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Fuel Performance Improvement Program

Quarterly/Annual Progress Report

April - September, 1977

Consumers
Power
Company

MASTER

EXXON NUCLEAR
COMPANY, Inc.

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Pacific Northwest Laboratories

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FUEL PERFORMANCE IMPROVEMENT PROGRAM

**Quarterly/Annual
Progress Report**

April - September, 1977

Compiled by C. E. Crouthamel

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

November 1977

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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 demonstrations. 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

D. S. Rowe (ENC)

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-clad 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 and maneuvering. Following an extended shutdown, for example, system start-up 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 leave the nuclear power plant operator with not only options to produce more net electrical power per cycle, but also would provide ways and means for load following; a practice not now used in most of the nuclear power industry.

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 identified and demonstrated in a testing and 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 experiments, in-reactor demonstrations, and performance evaluation 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.

- 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 some 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 some 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 reference and alternate fuel rod concepts will be inserted in reloads each year in the period between 1978 and 1981. The program will monitor the behavior of this fuel through its normal four-year use cycle.

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 sections 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 tasks: 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 post irradiation 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 integration 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 Task 1 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 associated with this task will result in the publication of two reports: one reviews the state-of-the-technology and evaluates postulated fuel-cladding interaction mechanisms; the other provides an assessment of the fuel concepts with potential for improved power ramping capabilities, presents those concepts that will be investigated as part of this program, and presents the technical bases which support their selection. As stated, the assessment report will provide the supportive information and rationale for selection of the primary fuel concepts to be evaluated in this program. Drafts of both reports have been completed and are being circulated for comment. Publication is expected in the near future.

2.1 State-of-the-Technology

A literature search was completed for information on postulated fuel-cladding interaction mechanisms and associated supportive data. Over 150 reports and papers and more than a dozen patents were reviewed. Information derived from the available reports and papers was combined with that from an earlier (1974) PNL/ENC study to produce a draft of the report entitled State-of-the-Technology Review of Fuel-Cladding Interaction. The main conclusions resulting from the state-of-the-technology review are summarized below.

During normal fuel rod operation, including anticipated temperature transients associated with power ramps, the principal mode of cladding failure appears to be stress-corrosion cracking (SCC). In general, it appears that stress (a mechanical process) and fission-product availability (a chemical process) are reciprocal and that a pin-hole-type failure may be the result of a high fission-product concentration

and a low-threshold stress, whereas an axial-split-type failure may be the result of a high threshold stress and low fission-product concentration. It is difficult to separate an SCC failure from a purely mechanical failure; however, crack shape and morphology should indicate the relative roles played by the mechanical and chemical processes. For the high-threshold stress situation, more crack branching and more fracture surface fluting should be expected.

It has not yet been clearly shown experimentally which material parameter (stress, strain, or strain rate) most uniquely defines the threshold for SCC. The current data base supports a threshold-stress concept (i.e., a sustained critical tensile stress in a sustained environment of aggressive fission-product species). Some plastic tensile strain is needed for SCC to occur, but there is question as to whether a critical strain is a prerequisite for SCC. Studies indicate that SCC of Zircaloy is strain-rate dependent; studies on materials other than Zircaloy indicate that SCC occurs within a narrow range of strain rates.

A biaxial stress state appears necessary for SCC to occur. Evidence indicates that SCC can occur at stresses ranging from close-to-the-yield stress to a-fraction-of-the-yield ($\sim 1/3$) stress. Depending on the fission-product species and concentrations, the data suggest that such a threshold range is likely to exist.

Crack initiation in the Zircaloy metal is the key step in fuel-cladding interaction failure. Once the oxide film on the inner surface of the cladding is fissured, small cracks appear to develop in the metal at certain sites (e.g., second-phase particles, pits, grain boundaries). The mechanism for the breakdown of the protective oxide film is under investigation. A critical stress is required for growth of the cracks and, once formed, the cracks propagate fairly rapidly, predominantly by transgranular cleavage. It appears that if the tensile stress or aggressive fission product environment is not sustained, propagation of the cracks will proceed at a slower rate (intergranular path favored in this case) and the crack can arrest.

Sensitivity of Zircaloy to embrittlement by fission products and impurities observed in the fuel-cladding interface region of irradiated LWR fuel rods is under investigation. The likely aggressive fission-product species of concern are iodine vapor (released from CsI), liquid cadmium, and liquid cesium, in that order. Characteristics of the cladding, fuel pellet and fuel rod that may influence SCC are identified.

At present, the changes in power that frequently occur during reactor operation are not monitored in sufficient detail to permit an evaluation of the fuel-cladding interaction phenomenon or to accurately predict the effects of fuel-cladding interaction. There is also a distinct difference between the results of power ramping experiments in test reactors and those from power reactors. It has been postulated that this difference may be due to disturbance of the fuel-cladding interface during physical manipulation of the test fuel rods.

Results of the state-of-the-technology evaluation show that the following remedial actions are likely to alleviate the fuel cladding interaction problem:

- Eliminate or impede contact and/or condensation of those fission products that promote stress corrosion cracking (e.g., iodine) or embrittlement (e.g., cadmium and cesium) in the Zircaloy cladding.
- Minimize or eliminate stress and/or strain localization in the cladding resulting from pellet/cladding thermal-mechanical interaction. For instance, minimizing the radial temperature gradient in the fuel would have the effect of reducing the thermal-mechanical driving force for fuel-cladding interaction, as well as reducing the release of aggressive fission products from the fuel.

These results, which pertain to LWR fuel operating in both BWR and PWR environments, were used to establish the criteria used to select the fuel concepts for evaluation.

2.2 Fuel Concept Assessment

Based on the results of the state-of-the-technology review, fuel concepts which are likely to reduce fuel-cladding interaction effects, and thus have the potential for improved power ramping capability, were identified and those concepts that will be investigated under this program were selected. The selection criteria are shown in Figure 2.1. The assessment was completed and the draft report, entitled Assessment of Fuel Concepts, was issued for comment. Final preparation and issuance of the assessment report will take place when the comments have been resolved and approval has been obtained from the Steering Committee.

I. TECHNICAL CRITERIA ADDRESSING DEFECT MECHANISMS:

GENERAL:

- REDUCE TENSILE STRAINS AT THE INSIDE SURFACE OF THE CLADDING AND/OR REDUCE PROPENSITY FOR STRESS CORROSION CRACKING (SCC) AND/OR EMBRITTLEMENT

SPECIFIC:

- A. REDUCE GENERAL AXIAL AND CIRCUMFERENTIAL STRAINS
- B. REDUCE LOCALIZED AXIAL AND CIRCUMFERENTIAL STRAIN CONCENTRATIONS
- C. REDUCE SUSCEPTIBILITY TO SCC AND EMBRITTLEMENT

II. TECHNICAL CRITERIA AFFECTING DEVELOPMENT AND LICENSABILITY

- DEVELOPMENT EFFORT
- FABRICATION EFFORT
- EFFORT TO LICENSE
- LICENSE MARGIN

III. DEGREE OF ACCEPTABILITY AND PROBABILITY OF DEVELOPMENT

- DEVIATION FROM EXISTING DESIGN BASES AND CRITERIA
- DEVIATION FROM CURRENT FABRICATION PRACTICE OR ESTIMATION OF COMMERCIAL VIABILITY

FIGURE 2.1. Selection Criteria for Candidate Fuel Concepts

3.0 TASK 2 - TESTING AND DEMONSTRATION PROGRAM

M. D. Freshley (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 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 Experiments; Task 2B, In-Reactor Experiments; and Task 2C, In-Reactor Demonstrations.

Task 2A activities concentrate on providing critical 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. Specifically, the Task 2A activities involve evaluating coated cladding and the characteristics of packed-particle type fuel.

Task 2B activities concentrate on developing an optimized in-reactor testing program that is responsive to the objectives of the Fuel Performance Improvement Program. The Task 2B program will consist of screening tests and primary tests. The screening tests consist of cycling highly-instrumented test rods in the HBWR followed by ramping of individual rods in the HBWR ramping rig. During the cycling phase of the HBWR irradiations, cladding elongation, fuel temperature, and internal gas pressure of the test rods will be monitored.

For the primary tests, rods will be irradiated to significant burnups in BRP followed by ramping in a suitable test reactor. These test results will provide a comparative evaluation of the fuel concepts being studied which have the potential for improved power-ramping capability.

Task 2C activities concentrate on design and process development of the primary fuel concepts that will be fabricated for the first reload demonstration irradiations in BRP. Some of the approximately 100 rods for each of the four reloads will be segmented to facilitate ramp testing following their irradiation in BRP.

3.2 Primary Fuel Concept Selection

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; 3) there is an adequate data base to facilitate application; and 4) the data base indicates that the concepts have potential for improved FCI behavior. The state-of-the-technology evaluation identified other fuel concepts with potential for improved power-ramping capability that may be evaluated as the program progresses.

The combination of the primary concepts to be included in fuel designs for initial testing in HBWR and irradiation in BRP has been selected. These include:

- 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 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. Other coatings, which may be more effective in reducing fuel/cladding friction and/or in providing a barrier between the fuel and cladding, may be evaluated in subsequent irradiations. For the initial irradiations, vipac will be in the form of irregular-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-pak fuel rods in the place of vipac.

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 experiments in support of the development and demonstration of the fuel concepts which will minimize fuel cladding interaction (FCI) failures. During fiscal year 1977, major emphasis was placed on the development of coated-cladding and packed-particle concepts which will be evaluated in the initial irradiations in Halden Boiling Water Reactor (HBWR) and Big Rock Point (BRP).

4.1 Coated-Cladding Studies

Based on the demonstrated success in preventing defects associated with power ramps reported by D. G. Hardy, et.al.^(4.1) graphite will be the coating material for the initial irradiations. Other coating systems will be evaluated for inclusion in future irradiations.

Evaluation of graphite dag systems, supplied by Acheson Colloids Company of Point Huron, MI, is in progress. One of these dags is being used in the CANLUB process. As indicated in Table 4.1, all three dags contain graphite flakes, which are approximately $1 \times 5 \times 5 \mu\text{m}$ in size. The fineness of flake size enhances the lubricating nature of these graphites over dags containing coarser flakes. Aquadag and RD-5 have water bases, while Dag 154 is an isopropanol base system. In addition, Dag 154 contains a cellulose-base thermoplastic binder, while Dag RD-5 contains sodium silicate. Dag RD-5 is also sold under the Deltaforge 31 designation for use as a forging die lubricant at temperatures up to and exceeding 300°C . The lubricating nature of graphite is normally dependent on the presence of moisture. Because moisture levels within the fuel rod must be at a minimal level, graphite coatings that rely solely on the presence of moisture for their lubricating characteristics, may exhibit limited lubricity during irradiation.

