experiments are exposed in the reactor during irradiation is essential in order to develop the technical bases. Activities were initiated to define irradiation history data requirements for HBWR and BRP.

Interim and end-of-life basin examinations will be supplemented by the destructive examination of selected rods in hot-cell facilities to evaluate the effects of the test parameters on fuel-cladding interactions and reactions. The requirements for these examinations is being assessed.

In addition to evaluating fuel performance and establishing data requirements, the Performance Evaluation Task will establish and maintain a program report

file and will establish and maintain a computerized data file system for use in evaluating program results and developing bases for commercial fuel design analysis. These activities have been initiated.

TABLE 7.1. Characterization Requirements for FPIP Task 2B Fuel Rods

CHARACTERISTIC	REQUIREMENT ⁽¹⁾	FORM OF INFORMATION	RESPONSIBLE ⁽²⁾ ORGANIZATION
1. Rod Identification	Rod no. on end plug and cladding tube serial no.	Provide inspection sheets.	QC ⁽³⁾
2. Rod Type	Refer to as Reference, Annular, Packed-Particle, Annular-Coated, Thermocouple, Elongation Sensor and Pressure Sensor rods.	Provide inspection sheet relating type of rod to identification number on end plug.	QC & MPE
3. Rod			
a. Length	Precision ± 0.002 in corrected to 68°F.	Provide data sheet.	QC
b. Weight	Precision ± 0.1 gram.	Provide data sheet.	MPE
c. Diameter	ENC: Hand micrometer, 3 locations with two orientations, 0 and 90°, precision ± 0.0001 inch. Kjeller: Nominal axial profilometer traces, 45° apart, axial, circumference reference on rod, precision ± 0.0001 inch.		MPE BNW (After shipping)
d. Component Location	Three full-length (including end caps) radiographs, 60° apart (overlapping segments). Include in the x-rays a stepped wedge of width dimensions measured to ± 0.001 inch. The stepped bar shall be centered along the rod length to provide an axial reference point for the two overlapping x-rays along the rod length. Resolution capability should be sufficient to detect a large, out-of-place chip of fuel, approximately 30 mils thick (viewed edgewise), and roughly 150 mils in the other dimensions.	Provide radiographs. Verify instrumented end cap internal arrangement.	QC ME & BNW

TABLE 7.1. (contd)

CHARACTERISTIC	REQUIREMENT ⁽¹⁾	FORM OF INFORMATION	RESPONSIBLE ⁽²⁾ ORGANIZATION
e. Fuel Enrichment Identification	Pellet identification mark or powder identification card noted on loading inspection sheet.	Record and retain inspection sheet (two signatures).	QC
	Passive assay	Provide passive assay data.	MPE
f. Fuel Column density.	Packed-particle and pellet axial density variation.	Provide gamma densitometer traces (neutron radiography optional).	MPE
	Packed particle average column density.	Provide powder fuel weight and column length and calculated density.	MPE
g. Weld Conditions	Radiograph, voids plus inclusions no greater than 7 mils	Provide radiographs. One orientation.	QC
h. Instrumentation	Verify thermocouple rods in working order with electrical continuity after autoclaving. (Seal with epoxy after checks.)	Provide test results for continuity and resistance.	MPE
i. Fill Gas	Certify >99.9% He of purchased fill gas.	Provide data sheet containing gas analysis.	QC
	Verify prototypic rod sample atmosphere >99.5% He (at least 50 ml free volume).	Provide analysis result.	QC
j. Loading procedure	General description of technique	Description (non-proprietary).	MPE
k. Surface Treatment	General description: cleaning, etching, autoclaving.	Description (non-proprietary).	MPE
4. Cladding			
a. Purchase Order	Reference.	On file at ENC.	MP
b. Fabricator Certs	Records of ingot, trex, tube, as available.	Provide copies of records.	QC
c. Overcheck Results	As available.	Provide data sheets.	QC

TABLE 7.1. (contd)