TABLE 4.1. Description of Dag Products Being Examined

Aquadag: An aqueous-based, colloidal graphite dispersion containing 22 wt% graphite in the form of flakes $1 \times 5 \times 5 \mu\text{m}$. It is supplied as a thick paste and must be diluted before use. Ammonium hydroxide is used in the dag as a deflocculant.

Dag 154: A dry film lubricant containing about 20 wt% solids, principally graphite in the form of flakes roughly the same size as those in Aquadag, with isopropanol as a vehicle. This dag contains a cellulose-based thermoplastic binder. As supplied, it has the consistency of heavy cream and is diluted with isopropanol before use.

Dag RD-5: A dry film lubricant also containing about 20 wt% solids, chiefly graphite in the form of flakes roughly the same size as those in Aquadag, with water as a vehicle and containing ammonium hydroxide as a stabilizer. It also contains sodium silicate, presumably as a high-temperature binder. As supplied, it has the consistency of heavy cream and is diluted with distilled water for application.

The sodium silicate present in Dag RD-5 is considered to promote lubrication when the dag is used in a low moisture environment like a fuel rod. Therefore, of the three dags being evaluated, RD-5 would probably yield the coating with the best lubricating characteristics.

Spectrochemical analysis results (Table 4.2) indicate high silicon and sodium levels in the RD-5 dag, reflecting the presence of sodium silicate in the system. Dag RD-5 also contains a significantly higher concentration of aluminum than the other dags. The remaining cation impurities are present at equally low levels in the three dags.

In the following discussion, experimental work pertinent to the development of the coated-cladding concept is described in detail. This work indicates, based on coating buildup characteristics, that Aquadag is not a prime candidate material.

TABLE 4.2. A Spectrochemical Impurity Analysis (ppm) for Graphite Dags Under Investigation

<u>Element</u>	<u>Dag Designation</u>		
	<u>Dag 154</u>	<u>Aquadag</u>	<u>RD-5</u>
Al	5	5	200
Ca	100	300	200
Cu	--	1	--
Fe	200	200	200
Mg	3	10	50
Mn	2	8	5
Mo	--	50	--
Na	100	100	3,000
Si	10	20	15,000

No other elements were detected. Duplicate samples were submitted. Analysis was semiquantitative and is good to $\pm$ factor of two.

4.1.1. Application Methods

The development of a method for applying graphite to the cladding inside surface has received major emphasis during the quarter. In the past, graphite coatings have been applied to various substrates with a variety of techniques including chemical vapor deposition, ion plating, swabbing, spraying and flood coating. Based on discussions with commercial applicators, the first two techniques are not appropriate for coating the interior surfaces of long pieces of small-diameter zircaloy cladding. The final three techniques require that the graphite be suspended in a liquid carrier. For the near term, the flood coating technique is the most viable one. Both the swabbing and spraying techniques would require development of specialized equipment. With flood coating technique, the inside of the cladding is literally flooded with the dag

by a moving plug. The technique is simple, requiring minimum equipment development and is amenable to coating tubes of virtually any length with minimum dag wastage and spillage. The technique is shown schematically in Figure 4.1. The rate of dag application is controlled by moving the Silastic plug downward through the tube which is held in the vertical position. At a travel rate of 30 cm/min or less, uniform coatings are achieved if the dag is diluted so that about $2.5 \mu\text{m}$ of graphite is deposited per pass and the tube is reversed end for end between passes.

It is anticipated that the graphite will be deposited to a thickness of about $13 \mu\text{m}$ (0.5 mil). In order to determine the rate at which the layer thickness increases, three-foot sections of cladding were flood coated with Dags RD-5 and 154. After drying at about 100°C , the dag was cleaned away from the end regions of the section and air gauge measurements made on the cleaned area, cladding immediately adjacent to the cleaned area, and in the middle of the tube. The measurements were repeated after each coating cycle. Figures 4.2 and 4.3 show coating thickness as a function of the number of coats. Dag 154 builds up at a higher rate and generally exhibits less thickness variation, i.e., $\pm 2.5 \mu\text{m}$, than RD-5. Some of the RD-5 coating thickness variation is probably due to the fact that it tends to be rougher than Dag 154. In both cases, the buildup in thickness is approximately linear.

These results clearly indicate that uniform graphite coatings can be applied to long cladding section with the flood coating technique.

APPLICATOR CONCEPT FOR FLOOD COATING WITH MINIMUM QUANTITY OF DAG

FIGURE 4.1. Schematic Drawing of Flood Coating Application Technique

FIGURE 4.2. Dag 154 Coating Thickness as a Function of Number of Coating Cycles

FIGURE 4.3. Dag RD-5 Coating Thickness as a Function of Number of Coating System

4.1.2 Coating Moisture Release and Off-Gas Behavior

Experiments are being conducted to evaluate moisture release and pickup and gas release behavior of the graphites. In order to ensure that volatile products are at low concentrations in the applied coatings, outgassing conditions for dags are being evaluated as well as the effect of storage conditions on the pickup of moisture by the dags.

Due to the potential hydriding of cladding, moisture, hydrogen, and hydrocarbon levels in the dags are of prime concern as well as the concentration of halogens, which can cause corrosion.

The maximum temperature at which outgassing of the applied dag can be performed is limited by the temperature to which the zircaloy cladding can be subjected without adversely affecting

cladding mechanical properties. Time at a high temperature may have to be limited as well to prevent property modifications. Based on thermogravimetric measurements (Figure 4.4), the loss of volatile components starts to occur at temperatures as low as 200°C. The weight loss at the lower temperatures is attributed to volatile components used to stabilize the dag suspension. At higher temperatures, i.e., above 400 to 500°C, more tightly-bound components, like oxygen, are released. Due to the high strength of the carbon-oxygen bonds, oxygen is released as predominantly carbon monoxide requiring the breaking of carbon-carbon bonds. Therefore, it is clearly evident that substantial outgassing can occur at temperatures above 450°C, which has been identified as the highest temperature to which cladding can be exposed for time periods up to a few hours without changing cladding properties. Outgas measurements have been made in both vacuum and in flowing argon at constant heating rates and under isothermal conditions. As shown in Figure 4.4, a spectrographic-grade

OUTGAS BEHAVIOR IN FLOWING ARGON

FIGURE 4.4. Outgas Behavior of Spectrographic Graphite and Bulk Dag RD-5 as a Function of Temperature at Heating Rate of 300°C/hr in Argon

graphite did not exhibit any significant weight changes during heating to 800°C, indicating that the weight decrease of the dag cannot be attributed to oxygen contamination of the system resulting in oxidation of the test specimen. Weight-loss runs have been made to date largely on bulk dag specimens, 0.13 to 0.65 cm thick, which have been dried at 100°C in air for more than 6 hr to remove the liquid carrier.

After drying in air, Dag 154 loses approximately 21 wt% while RD-5 and Aquadag lose 13 to 14 wt% on heating to 800°C in either vacuum or argon. The higher weight loss of Dag 154 is attributed to the presence of the thermoplastic resin.

Variation of heating rates between 150 to 400°C/hr was not found to affect the total weight loss. However, due to potential effects of high heating rates on the integrity of the coatings, most experimental runs are being made at a relative low heating of 150°C/hr. At high heating rates, it is possible that rapid evolution of the gas could affect the aligned structure which could in turn affect gas release behavior. Such an effect would be less likely to happen at lower heating rates.

Experiments are in progress that will define outgassing behavior of the dags under isothermal temperatures of 350 to 450°C. Initial results indicate that after outgassing for 180 min at 450°C (Figure 4.5), Dag RD-5 continues to lose weight. After 3 hr at 450°C, the graphite contains in excess of 1900 ppm moisture. Additional tests are being performed for extended time periods to determine whether moisture content can be reduced to lower and more acceptable levels. In addition the composition of retained gases are being analyzed. Tests are being developed to define whether the presence of moisture in the coating will adversely effect cladding performance even when the total moisture within the fuel-lubricant system is within present fuel specifications.

FIGURE 4.5. Outgas Behavior of Bulk Dag RD-5 in Vacuum During Heating to 450°C and During a Three Hour Hold at Temperature

At an outgas temperature of 350°C to 450°C, the release of oxidants, like water, will result in cladding oxidation. After outgassing at 400°C in vacuum, cladding specimens have been observed to be oxidized as indicated by darkening under the area where the coating had been applied and on uncovered areas as well. Scanning electron microscope (SEM) examination of outgassed specimens revealed the presence of a 0.1 to 0.2 μm layer, judged to be oxide. The thickness of the oxide layer will be determined more accurately by weight-gain techniques.

4.1.3 Orientation of Graphite Platelets

The individual graphite platelets in the dags are approximately $1 \times 5 \times 5 \mu\text{m}$ in size with the $1 \mu\text{m}$ direction being perpendicular to the layer planes. Because lubrication in graphite occurs by shear between layer planes, lubrication would be most effective if the individual platelets were parallel to the surface being lubricated. The alignment of platelets on dag-coated glass slides was examined by X-rays, and the Bacon Anisotropy factor^(4.2) was computed. The Bacon Anisotropy factor is a measure of basal plane alignment in graphites. Values range between zero and infinity with randomly-oriented graphite being 1 and a perfect graphite, single crystal being infinity. A value of about 20 was measured on as-deposited dags, 13.4 on as-deposited pyrolytic graphite, and 98 on stress-annealed pyrolytic graphite. The values measured for the dags indicate very good alignment of the crystallites parallel to the deposition plane.

4.1.4 Abrasion of Coating by Pellets During Loading

Because the coatings will be subjected to abrasion by the UO_2 pellets during the rod loading operation, it is necessary to evaluate coating damage during pin loading. Two-meter long tubes of BRP-sized Zircaloy were flood coated with dag to a thickness of about $13 \mu\text{m}$ (0.5 mil) followed by buffing with cotton felt. Five simulated UO_2 pellets were made by coring a ZrO_2 rod 0.8 cm long and 0.944 cm (0.3715 in.) diameter weighted with lead slugs to produce a total pellet weight equal to that of a 93% TD UO_2 pellet. These pellets were carefully weighed and slid the length of a horizontal coated tube for a total of 100 passes without rotating the tube. All debris was collected and weighed, as were the pellets. The experiments were done with both RD-5 and Dag 154 coatings. Total weight loss for both materials was about 1 mg amounting to about 0.2% of the coating weight.

Examination of the coating showed scratches in the coating but no penetration through the coating to the Zircaloy. Tests will be repeated with actual UO_2 pellets.

4.1.5 Coating/Zircaloy Adherence

During irradiation, the graphite coating will be exposed to thermal cycling, and for maximum effectiveness the coating must remain adherent during cycling. In order to assess the adherence during thermal cycling, half-sections of BRP-sized Zircaloy cladding were coated with the three dags and then thermal cycled in a well-gettered Ar atmosphere. Adherence was judged visually and by testing with cellophane tape. The results are summarized in Table 4.3. Adherence of the RD-5 was consistently superior to that of the other materials. Coatings were evaluated which had been either slowly or rapidly heated to 450°C to remove volatile components prior to thermal cycling. Heating rate does not appear to have affected test results. Grit blasting with $50\text{ }\mu\text{m}$ Al_2O_3 particles did not appear to enhance adherence. However, an optimum surface conditioning may exist which would enhance adhesion.

4.1.6 Thermal Conductivity of Coatings

Because the graphite coating will occupy a portion of the critical pellet-cladding gap, it is important to know the thermal conductivity of the coating layer. There is an abundance of conductivity data available for solid graphites, but none for layered, highly-oriented, deposited dag. Therefore, the thermal conductivity of dag was determined experimentally. The method used required a specimen in the shape of a disc 0.6 cm diameter by 0.038 cm thick. Only one of the three dags, Dag 154, could be prepared to this thickness in crack-free specimens. Other materials exhibited interlayer cracking. Because the

**TABLE 4.3. Results of Thermal Cycling Tests
On Lubricant-Zircaloy Adherence**

<u>Dag Type</u>	<u>Heating Rate Max. Temp.</u>	<u>No. of Cycles</u>	<u>Observations</u>
Aquadag	$\frac{16^{\circ}\text{C}/\text{min}}{300^{\circ}\text{C}}$	20	Coating appeared adherent, visually. Not tested with tape. (a)
Aquadag	$\frac{4^{\circ}\text{C}/\text{min}}{300^{\circ}\text{C}}$	21	Coating appeared adherent, but tape removed coating. (b,c)
Aquadag	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	11	Coating severely buckled. (a)
Aquadag	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	11	Coating severely buckled. (a)
Dag 154	$\frac{16^{\circ}\text{C}/\text{min}}{300^{\circ}\text{C}}$	20	Coating appeared adherent, visually. (a) Not tested with tape.
Dag 154	$\frac{4^{\circ}\text{C}/\text{min}}{300^{\circ}\text{C}}$	21	Coating appeared adherent, but tape removed coating. (c)
Dag 154	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	11	Coating severely buckled. (a)
Dag 154	$\frac{4^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	11	Coating severely buckled. (a)
RD-5	$\frac{16^{\circ}\text{C}/\text{min}}{300^{\circ}\text{C}}$	20	Coating appeared adherent visually. Not tested with tape. (a)
RD-5	$\frac{4^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	21	Coating appeared adherent, but tape removed coating. (c)
RD-5	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	11	Coating appeared adherent. No buckling. (a)
RD-5 (baked at $6^{\circ}\text{C}/\text{hr}$ to 450°C)	$\frac{16^{\circ}\text{C}/\text{min}}{350^{\circ}\text{C}}$	20	Coating adherent with no buckling, but could be removed with tape.
RD-5 (baked at $6^{\circ}\text{C}/\text{hr}$ to 450°C)	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	22	Coating adherent with no buckling, but could be removed with tape.
Dag 154 ($50\text{ }\mu\text{m Al}_2\text{O}_3$ grit-blasted surface)	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	21	Poorly adherent on both grit-blasted and shiny surfaces with buckling.
RD-5 ($50\text{ }\mu\text{m Al}_2\text{O}_3$ grit-blasted surface)	$\frac{16^{\circ}\text{C}/\text{min}}{500^{\circ}\text{C}}$	21	No buckling, but coating could be removed from blasted and shiny surfaces with tape.

(a) +Behavior of $2.5\text{ }\mu\text{m}$ coating only.

(b) *None of the dags could be removed with tape before the cycling test.

(c) ++Identical behavior was noted for 2.5 and $25\text{ }\mu\text{m}$ coatings.

graphite flake is essentially the same in the three dags and because all three dags yield similar structures, the results for Dag 154 are believed to be representative of all three dags.

The thermal diffusivity was measured on two discs of Dag 154, prepared by dipping to give a structure similar to that applied by flood coating. Measurements were made by the flash technique reported by W. J. Parker, et al.,^(4.3) on specimens heated in purified argon at temperatures from 125 to 500°C.

The thermal conductivity was calculated from measured values of thermal diffusivity using the relationship

$$K = 4.186 \alpha \cdot C_p \cdot \rho$$

where

K = thermal conductivity (W/cm°C)

α = thermal diffusivity (cm²/sec)

C_p = heat capacity (cal/g°C)

ρ = density (g/cc)

Measurements were made transverse to the basal planes and are valid for heat transfer in that direction only. Heat transfer will be in the same direction during fuel operation. The density used was 1.42 g/cm³, determined by physical measurements of a bulk deposit. The values for C_p are those given for grade GBH graphite in Thermophysical Properties of High Temperature Solid Materials, edited by Y. S. Touloukian.^(4.4) No thermal expansion correction was used for either α or ρ , because no appropriate data are available. Such corrections are expected to be small at the temperature of interest.

Because specimens were not outgassed prior to the first measurement cycle, they showed a nonreversible increase in conductivity after being heated to 350-400°C. This increase is probably due to elimination of organic material in the as-deposited samples. Once higher temperatures were reached, no hysteresis was noted

in the data, indicating that there were no further changes in structure. The thermal conductivity of the Dag 154 at 100°C is about 10% of that for pyrolytic graphites measured in the same (c-axis) direction reported by B. T. Kelley.^(4.5) The latter material shows a decrease in thermal conductivity with increasing temperature, while the value for the dag increases slightly with temperature in this range. The pyrolytic graphite is essentially pore-free, while the dags are 50% dense or less. Entrapped gas in the pores of the solid dag may result in the positive temperature coefficient.

A least-squares fit of the data was obtained, eliminating data points from specimens not previously heated to at least 400°C. The equation for thermal conductivity of the dag is:

$$K = 4.33 \times 10^{-3} + 6.64 \times 10^{-6}T - 6.19 \times 10^{-9}T^2$$

where T = temperature (°C). The equation and data points are included in Figure 4.6.

FIGURE 4.6. The Thermal Conductivity of Dag 154 as a Function of Temperature

Subsequent to the calculation of a density from physical measurements, which is difficult because of the irregularities in the specimen, the mercury displacement technique was used to measure the density better before and after baking. Using this technique, the Dag 154 had a density of 1.21 g/cm^3 before baking, in fair agreement with the 1.42 g/cm^3 by physical measurements. After baking to 300°C in vacuum, however, the density dropped to only 0.90 g/cm^3 . The RD-5 dag had an as-deposited density of 1.30 g/cm^3 , dropping to 1.18 g/cm^3 after baking. The net effect of the density decrease of thermal conductivity would be to multiply the K resulting from the equation above by a factor of 0.83 for baked RD-5, and 0.63 for baked Dag 154. The equation would only represent beginning-of-life conductivity for the graphite. As the pellet and cladding make contact during irradiation, the graphite will be densified by the interfacial pressure so that its conductivity will increase. This increase would amount to a multiplier of 1.59 for the conductivity equation if the dag were pressed to theoretical density.

The comparatively low thermal conductivity measured on the dag is probably attributable to the rather low density of the deposit. Photomicrographs of the deposits reveal only a small number of large pores, so much of the porosity must be very fine. The expected effect of neutron irradiation will be to decrease the conductivity of the individual graphite crystals further, to perhaps 10% of the unirradiated value.^(4.6) The major thermal resistance of the graphite is apparently due to its low density, so the order-of-magnitude change in the conductivity of the graphite flakes will probably not be reflected in a similar change in the conductivity of the dag.

In order to obtain a rough upper limit on the effects of the coating, the temperature drop across the graphite coating was approximated. Using the relationship

$$\Delta T = \frac{Qt}{K}$$

where

ΔT - temperature drop, °C

Q = heat flux, W/cm²

t = coating thickness, cm

K = thermal conductivity, W/cm-°C

and assuming that $Q = 57.4 \text{ kW/m}$ (175 W/cm^2), $t = 0.001 \text{ in.}$ ($2.54 \times 10^{-3} \text{ cm}$) and $K = 3.8 \times 10^{-3} \text{ W/cm-}^\circ\text{C}$, a temperature drop across the coating of 117°C was calculated. This is for the worst case, Dag 154, with a density of 0.9 g/cm^3 . The assumed power is high and the coating thickness is also high. At 45.9 kW/m and $13 \text{ }\mu\text{m}$ for these parameters, the ΔT across the graphite is 47°C .

4.2 Packed-Particle Fuel Studies

In support of the packed-particle (vipac) fuel concept, studies have been initiated on the moisture release and pickup characteristics of the vipac feed material. The results of these studies will be used to define outgassing conditions required in order to meet the specification for moisture level in the fuel and to define moisture level control during fuel pin loading operations and processing of the powder.

In order to achieve the required smear densities, the packed particle fuel feed material is dynapacked followed by sintering at 1700°C . To meet required fuel densities, resulting materials must have densities exceeding 98% of theoretical. Sizing of the feed material is required to achieve high packing efficiencies. To gain the desired particle size distribution, crushing of the dynapacked material is required. Based on prior experience, crushing was judged to be best performed prior to the sintering step based on the observation that the sintered and crushed materials are rougher and have higher surface areas than.

crushed and sintered materials. The higher surface area could result in higher pickup of contaminants and could be more difficult to compact to required densities as well.

Initial moisture release measurements were made on crushed and sintered, depleted UO_2 powder with a density of about 99% TD. The powder was outgassed for 3 hr in a vacuum of 10^{-4} mm at 450°C . Listed in Table 4.4 are moisture contents after outgassing and after storage in dry argon for 24 hr.

TABLE 4.4. Moisture Content in Vipac Feed Material
After Outgassing in Vacuum, ppm

<u>Fraction Size</u>	<u>After Outgassing</u>	<u>After Storage in Dry Argon for 24 hr</u>
-6 +10 mesh	<5, <5, <5	<5, <5, <5
-20 +65 mesh	<5, <5, 8	5, 8, 5
-200 mesh	34, 32, 30	35, 32, 33
Weighted Average ^(a)	9	9
50/25/20 Blend of above size fractions	7, 7, 8	11, 10, 10

(a) <5 ppm assumed equal to 2.5 ppm.