CHARACTERISTIC	REQUIREMENT ⁽¹⁾	FORM OF INFORMATION	RESPONSIBLE ⁽²⁾ ORGANIZATION
d. Surface Treatment	Describe and characterize result: routine treatment of inner and outer surfaces, also special preparation in vicinity of welds, especially with coated cladding.	Provide description (non-proprietary).	MPE
e. Lot identity	Lot identification.	Retain tubing lot identification.	QC
f. Tube identity	Tubes identified with respect to section in original tube.	Identification mark on tube. Provide inspection sheet showing location and orientation.	QC & MPE
g. Dimensions	ID, OD, wall to precision ± 0.0002 in.	Retain charts.	MPE
h. Coating Process	Summarize process procedures. Measure average thickness on each cladding. Characterize thickness variability on samples.	Provide summary, data sheets, photos.	MPE
i. Coating Morphology	Characterize particle orientation and coating density (%TD). Characterize defects before and after loading.	Provide results.	BNW
j. Coating Chemistry	Measure oxide layer thickness. Measure H pickup in clad after bakeout. Chemically analyze coating after bakeout.	Provide results.	QC & MPE
5. Fuel (Both Enrichments)			
a. Material Purchase Order	Reference	On file at ENC.	MP
b. Processing Procedure	General description of technique.	Describe (non-proprietary).	MPE
c. Pellet Location in rod	Must follow behavior of each pellet.	Provide loading record with identity of each pellet and its location in each rod.	MPE
d. Dimensions	Each pellet, diameter and length to precision ± 0.0001 in.	Provide data sheets.	MPE

TABLE 7.1. (contd)

CHARACTERISTIC	REQUIREMENT ⁽¹⁾	FORM OF INFORMATION	RESPONSIBLE ⁽²⁾ ORGANIZATION
	On a 70 pellet/lot basis measure end squareness chamber, dish, land, taper, etc., as applicable.	Provide data sheets. 70 pellets/lot (each pellet of low enrichment).	QC
	On a 70 pellet/lot basis measure the min.-max. diameter difference (circumferentially) to ± 0.0001 in. precision.	Provide data sheets. 70 pellets/lot (each pellet of low enrichment).	QC
	On a 70 pellet/lot basis measure the annular pellet hole diameter (top and bottom) to ± 0.0001 in. precision.	Provide data sheets. 70 pellets/lot (each pellet of low enrichment).	QC
	Photograph packed-particle on SEM for surface condition and size distribution.	Provide photos.	MPE & QC
e. Chemical Content	Analyze 5 pellet/lot or five 7 g. samples of powder fuel for total U, U/O ratio, N, Fe, F, Cl, H ₂ O, B, Cd, other as feasible. Perform isotopic analysis of U for masses 234, 235, 236, 238 on 2 pellet/lot or two 7 g samples of powder fuel. Results to be standard precision.	Provide copies of analysis results (from records if low enrichment).	QC
f. Metallographic Analysis	Photos of 5 pellets/lot or 5 (course mesh) powder fuel samples/lot adequate to determine grain size, porosity size and distribution, and the presence of second phases, 250X and 500X, as-polished and as-etched.	Provide photos with proper identification.	QC
g. Resinter Test for Densification properties	Standard 5 pellets/lot.	Provide data sheets. (From records if low enrichment.)	QC & MPE
h. Absorbed gas and water vapor	Analyze 5 pellets/lot for N ₂ , O ₂ , H ₂ O, total H, and total gas, standard precision.	Provide data sheets. (From records if low enrichment.)	QC

TABLE 7.1. (contd)

CHARACTERISTIC	REQUIREMENT ⁽¹⁾	FORM OF INFORMATION	RESPONSIBLE ⁽²⁾ ORGANIZATION
i. Batch Identification	Each rod.	Reference in inspection records.	MPE
j. Fuel Density	Immersion density, 10 pellets/lot or 3 samples (+6 mesh) powder/lot.	Provide data sheet. (From records if low enrichment.)	QC
	Geometric density, 70 pellets/lot.	Provide data sheet. (From records if low enrichment.)	QC
k. Fuel Weight	±0.1 gram each rod.	Provide data sheet.	MPE
l. Pellet Defects	Categorize defects on cylindrical surface and defects that may produce small chips.	Provide photos of pellets representing marginal appearance, average, excellent appearance.	QC
		Record of marginal, average or excellent on loading sheets for each pellet.	MPE
6. Insulator Disks			
a. Dimensions	Diameter to ±0.0005 in., length to ±0.001 in., 5 disks.	Provide data sheet.	MPE
b. Material	General chemical analysis, 1 pellet	Provide data sheet.	QC
c. Absorbed Gas	Moisture and total gas, 1 pellet.	Provide data sheet.	QC
d. Density	Typical immersion density, 5 pellets.	Provide data sheet.	QC
7. Zircaloy Frits	Moisture and total gas, 3 samples of 2 types of frit in tandem (after bakeout and 24 hr storage).	Provide data sheet.	QC
8. Miscellaneous			
a. Springs	Surface cleanliness.	Describe cleaning (non-proprietary).	MPE
b. Instrumentation Checkout	Inspect end caps for damage.	Radiograph, single orientation.	QC & MPE