Of the three size fractions examined in triplicate, the coarsest material retained little moisture after outgassing. Moisture retention levels increase with decreasing particle size with the minus 200 mesh material containing 32 ppm moisture. In addition to the three size fractions, a master blend was evaluated containing 55% coarse, 25% intermediate, and 20% fine material. The moisture content in the mixed powder is approximately equal to the calculated concentration based on the individual components, indicating consistency between measurements. As shown in Table 4.4, moisture levels in material stored in dry argon for 24 hr are slightly higher than in those analyzed immediately following outgassing.

The moisture content will be examined after storage in dry argon and helium for time periods as long as ten days. Powders will be outgassed at higher temperatures in order to achieve lower moisture levels.

REFERENCES

- 4.1 D. G. Hardy, A. S. Bain, and R. R. Meadowcraft, "Performance of CANDU Developed Fuel in the NRU Reactor Loops," presented at ANS Topical Meeting on Water Reactor Fuel Performance, St. Charles, IL, 1977.
- 4.2 G. E. Bacon, "A Method for Determining the Degree of Orientation of Graphite." J. Appl. Chem. 6:477, 1956.
- 4.3 W. J. Parker, R. J. Jenkins, C. P. Butler, G. L. Abbott, J. Appl. Phys. 32:1679, 1961.
- 4.4 Thermophysical Properties of High Temperature Solid Materials, Vol. 1, Elements. Y. S. Touloukian, ed., Macmillan Co., NY, p. 238, 1967.
- 4.5 B. T. Kelley, Chemistry and Physics of Carbon, Vol. 5. P. L. Walker, ed., Marcel Dekker, NY, 1969.
- 4.6 R. J. Price, "Thermal Conductivity of Neutron-Irradiated Reactor Graphites." Carbon 13:201, 1975.

5.0 TASK 2B - IN-REACTOR EXPERIMENTS

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The objective of the in-reactor experiments task is to conduct irradiation tests to study the detailed behavioral characteristics and performance limits of those fuel rod concepts which have potential for improved operating behavior in LWRs. Of particular interest 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.

The in-reactor experiments are to be conducted with improved fuel concepts which have the potential for minimizing the generic FCI problem. Considering the restrictions imposed by the testing and demonstration facilities, the testing criteria, and the scope of the program, the experiments will necessarily be conducted at operating conditions different than commercial plants. In some cases, the operating conditions may more closely simulate those which exist in a PWR, i.e., the rod differential pressures and associated cladding stress conditions during operation in Big Rock Point (BRP) are similar to a PWR. In other cases, the operating conditions may more closely simulate a BWR, i.e., cladding coolant heat transfer in both BRP and the Halden Boiling Water Reactor (HBWR) is in the boiling mode. In other cases, the operating conditions may be typical of neither commercial reactor type, i.e., the coolant pressure in HBWR is lower than for both LWR types. However, because the experiments will be conducted in a comparative manner, the test results will be indicative of the degree of amelioration of FCI for both PWR and BWR fuel rod designs.

Selected improved fuel concepts will be irradiated in instrumented test vehicles in test reactors. Comparative parametric experiments will be subjected to irradiation in a power cycling-ramping mode. The power cycles

and ramping conditions will be of adequate duration, differential power, and ramp rate to provide indications of 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.

The following discussion presents the status of the in-reactor planning and design activities for the in-reactor experiments. The discussion proceeds logically from determination of the testing criteria and general testing plan, to selection of the testing facilities, to, finally, test rod and rig design.

5.1 General Testing Plan Development

5.1.1 General Testing Criteria

The general testing criteria for the in-reactor experiments were established as follows:

a) Testing scheme should address the defect mechanism.

This criterion is based upon a presumed defect mechanism that is localized fuel-cladding mechanical interaction accentuated by fission product chemical attack. The mechanism may occur during any period after fuel-cladding contact has been made; however, it is particularly prevalent during a power increase when differential thermal expansion between the fuel and the cladding leads to high localized stresses in the cladding. A possible increase of released fission products caused by the increased fuel temperatures during the power increase, together with fission products previously released, may result in an increased propensity for stress-corrosion cracking. Therefore, the testing scheme should address strain concentrations, as well as the general strain condition in the cladding, the propensity for stress-corrosion cracking, and the differential expansion between fuel and cladding associated with a power increase.

- b) Operating conditions should be relatable to commercial conditions.

The test conditions should approximate or be more severe than the anticipated failure conditions in commercial reactors. In this way, behavioral differences between possible improved concepts may be more easily evaluated. This criterion does not mean that the test conditions should be exactly the same as those which exist in a commercial reactor. This would not be possible, because every rod in a commercial plant is exposed to different operating environments and histories.

- c) Testing phases should be comparable.

The results from the two testing phases, i.e., screening and primary, should be relatable and comparable within themselves. A reference design should be included in all tests so that comparisons of results can be made to a constant datum.

- d) Tests should be scheduled to maximize information feedback.

The tests should be scheduled and results obtained in a manner which maximizes feedback of behavioral information for inclusion into the selection process for later tests. Instrumentation affixed to operating rods will provide early behavioral information.

- e) Tests should provide supportive information for design analysis.

The tests should provide supportive information for data and design analysis, as well as information suitable for any operational safety uncertainties.

- f) Number of tests should provide adequate indication of degree of improvement.

Finally, the number of tests should provide an adequate indication of the degree of improvement of tested concepts with respect to the reference design.

5.1.2 General Testing Plan

It was anticipated that several possible improved concepts might be selected for inclusion in the in-reactor parameter testing task. Most efficient utilization of the available resources for irradiation testing of the improved concepts results in a reduction in the number of possible concepts as testing proceeds. Therefore, the irradiation testing plan was separated into two phases: 1) a screening phase that would compare the general behavior of the potential improved concepts to the reference design; and 2) a primary phase that would compare the best performing concepts from the screening phase to the reference design. The screening phase irradiations in HBWR will utilize non-preirradiated rods. However, the primary phase will utilize rods previously irradiated in the BRP demonstration irradiations as part of Task 2C for subsequent ramp testing in a test reactor.

5.1.3 General Test Plan Characteristics - Screening Phase

The general testing criteria define the required characteristics of the screening phase test plan:

a) Defect Mechanism

In order to address the defect mechanism, adequate burnup should be achieved to produce the onset of fuel swelling and produce a significant fission product inventory within the rods. Burnup levels in the range from 8-20 MWd/kg are considered adequate.

During power cycling operation, the steady-state power levels should be maintained for periods adequate to promote a significant degree of stress equilibration (i.e., reconditioning) between the fragmented fuel and the cladding. No more than two power cycles per 10 MWd/kg burnup were selected.

Overpower ramps will be necessary to produce significant differential-thermal-expansion-induced cladding stresses. Ramps in test reactors have typically overpredicted differential powers which have caused failures in commercial reactors. Therefore, ramp power levels should be designed to be significantly higher than anticipated for commercial plants.

b) Operating Conditions

Most cladding failures during power increases in commercial reactors have been associated with the peak power rods. During normal operation, the peak linear power in current generation PWRs and BWRs is approximately 44.5 kW/m (13.5 kW/ft). Therefore, this power level was selected as the peak steady-state power during the preramp cycling operation.

Several normal reactor maneuvers can result in power increases in fuel rods (e.g., a general total reactor power increase after extended operation at reduced power), rod banking or balancing of burnup in various regions of the core, or assembly shuffling during refueling. Whole rods or regions of rods may undergo power increases from a few to hundreds of percent during these operations. It is not possible to simulate all these power increases. Therefore, selection of a typical differential power for preramp cycling operation must be somewhat arbitrary. Selection of a small differential power would be expected to produce insignificant results. Selection of a large differential power would also be expected to produce insignificant results, because the original power from which the increases were initiated would have to be so low that no meaningful fuel-cladding interaction or "deconditioning" would have occurred prior to the cycle. Therefore, a moderate differential power from a level where some "conditioning" or

"reconditioning" (selection of terminology depends upon whether the rod has experienced prior cycles) has occurred would produce the most meaningful results. A value of 65% of the peak power was selected as the power level for the lower power portion of a cycle. Thus, return to the peak power from the lower portion of a power cycle will result in a 54% power increase. The ramping to the over-power condition after the cycling operation will also start from the lower cycling power level, i.e., 29 kW/m (8.8 kW/ft). Multiple power cycling prior to ramping is expected to produce a measure of cumulative damage in the cladding.

The selection of a ramping differential power is also somewhat arbitrary. Although as mentioned in the previous section (5.1.3a), it should be high, because test reactor results typically overpredict the conditions in commercial reactors that result in FCI failures. Therefore, a differential ramping power of approximately 100%, i.e., peak power of about 57.5 kW/m (17.5 kW/ft), has tentatively been selected. In order to provide adequate testing flexibility, the maximum capability for ramping differential power will be designed for 125%, i.e., peak power of 65 kW/m (19.8 kW/ft), after a burnup of 20 MWd/kg. Final selection of actual differential ramping powers will be based upon modeling analyses, cycling results, and interim examination results, and will be determined just prior to the actual ramp testing.

The peak power increase rate dictated by a commercial reactor system should be the upper limit for power increase rates. This rate, approximately 10% of peak power per minute, will be used during the ramp testing. Rates during the preramp cycling operation will be limited to the rate of power increase of the test reactor.

In order to adequately characterize test operating conditions, test vehicle instrumentation is required. Measurement of coolant temperatures, coolant flow, axial power distribution, radial power tilts and rod failure are necessary.

c) Comparable Tests

The screening phase irradiations are intended to compare the behavior of the improved fuel concepts to each other and to the reference rod design. The rods or groups of rods under test should operate under similar conditions. This will be accomplished by irradiating groups of rods side by side during the cycling testing, although individual irradiations might also be completed satisfactorily if carefully controlled.

d) Information Feedback

The rapid feedback of test results for evaluation and inclusion in the decision-making process for subsequent experimentation is vital to achieving the objectives of the test program. Because irradiation tests cover long periods of time, utilization of on-line instrumentation, interim examination, and properly sequenced postirradiation examinations will be required to optimize feedback of results. The general test schedule and information flow diagram for the entire Task 2B experiments is illustrated in Figure 5.1. Test descriptions are presented in Section 5.3. The obvious flow of information from interim and postirradiation examinations has been omitted for reasons of simplicity. The schedule clearly illustrates the need for timely information flow from earlier tests to later tests.

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

e) Supportive Information

The bulk of direct supportive information as a function of irradiation time (e.g., fuel centerline temperatures, cladding elongations, and internal rod pressures) will be generated in the screening phase because these are the only rods that can be instrumented. Other supportive information in the form of fission gas release, fuel temperature, cladding strain, etc., will be generated during the post-irradiation examinations and, to a lesser extent, during the interim examinations.

f) Number of Tests

The number of tests should be adequate to rank the behavior of the improved concepts. This will allow selection of the improved concept(s) to be tested in the primary phase.

5.1.4 General Test Plan Characteristics - Primary Phase

Several of the items described above for the screening test plan characteristics are applicable to the primary phase tests. Among these are the discussions of burnup, ramp differential powers, ramp rate, and feedback of results. Characteristics which can be expected to differ for the primary phase are rod-power history and cycling operation, actual rod-power level, and direct supportive information.

The general power cycling and rod powers during the demonstration irradiations will be determined by the actual operation of BRP. Control of these characteristics is limited to rod placement in the assembly and placement of the assembly within the core. Ramping conditions for rods preirradiated in BRP will be based upon modeling analyses and interim examination results. Figure 5.1 also illustrates the primary phase scheduling and information flow. The source of the rods for the primary phase ramp tests from the BRP demonstrations is also

shown. Rods are available from BRP approximately once per year. A series of four primary phase test groups from BRP is tentatively planned. After power ramp testing, an optional reconditioning and reramping is illustrated even though these options are not within the currently planned program scope. Selection of the improved concepts to be tested in the primary phase will be based upon the screening-phase results.

5.1.5 Behavioral Indices

The behavioral indices, method of measurement, and time of measurement are listed in Table 5.1. Interim and postirradiation methods of measurement are grouped under the common heading of PIE. Obviously where destructive examinations are required, they would not be applicable to the interim examinations. Where in-reactor instrumentation is listed, the method of measurement is applicable only to the screening phase tests. Failure of rods during cycling or ramping is considered the number one index. However, a careful weighing of the primary indices will be used to evaluate the testing results.

5.2 Selection of Testing Facilities

The use of test reactors for simulation of commercial operating conditions requires compromises between design characteristics, operational characteristics, and experimental costs. Several of these characteristics were compared for three reactors: the Halden Boiling Water Reactor (HBWR) in Halden, Norway, the General Electric Test Reactor in Vallejos, California, and the R-2 reactor in Studsvik, Sweden. Other reactors were considered but were found to be either representative of the three listed above or were incapable of accommodating the planned tests. The comparison of the three reactors is listed in Table 5.2 and is divided into three categories: reactor characteristics, test facility capabilities, and instrumentation capabilities.

TABLE 5.1. Behavioral Indices

<u>Index</u>	<u>Primary</u>	<u>Method of Measurement</u>
1. <u>Cladding Breach</u>	During Test: PIE:	Fission Gas Monitors Elongation Monitors Collapsible Bellows Leak Check Eddy Current Visual Exam Neutron Radiography
2. <u>Incipient Breach</u>	During Test: PIE:	Elongation Sensors Eddy Current Ultrasonics Metallographic Visual Exam
3. <u>Localized and General Strain Accumulation</u>	During Test: PIE:	Elongation Sensors Length Diameter Residual Gap Metallographic
4. <u>Propensity for SCC</u>	During Test: PIE:	Rod Pressure (Selected) Fuel Temperature (Selected) Fission Gas Release Fission Product Migration (SEM, EMX, Autoradiography) Gamma Scan Metallographic
5. <u>Chemical Interaction</u> <u>Mechanical Interaction</u> <u>Temperature Distribution</u> <u>Cladding Mech. Condition</u>	<u>Secondary</u> During Test: PIE:	Fuel Temperature (Selected) SEM EMX Chemical Analysis Metallographic Radiographic Autoradiography Gamma Scan Hardness Mechanical Properties

TABLE 5.2. Comparison of Testing Facilities

Item	H8WR (Norway)	GETR (USA)	R-2 (Sweden)
<u>Reactor</u>			
Type	Heavy Water	LWR	MTR
Core Height, m(in.)	1.80 (71)	0.91(36)	0.61(24)
Thermal Flux	Low	Typical	Typical
Fast Flux	Low	Slightly High	Slightly High
Peak Power Shift/Cycle, Axial cm/%	Nil	15/20	N.A.
<u>Test Facility</u>			
Type/Cycling	Rig	Capsule	Capsule
No. Rods/Ass'y-Cycling	6-12	1	4
Type/Ramping	Rig	Capsule	Loop
No. Rods/Ass'y-Ramping	1	1	1
Facility Location	In-Core	Pool	In-Core
Test Pressure, MPa(psi)	3.45(500)	7.24(1050)	7.24(1050)
Fueled Length, m(in.)	0.5(20)	0.9(36)	0.6(24)
Peak/Ave.Power-Fueled Length	1.1	1.8	1.5
Peak Power-Cycling	Adequate	Adequate	Adequate
Peak Power-Ramping	Adequate	Adequate	Adequate
Peak Ramp Rate, $\frac{\text{kW}}{\text{m-sec}}$ ($\frac{\text{kW}}{\text{ft-min}}$)	0.058(1.07) ⁽¹⁾	>0.075(1.35)	0.075(1.35)
Fuel Enrichment, %	10-12	2-5	2-5
Cycling Method	Absorber Screen	Move Capsule	Move Capsule
Ramping Method	³ He	Move Capsule	³ He
Cooling Mode-Cycling	Natural or Forced	Natural	Natural
Cooling Mode-Ramping	Natural	Natural	Forced (downflow)
<u>Instrumentation</u>			
Flowmeters	Yes	No ⁽²⁾	No
Coolant Temperatures	Yes	Yes	No
Axial Power Distribution	Yes	No ⁽²⁾	No ⁽³⁾
Radial Power Distribution	Yes	No	No ⁽³⁾
Failure Monitor-Vehicle	Yes	Yes	Yes
Coolant Pressure	Yes	Yes	Yes
Rod Elongation	Yes	No	No ⁽³⁾
Fuel Temperature	Yes	No	No ⁽³⁾
Rod Pressures	Yes	No	No ⁽³⁾
Rod Failures	Yes	No	No ⁽³⁾

(1) Capability currently limited to 35 W/cm. Possibility of increase is being studied.

(2) Calculated based on previous experience.

(3) Capability claimed but not proven.

A comparison of each test reactor shows that they offer either:
1) extensive instrumentation capability; or 2) a fuel rod operating environment approximating a commercial power reactor, but, not both. Clearly compromises must be made in test reactor selection.

5.2.1 Screening Phase Tests

The discussion of the general testing plan, Section 5.1, presents a need for a high degree of instrumentation in the screening phase tests. Those tests are to be comparative to the goal of obtaining a simple ranking of improved power cycling behavior and not a quantitative measure of improvement. While it is desired to have a fuel rod operating environment typical of a commercial power reactor, it is not considered essential for purposes of comparative screening of the concepts. Based upon the demonstrated instrumentation capabilities and general suitability for program needs, the HBWR has been selected for the screening phase tests.

An agreement to conduct screening irradiation tests in the HBWR was prepared and is in place. The agreement is patterned after an existing agreement between PNL and the Instituttt for Atomenergi (IFA) and is being handled as a supplement to the base agreement between the Federal Energy Regulatory Commission and IFA.

5.2.2 Primary Phase Tests

Primary phase testing is intended to define the degree of behavioral improvement of one, or two, improved concepts as compared to the reference fuel by means of ramp testing. This phase of testing will, therefore, require a test reactor environment more typical of a commercial power reactor. Furthermore, the test reactor must be capable of ramping fuel rods or rodlets previously irradiated in BRP. An important requirement is that the test reactor is capable of utilizing the

enrichment of BRP fuel rods. Clearly the HBWR cannot provide this environment and another test reactor will be used for the primary phase ramp testing. The selection is yet to be made.

5.3 Testing Plan Development-Screening Phase

5.3.1 Detailed Testing Scheme

Based upon the general testing criteria, the general testing plan, and the selected test reactor for the screening phase, a more detailed testing plan was developed. This plan consists of:

- a) alternate power cycling of two six-rod clusters in an HBWR test rig between a peak "conditioning" power of 44.5 kW/m (13.5 kW/ft) and a "reconditioning" power of 29.0 kW/m (8.8 kW/ft),
- b) power ramping of the six individual rods from the lower cluster in an IFA ramping rig after a peak burnup of 8-10 MWd/kg,
- c) movement of the upper six-rod cluster into the lower position and insertion of six fresh rods into the upper position,
- a) repetition of step a,
- e) repetition of steps b and c, with the lower ramped rods at a peak burnup of 15-20 MWd/kg,
- f) repetition of steps d and e, and
- g) ramping the last six upper cluster rods individually after a burnup of 8-10 MWd/kg.

The physical layout and power distributions of the rods during cycling and ramping are illustrated in Figure 5.2. Six rods from the lower cluster and six rods from the upper cluster will be ramped tested after 10 MWd/kg burnup. Twelve rods from the

FIGURE 5.2. Power Distributions During Screening Tests in IFA-518

lower cluster will be ramp tested after 20 MWd/kg burnup. Thus, 18 rods will be ramped from the lower cluster and six rods from the upper cluster.

Three-fourths of the rods will be ramped following irradiation in the lower cycling position. The highest power in rods from the lower cluster occurs at the top of the fuel column. The degree of fuel-cladding mechanical interaction can be expected to be the greatest at the peak power position. Therefore, with respect to cladding elongation, rods from the lower position can be expected to provide a known gage-length. This will decrease uncertainties in data reduction.

The general cycling and ramping schemes are illustrated in Figure 5.3. Figure 5.3, section a, illustrates the cycling and ramping scheme for the initial lower six-rod cluster, and

(a) INITIAL LOWER SIX-ROD CLUSTER
(b) SECOND AND THIRD LOWER SIX-ROD CLUSTER

FIGURE 5.3. General Cycling - Ramping Schemes

section b illustrates the cycling and ramping scheme for the second and third lower six-rod clusters. The final upper six-rod cluster would be ramped after a burnup of 10 MWd/kg, the midpoint burnup shown in Figure 5.3, section b. The burnup periods illustrated should be considered idealized. Final selection of burnups will be based upon the anticipated behavior that will be calculated from modeling codes. Final ramping differential powers will also be selected based upon anticipated behavior. Interim examinations are planned between the cycling and ramping portions of the tests to ascertain rod conditions prior to ramping.