TABLE 7.1. (contd)

CHARACTERISTIC	REQUIREMENT ⁽¹⁾	FORM OF INFORMATION	RESPONSIBLE ⁽²⁾ ORGANIZATION
c. Welding Procedure	General description of assembly procedure.	Describe (non-proprietary).	MPE
9. Archives			
a. Cladding	One 12 in. sample from each tube used for fuel rods, half as coated and baked out and half uncoated.	Retain	MPE collect MP retain
b. Fuel	Five pellets from each batch or 40 g packed-particle fuel.	Retain	MPE collect MP retain
c. End Pellets	Five end pellets from each pellet lot.	Retain	MPE collect MP retain
d. End Plug Material	Six in. of bar stock from each lot.	Retain	MPE collect MP retain
e. Springs	One from each wire lot.	Retain	MPE collect MP retain
f. Rods	One of each fuel form with non-instrumented end caps.	Retain	MPE collect MP retain

1. Quantity of samples is minimum recommended.

2. Responsible organization indicates organization with primary responsibility but may be changed by written direction of ENC Task Leader.

3. BNW = Battelle Pacific Northwest Labs.

ME = ENC Mechanical Engineering

MP = ENC Materials & Purchasing

MPE = ENC Materials & Process Engineering

PO = ENC FPIP Program Office

QC = ENC Quality Control

8.0 TASK 4 - ESTABLISHMENT OF TECHNICAL BASES

K. R. Merckx (Exxon Nuclear Company)

An initial draft of a Technical Bases for the Big Rock Point Reference Fuel for the FPIP was written and distributed for comment. An attempt was made in the initial draft to separate out the fuel rod associated design criteria and bases, introduce general LWR technology, incorporate FRAP-MATPRO analyses and bases, and reorganize the bases so that they could be extended to accommodate the fuel concepts being investigated as part of this program. This initial draft has brought to focus some of the problems of converting commercial technical bases into a project oriented bases where a number of fuel concepts are to be considered using general technology. Initial comments have been received and some reorganization of the bases has been made so that the FPIP objectives can be met more effectively.

8.1 Data Requirements

Data collected in establishing the FPIP Technical Bases is being stored on two files that are continually updated and accessible to the project. One file is a reference file of reports, their summaries, and key words. This file is expanded as new material is reviewed for incorporation into the bases. Literature on the performance evaluation programs, general data file structures and the annular, packed-particle, and graphite-coated cladding fuel concepts has been reviewed and entered into the file. Such data sources will be used along with project developed data to establish the technical bases for the new fuel concepts.

A second file of performance data on the improved fuel concepts has been started that is an extension of the restructured FRAP-S Verification File. Some data from the literature has been entered into this file to identify problems that will be encountered in characterizing the new fuel concepts. An initial version of a report describing the FPIP operational history generation for Big Rock Point irradiated fuels has been written and distributed for review. A general description of

the data sources for the complete FPIP-BRP performance file is described in this report and the performance file has already been expanded to include some of the FPIP generated data. Thus, the performance file from literature sources and the FPIP generated performance file will be of similar structure. The establishment of technical bases, performance files and a project performance evaluation program is being used to establish the data, modelling, and verification requirements encountered in developing the technical bases. This interpretation and verification oriented approach assures that the data collected for the FPIP-generated performance files will be functional and recognize the relative roles of characterization, operational, and behavior measurement data.

APPENDIX A

SI UNIT CONVERSION TABLE

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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