The screening phase test rig is illustrated in Figure 5.4, and its characteristics are summarized in Table 5.3. Power

FIGURE 5.4. IFA-518 Test Rig Schematic

control during preramp cycling operation will be by rig placement, reactor control, and a silver absorber shield that alternates location between the upper and lower cluster positions. Power control during ramping will be by a combination of ^3He pressure in coils of tubing surrounding the test rod and reactor power. A short steady-state power operating period at the previous peak power that occurred in the cycling rig will be performed to reduce any uncertainties in the fuel-cladding stress condition that might be caused by handling during the interim examination. Rig instrumentation includes inlet and outlet flowmeters and thermocouples to determine total rig power, self-powered axial and radial neutron detectors to determine axial power variation and radial flux tilts, and a failure detector that measures fission products in the coolant.

TABLE 5.3. Screening Phase IFA-518 Test Rig Characteristics

Item	Description
<u>Test Reactor</u>	Halden Boiling Water Reactor
<u>Rig Characteristics</u>	
Geometry-Cycling	Two six-rod clusters, one above the other
Cooling Mode-Cycling	Natural convection
Peak Conditioning Power, kW/m (kW/ft)	44.5 (13.5)
Peak Reconditioning Power, kW/m (kW/ft)	29.0 (8.8)
Peak/Average Power, Lower Cluster	1.1
Peak/Average Power, Upper Cluster	1.2
Power Control-Cycling	Alternating silver absorber shield
Rig Geometry-Ramping	Single rods centered at core midplane
Cooling Mode-Ramping	Forced Cooling
Peak Ramping Power Capability, kW/m (kW/ft)	65 (19.8)
Peak/Average Power-Ramping	1.03
Ramping Rate, kW/m-sec (kW/ft-min)	0.058-0.075 (1.07-1.35)
Power Control-Ramping	³ He plus reactor control
Coolant Flow	Inlet and outlet flowmeters
Coolant Temperature	Inlet and outlet thermocouples
Axial and Radial Power	Eight self-powered detectors
Failure Monitor	Outlet fission product monitor
<u>Rod Characteristics</u>	
Rod Design Basis	BWR-6, 8 x 8, ENC design
Rod Diameters, mm	12.29 x 10.26
Fueled Length, mm	463
Fuel Enrichment, % ²³⁵ U	11.0
No. Power Cycles, 10 MWd/kg	2
No. Power Cycles, 20 MWd/kg	4
No. Rods Ramped from Lower Cluster @ Burnup, MWd/kg	6 @ 8-10 12 @ 15-20
No. Rods Ramped from Upper Cluster @ Burnup, MWd/kg	6 @ 8-10
Combination Rod Elongation and Failure Monitors	All rods except those with pressure monitors
Fuel Centerline Temperature	The six rods in the initial lower cluster
Rod Pressure Monitors	Two rods in the initial lower and upper clusters
Axial End Power Peaking	Prevented by use of intermediate enrichment pellets on ends of fuel column

The design basis for screening phase rods is a design of current vintage, i.e., the ENC design for an 8 x 8 assembly for a BWR-6. Use of rods of significantly smaller diameter would result in unacceptably high ^{235}U enrichments and the associated uncertainties in predicting total rod power and internal rod power distributions. The fueled length for all rods will be 463 mm which is the maximum length that can be accommodated in the ramping rig. The standard 12.29 mm OD x 10.26 mm ID Zircaloy-2 tubing used in ENC designs will be used for all test rods and the tubing will originate from the same cladding lot. All fuel types will utilize the same fueled length and plenum length.

Differences in power due to changing smear densities will be partially compensated for by appropriate placement in the anticipated flux tilt. An initial enrichment of 11.0% ^{235}U will allow ramping powers of 65 kW/m (19.8 kW/ft) after a burnup of 20 MWd/kg. Rod instrumentation includes combination LVDT rod-elongation/failure-monitors in the bottom of all rods except those with internal pressure monitors; fuel centerline thermocouples in the upper portion of approximately four rods in the initial lower cluster; and LVDT internal rod pressure monitors in the bottom of several rods from the first and second loading 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. A test rod with a pressure transducer and in-fuel thermocouple attached is illustrated in Figure 5.5.

The cycling portion of the screening phase will be accomplished in a test rig identified as IFA-518. Ramping will be completed in one of the IFA ramping rigs.

FIGURE 5.5. HBWR Test Rod

5.3.2 Test Matrix

Based upon the results of the Task 1 studies, four primary fuel concepts were selected for testing in the HBWR screening phase tests: 1) reference fuel, 2) annular fuel, 3) annular fuel combined with graphite-coated cladding, and 4) packed particle fuel. Design characteristics of these rods for HBWR are listed in Table 5.4. The annular fuel design maintains the reference fuel-cladding gap and incorporates an annular hole equivalent to 10% of the volume of a solid cylindrical pellet. The annular-coated design utilizes a graphite coating on the inner surface of the cladding. The packed-particle design will utilize angular particles (vipac) for the initial tests. If spherical particles become available for subsequent tests, they will replace this vipac fuel.

The test matrix for the HBWR screening tests in IFA-518 is illustrated in Figure 5.6. The matrix illustrates the four primary fuel types and the test rod instrumentation. The

TABLE 5.4. Design Characteristics of Test Rods

Characteristic	Reference	Annular-Coated	Vipac	Annular
<u>Fuel</u>				
Form	Pellet	Pellet	Particles ^(a)	Pellet
Material	UO ₂	UO ₂	UO ₂	UO ₂
Diameter, mm	10.26	10.26 ^(b)	--	10.26
Length, mm	8.13	8.13	--	8.13
L/D Ratio	0.79	0.79	--	0.79
Density, % T.D.	94	94	≥98	94
Smear Density, % T.D.	87	79	87	79
Dish (% pellet vol.)	Yes(1)	No	--	No
Chamfer	No	No	--	No
Annulus, mm	No	3.25	--	3.25
Enrichment, % ²³⁵ U	11.0	11.0	11.0	11.0
<u>Cladding</u>				
Material ^(c)	Zr-2	Zr-2	Zr-2	Zr-2
Heat Treatment ^(d)	CW/SR	CW/SR	CW/SR	CW/SR
O.D., mm	12.29	12.29	12.29	12.29
I.D., mm	10.52	10.52 ^(b)	10.52	10.52
Wall Thickness, mm	0.89	0.89	0.89	0.89
R/T Ratio	6.90	6.90	6.90	6.90
<u>Rod</u>				
Diametral Gap, mm	0.25	0.25	--	0.25
G/D Ratio	0.024	0.024	--	0.024
Fuel Vol./Gas Vol. Ratio ^(e)	6.20	3.67	4.75	3.67
Plenum Spring	Inconel X-750	Inconel X-750	Inconel X-750	Inconel X-750
Insulator Pellets	Al ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃	Al ₂ O ₃
Anti-peaking Pellets ^(f) (enrichment, % ²³⁵ U)	Ref (2.78)	Ref (2.78)	Ref (2.78)	Ref (2.78)
Fill Gas	He	He	He	He
Internal Pressure at R.T., kPa	100	100	100	100
Fueled Length, ^(f) mm	463	463	463	463
Lubrication	--	Graphite	--	--

(a) Particles produced from dynapacked-crushed-sintered material.

(b) Fuel-coating gap will be adjusted to maintain the same as the fuel-cladding gap in the reference by adjusting pellet diameter or cladding diameter inside.

(c) Zr-2 = Zircaloy-2

(d) CW/SR = cold-worked and stress-relieved

(e) Assumes porosity within fuel is not available as gas space.

(f) Fueled length includes an anti-peaking pellet of the reference geometry on each end of the fuel stack.

ROD TYPE	<u>H1LC</u>	<u>H1UC</u>			TOTAL TESTED	TOTAL TESTED BURNUP, MWd/Kg
		<u>H2LC</u>	<u>H2UC</u>			
			<u>H3LC</u>	<u>H3UC</u>		
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

FIGURE 5.6. Task 2B: Screening Phase Test Matrix

matrix provides for: 1) equal numbers of test rods of each design at two burnup levels, 2) fuel temperature measurements on each design with duplicate measurements on the two concepts which have the smallest data base, 3) internal rod pressure measurement on each design to burnups of 15-20 MWd/kg, 4) duplicate rod elongation measurements on each design to burnups of 15-20 MWd/kg, and 5) for comparative purposes, all rod designs included in each test cluster.

The design package and experimental proposal for the screening tests have been submitted to IFA.

5.4 Testing Plan Development-Primary Phase

Based upon the results of the screening phase tests, the most promising improved concepts will be selected from the rods available in the Task 2C demonstration irradiations for subsequent ramp testing in the primary phase.

As shown in Figure 5.1, approximately 15 primary phase ramping tests will be performed. Rods from the first two BRP series, representing at least two burnup levels, will be selected for testing. The general primary testing scheme is illustrated in Figure 5.7. As mentioned previously, neither the test reactor nor the ramping powers have been selected. In order to 1) maintain the flexibility of choice of the available test reactors for the primary phase ramp testing, 2) reduce the associated test vehicle design and fabrication costs, and 3) maintain reasonable compromises between axial powers, temperatures and fission product distributions in the steady-state and power ramping portions of the testing, an adequate number of rodlets of each of the four primary types will be included in segmented rods in the first two BRP demonstration irradiations (Task 2C). The segmented rods will be fabricated from three segments of approximately equal fueled length. The central rodlet from these segmented rods will then be available for ramp testing in the primary phase of Task 2B. Rodlets from the lower segment will be available as backup rods. Rodlet designs incorporating the reference fuel combined with the graphite-coated cladding will also be included in the segmented rodlets. Inclusion of rodlets of this design will allow separation of the individual effect of the coating, if deemed necessary based upon the test results from the primary concepts. The segmented rods will be placed in the highest power positions that are available for the demonstration rods within the fuel assemblies.

FIGURE 5.7. General Primary Testing Scheme

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 potential for improved power cycling capability. This will be accomplished through irradiation of a significant number 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 Big Rock Point (BRP) reactor. A total of approximately 400 fuel rods will be irradiated in four reloads with about 100 rods in each.

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-0 Design and Fabrication

Initial Series-0 fuel rods have been fabricated and will be inserted into the BRP reactor for irradiation. This series consists of 28 well-characterized "carrier" fuel rods containing standard ENC pellets, four rods with chamfered pellets and eight rods of annealed cladding. All fuel rods were specially characterized with regard to pellet and cladding dimensions. Table 6.1 lists the characteristics of the rods along with Quality Control release numbers and archive samples. The pellet characterization data are shown in Table 6.2 for all of the pellets loaded into these fuel rods. The rods were loaded into fuel assembly number G3-14 according to the fuel rod position map shown in Figure 6.1.

TABLE 6.1. Characteristics of Series-0 Rods, Fuel Bundle No. G3-14

Rod Serial No.	Quantity	Rod Process	Special Cladding Treatment	Cladding Lot Number	Pellet Process	Enrichment	Pellet Lot No.	Archive Sample
IL400001 IL400002	2	Standard	Annealed	0254-000-5479	Characterized	3.82	99-2	5 pellets from lot 99-2
IN400001 IN400002	2	Unautoclaved as etched	Annealed	0254-0000-5479	Characterized	3.82	99-2	5 pellets from lot 100-1
IO400001 IO400002	2	Unautoclaved as ground	Annealed	0254-0000-5479	Characterized	3.82	99-2	5 pellets from lot SR-26-1 R1
IP400001 IP400002	2	Anodized	Annealed	0254-0000-5479	Characterized	3.82	99-2	2 full-length claddings from lot 0254-2000-5479
IZ300001 through IZ300004	4	Standard	None	0254-2000-5027	Undished Chamfered	2.52	SR26-1R1	1 fuel rod serial no. 1H300059A
IH400061A through IH400070A	10	Standard	None	0254-2000-5027	Characterized	3.82	99-2	
IH300042A through IH300058A IH300060A	18	Standard	None	0254-2000-5027	Characterized	2.52	100-1	

TABLE 6.2

Pellet Characterization Data

	<u>Pellet Lot.No.</u>	<u>Sample Size</u>	<u>Average</u>	<u>Standard Dev.</u>	<u>95/95 Range</u>	
					<u>Upper</u>	<u>Lower</u>
Density (%)	99-2	100	93.79	0.2156	94.27	93.31
Diameter (mm)	99-2	100	9.433	0.0077	9.449	9.415
Length (mm)	99-2	100	7.565	0.0783	7.739	7.390
Dish Vol. (%)	99-2	100	1.869	0.06239	2.008	1.730
Density (%)	100-1	100	93.81	0.2309	94.33	93.29
Diameter (mm)	100-1	100	9.434	0.0063	9.448	9.419
Length (mm)	100-1	100	7.542	0.0842	7.729	7.354
Dish Vol. (%)	100-1	100	1.908	0.06842	2.060	1.755
Density (%)	SR26-1*	50	93.15	0.2908	93.84	92.46
Diameter (mm)	SR26-1*	50	9.435	0.0030	9.442	9.428
Length (mm)	SR26-1*	50	7.573	0.0901	7.787	7.358
Cham. Angle (deg.)	SR26-1*	50	52.76	2.65	58.7	46.8

			IZ3 00002	IH3* 00046A	IH3* 00049A		IZ3 00003	IH3* 00050A		
		IH3* 00058A	IL4 00001				IH4* 00068A	IH3* 00044A		
IZ3 00001	IH3* 00043A		IL4 00002				IH4* 00065A			IZ3 00004
	IH4* 00061A	IH4* 00067A						IN4 00002	IN4 00001	IH3* 00054A
IH3* 00048A										IH3* 00047A
IH3* 00057A					IH4 00069A		IH4 00070A			
	IP4 00002	IP4 00001						IH4* 00066A	IH4* 00063A	IH3* 00045A
IH3* 00056A	IH3* 00055A		IH4* 00064A				IO4 00001			IH3* 00042A
			IH4* 00062A				IO4 00002			
			IH3* 00053A	IH3* 00052A			IH3* 00051A	IH3* 00060A		

SPACER S/N THIS SIDE

* CHARACTERIZED PELLETS(REFER TO TABLE 6. 1)

FIGURE 6.1. Fuel Rod Position Map of Big Rock
Point Reload G-3 Fuel Bundle No. G3-14

Reactor operating instructions for fuel assembly G3-14 have been developed for a core position (Cycle 15) in an interior power location, when possible, next to a fluxwire position. The assembly should experience adjacent control rod movement during its incore lifetime. Future fuel shuffles will preserve bundle orientation with respect to operating control rods. These test assemblies are neutronicallly compatible with the other G-3 reload assemblies.

6.2 Series 1 Design and Analysis

6.2.1 Neutronics Design and Fuel Management

Calculations were performed to determine power distributions in the FPIP Series 1 fuel assemblies that contain full-length demonstration fuel rods. Figures 6.2 and 6.3 show the rod type and enrichment distributions of the full-length demonstration rod FPIP fuel assembly design. As seen in Figures 6.2 and 6.3, the only FPIP rod type which required an enrichment change was the annular-pellet type. The enrichment of the annular pellets was selected on the basis of minimizing the change in the local peaking factors while utilizing a standard enrichment.

When compared to the G-3 reference assembly, the insertion of twenty-four (24) FPIP rods in the assembly at the beginning-of-life, with in-channel voids of 25%, results in a similar power distribution with a slight increase ($\sim 2mk$) in the assembly K_{∞} .

For these calculations, the 2.52 enriched standard pellet FPIP rods required no special treatment and were calculated as a standard G-3 fuel rod. The neutronic properties of the coated-cladding annular pellet rods were assumed to be identical to those of a non-coated annular pellet.

The annular pellet rods with an enrichment of 2.69 wt% ^{235}U yield calculated local powers 1% lower than those of the G-3 reference fuel rods. The packed-powder rods, compared to the

(H)	3.82 WT% U235, UO ₂	(5)	2.52 WT% U235, UO ₂ AND COATED CLADDING SEGMENTED ROD	
(HG)	1.2 WT% Gd ₂ O ₃ IN 3.82 WT% U235, UO ₂	(6)	2.52 WT% U235, UO ₂ AND 2.69 WT% U235, ANNULAR PELLET SEGMENTED ROD	
(1)	(M)	2.52 WT% U235, UO ₂	(7)	2.52 WT% U235, UO ₂ AND 0.41 WT% U235 ANNULAR PELLET/COATED CLADDING SEGMENTED ROD
(L)	1.50 WT% U235, UO ₂	(8)	2.52 WT% U235, UO ₂ AND 2.69 WT% U235 ANNULAR PELLET/COATED CLADDING SEGMENTED ROD	
(2)	2.69 WT% U235, UO ₂ ANNULAR PELLET	(X)	INERT ROD	
(3)	2.69 WT% U235, UO ₂ ANNULAR PELLET/COATED CLADDING	(TIE ROD)	TIE ROD	
(4)	2.52 WT% U235, UO ₂ PACKED POWDER	(SPACER CAPTURE ROD)	SPACER CAPTURE ROD	

FIGURE 6.2. Rod Type and Enrichment Distributions of DOE Series 1 Demonstration Assembly No. 1 and 2

FIGURE 6.3. Rod Type and Enrichment Distributions of DOE Series 1 Demonstration Assembly No. 3 and 4

reference fuel rods, have almost the identical quantity of fuel per column unit length. As a result, the calculated local peaking factors using the same enrichment as the standard rods are very similar to those for the G-3 reference fuel rods.

The uncontrolled and controlled calculated local power distribution of the G-3 reference assembly are shown in Figures 6.4 and 6.5, respectively. Figure 6.6 shows the uncontrolled calculated local power distribution of the FPIP assembly. The calculated local power distribution of the FPIP assembly for the two possible control rod positions in the BRP reactor are shown in Figures 6.7 and 6.8.

A fuel bundle depletion calculation has not yet been performed. The method for depleting the annular pellets is currently being investigated. The local power distribution, as a function of exposure, will be determined where these calculations have been made. It is anticipated that these calculational results will show that the burnup-dependent characteristics of the FPIP rods will be very similar to those of standard rods.

Calculations have not yet been performed for FPIP fuel assemblies containing segmented rods. Calculations need to be performed to determine the extent of power peaking in and adjacent to such rods.

6.2.2 Thermal-Hydraulic Design

Fuel temperatures have been calculated for the reference and conceptual fuel designs to be used in the power cycling program in the BRP Reactor. The conceptual fuel designs examined include the annular and vipac fuel designs.

The calculations were performed using an NRC-approved model, with required adjustments for material properties, along with conservative assumptions for densification, swelling, and

25% IN-CHANNEL VOID

1.02	0.88	1.19	1.08	1.03	1.01	1.03	1.08	1.19	0.88	1.02
0.88	1.12	0.97	1.20	1.11	1.08	1.11	1.20	0.97	1.12	0.88
1.19	0.97	1.14	1.02	0.93	0.47	0.93	1.02	1.14	0.97	1.19
1.08	1.20	1.02	0.93	0.88	0.84	0.88	0.93	1.02	1.20	1.08
1.03	1.11	0.93	0.88	0	0.88	0	0.88	0.93	1.11	1.03
1.01	1.08	0.47	0.84	0.88	0.87	0.88	0.84	0.47	1.08	1.01
1.03	1.11	0.93	0.88	0	0.88	0	0.88	0.93	1.11	1.03
1.08	1.20	1.02	0.93	0.88	0.84	0.88	0.93	1.02	1.20	1.08
1.19	0.97	1.14	1.02	0.93	0.47	0.93	1.02	1.14	0.97	1.19
0.88	1.12	0.97	1.20	1.11	1.08	1.11	1.20	0.97	1.12	0.88
1.02	0.88	1.19	1.08	1.03	1.01	1.03	1.08	1.19	0.88	1.02
CONTROL ROD BLADE										

Km = 1.2379

FIGURE 6.4. Uncontrolled Calculated Local Power Distribution of Big Rock Point G-3 Reference Assembly

25% IN-CHANNEL VOID

1.01	0.92	1.29	1.23	1.21	1.23	1.28	1.37	1.51	1.13	1.31
0.79	1.10	1.03	1.34	1.30	1.30	1.38	1.51	1.24	1.44	1.13
0.89	0.87	1.15	1.11	1.07	0.57	1.14	1.27	1.44	1.24	1.51
0.70	0.99	0.98	0.99	1.00	0.99	1.07	1.14	1.27	1.51	1.37
0.61	0.87	0.87	0.93	* 0	1.01	0	1.07	1.44	1.38	1.28
0.59	0.81	0.46	0.87	0.95	0.98	1.01	0.99	0.57	1.30	1.23
0.62	0.84	0.83	0.87	0	0.95	0	1.00	1.07	1.30	1.21
0.70	0.91	0.88	0.87	0.87	0.87	0.93	0.99	1.11	1.34	1.23
0.81	0.74	0.93	0.88	0.83	0.46	0.87	0.98	1.15	1.03	1.29
0.60	0.81	0.74	0.91	0.84	0.81	0.87	0.99	0.87	1.10	0.92
0.65	0.60	0.81	0.70	0.62	0.59	0.61	0.70	0.89	0.79	1.01

* SCR ORIENTATION ROD

Km = 1.0551

FIGURE 6.5. Controlled Calculated Local Power Distributions of Big Rock Point G-3 Reference Assembly

25% IN-CHANNEL VOID

Km 1.2396

* TEST ROD LOCATIONS

** SCR ORIENTATION ROD

FIGURE 6.6. Big Rock Point Uncontrolled Calculated Local Power Distribution of Fuel Performance Improvement Program Test Assembly

25% IN-CHANNEL VOID

1.02	0.92	*	*	*	1.23	*	*	*	1.14	1.31
0.80	1.10	1.03	1.34	1.30	1.31	1.38	1.51	1.24	1.44	1.13
*	0.89	0.87	1.15	1.11	1.07	0.57	1.14	1.27	1.44	*
*	0.70	0.99	0.98	0.99	1.00	0.99	1.07	1.14	1.27	*
*	0.60	0.87	0.87	0.93	**	0	1.07	1.14	1.38	*
0.59	0.81	0.46	0.87	0.95	0.98	1.01	0.99	0.57	1.31	1.23
*	0.61	0.84	0.83	0.87	0	0.95	0	1.00	1.07	*
*	0.70	0.92	0.88	0.87	0.87	0.87	0.93	0.99	1.11	*
*	0.81	0.74	0.94	0.88	0.83	0.46	0.87	0.99	1.15	*
0.60	0.82	0.74	0.92	0.84	0.81	0.87	0.99	0.88	1.10	0.92
0.65	0.60	*	*	*	0.59	*	*	*	0.80	1.02

* TEST ROD LOCATIONS

Km = 1.0562

** SCR ORIENTATION ROD

FIGURE 6.7. Big Rock Point Weak Controlled Local Power Distribution of Fuel Performance Improvement Program Test Assembly

25% IN-CHANNEL VOID

1.31	1.13	*	*	*	1.23	*	*	*	0.92	1.02
1.14	1.44	1.24	1.51	1.38	1.31	1.30	1.34	1.03	1.10	0.80
*	1.49	1.24	1.44	1.27	1.14	0.57	1.07	1.11	1.15	0.88
*	1.35	1.51	1.27	1.14	1.07	0.99	1.00	0.99	0.99	0.69
*	1.28	1.38	1.14	1.07	** 0	1.01	0	0.93	0.87	0.87
1.23	1.31	0.57	0.99	1.01	0.98	0.95	0.87	0.46	0.81	0.59
*	1.22	1.30	1.07	1.00	0	0.95	0	0.87	0.83	0.84
*	1.22	1.34	1.11	0.99	0.93	0.87	0.87	0.87	0.88	0.92
*	1.28	1.03	1.15	0.99	0.87	0.46	0.83	0.88	0.94	0.74
0.92	1.10	0.88	0.99	0.87	0.81	0.84	0.92	0.74	0.82	0.60
1.02	0.80	*	*	*	0.59	*	*	*	0.60	0.65

Km 1.0562

* TEST ROD LOCATIONS

** SCR ORIENTATION ROD

FIGURE 6.8. Big Rock Point Weak Controlled Local Power Distribution of Fuel Performance Improvement Program Test Assembly

cracking. It should be pointed out that no new models were proposed or applied in these calculations. The fuel temperatures were calculated at beginning-of-life conditions and exposures up to and including 500 equivalent full-power days.

The results indicate that the annular pellet design is licensable even if overly conservative assumptions of cracking and swelling are made. Fuel temperatures in excess of fuel melting are calculated for the packed-particle fuel under the same set of assumptions.

The results of the packed-particle fuel temperature calculations, performed using reasonably conservative assumptions for densification and cracking, indicate that the packed-particle design should also be licensable.

Reference Fuel. The reference fuel for this program is defined as the standard Reload G-4 fuel and will be defined in a formal program design report to be issued. The fuel temperatures were calculated using an NRC-approved model.

The maximum fuel temperature and volumetric average temperature were calculated as a function of the fuel rod linear heat generation rate (LHGR) for beginning-of-life conditions. These temperatures represent the highest calculated fuel temperatures as a function of exposure. The results of the calculations showed no fuel melting for linear heat generation rates as high as 59.0 kW/m (18.0 kW/ft), which exceeds the 122% overpower peak LHGR of 44.3 kW/m (13.5 kW/ft).

Annular Fuel. The maximum and volumetric average fuel temperatures were calculated for the annular fuel pellet design using an NRC approved computer code. Several sets of calculations were performed to estimate the effects of pellet cracking and swelling on the fuel temperatures.

Incomplete information was available to formulate models for gas release, densification, cracking and swelling. However, information was available from previous annular fuel test programs for mixed-oxide fuel in the BRP reactor which showed that overall performance of the mixed-oxide rods to burnups of 30,000 MWd/t was the same as that previously observed for solid UO_2 pellet fuel. Because the pellet characteristics as listed above affect fuel temperatures at exposures greater than zero, it can be assumed that no significant differences exist between mixed-oxide and UO_2 for burnups approaching 30,000 MWd/t. Therefore, the base-case, fuel-temperature calculations for the annular fuel were performed in the same manner as for the reference fuel design. The results of the base-case calculations indicate no fuel melting for rod LHGR values up to 59.0 kW/m (18.0 kW/ft).

Conservative estimates of fuel temperatures were determined to account for the uncertainty in the fuel densification model for the annular design. This was accomplished by assuming:

- 1) instant fuel densification, and
- 2) instant fuel-column slumping to the extent of completely filling the pellet central void region.

For rod powers near the overpower maximum of LHGR 44.3 kW/m (13.5 kW/ft), the results indicate increases of 13% and 15% for the maximum and average fuel temperatures, respectively, when compared against the base-case temperature results. When compared against the reference fuel design temperatures, increases of 3% to 8% are realized for the maximum and average fuel temperatures, respectively. In addition, the results of the conservative calculations for the maximum fuel temperature indicate no fuel melting for LHGR values up to 59.0 kW/m (18.0 kW/ft). In addition to the calculations described above

sensitivity studies were performed to determine the effects of pellet cracking and swelling on the fuel temperatures. Fuel temperature calculations were performed as a function of pellet exposure for two cases:

- 1) no pellet cracking, and
- 2) no pellet cracking and swelling.

As can be expected, the maximum fuel temperatures no longer occur at beginning-of-life conditions as was determined for the base-case calculations. The peak temperatures occur sometime in life (~ 400 full-power days) since no gap closure due to cracking and swelling is permitted. Due to these overly-conservative assumptions, fuel melting is realized for some LHGR value less than 59.0 kW/m (18.0 kW/ft) for exposures as low as 50 full-power days. In determining the occurrence of fuel melting, the exposure dependence of the UO_2 melting temperature was taken into account. The results of the no-cracking and no-swelling calculations indicate a peak LHGR greater than the overpower peak for exposures less than 300 full-power days; the peak LHGR decreases almost linearly to 42.0 kW/m (12.8 kW/ft) at 500 full-power days. The results of the calculations for both cases indicate peak LHGR values greater than the peak 100% power value 36.4 kW/m (11.1 kW/ft).

The results of the preliminary calculations for the annular pellet design indicate that the design may be licensed without restricting the plant operation even with the overly conservative assumptions of no pellet cracking and swelling.

Packed-Particle Fuel. The maximum and volumetric average fuel temperatures were calculated for the packed-particle fuel design using an NRC-approved computer code. Three sets of calculations were performed to determine the effects of fuel densification and cracking. The calculations for the three cases separately assumed:

- 1) the NRC-approved model was applicable to vipac fuel,
- 2) no credit for fuel cracking was allowed, and
- 3) the fuel instantly densified and credit was allowed for fuel cracking.

Incomplete information was available to formulate models for densification, cracking and swelling. However, an estimate of the packed-particle fuel thermal conductivity was provided for beginning-of-life and early irradiation conditions.

For Case 1, as defined above, the calculated results indicate no fuel melting for peak LHGR values up to 49.2 kW/m (15.0 kW/ft). At higher LHGR values, the maximum temperature exceeds fuel melting and yields high fuel temperatures. This is probably due to the uncertainty of molten fuel material properties. The results for cases 2 and 3 yield similar results for the fuel design. It should be noted that, due to the low initial density of the fuel column, all fuel temperatures increased monotonically with irradiation. Thus, the highest calculated fuel temperatures were obtained at 500 equivalent full-power days. It is therefore anticipated that peak fuel rod LHGR values will be required in licensing the fuel to exclude fuel melting as a function of exposure.

If assumptions were made to exclude cracking and/or swelling, peak LHGR values as low as 23 kW/m (7 kW/ft) are required to preclude fuel melting for the range of exposures considered. These overly-conservative assumptions may not impact the licensability of the vipac fuel because: 1) the decrease in fuel assembly reactivity will reduce the assembly (rod) LHGR, and 2) the peak rod LHGR values exceed the anticipated current and future cycle MAPLHGR limits for the reload fuel.

6.2.3 Mechanical Engineering

The design characteristics for the four rod types selected for Series 1 full-length demonstration rods in BRP batch G-4 are listed in Table 6.3. Four rod designs have also been selected for Series 1 segmented rods to go into BRP reload batch G-4. These consist of three-segment rods including: 1) rods with reference fuel in the upper and middle segment and reference fuel with coated cladding in the lower segment, 2) rods with reference fuel in the upper segment and annular fuel in the middle and lower segment, and 3) rods with reference fuel in the upper segment and annular fuel with coated cladding in the middle and lower segment, and 4) rods with reference fuel in the upper segment and packed-powder fuel in the middle and lower segment. Design characteristics are given in Table 6.4.

6.3 Fuel Process Development

6.3.1 Annular Pellet Process Development

The die-and-punch design was completed for both Halden fuel pins and BRP demonstration rods. Delivery of the tooling is scheduled for early October.

6.3.2 Coated-Cladding Process Development

Laboratory work continued in support of coated-cladding development. Methods of application and controlled film thickness are being developed. The main emphasis has been on developing outgassing parameters for the coating. Processing conditions are being studied in an attempt to achieve 10 ppm moisture equivalents in both fuel and coating.

6.3.3 Vipac Fuel Process Development

The results of the experiments with an active urania powder in an attempt to produce high-density feed stock resulted in a decision to use the packed-particle process, and several

TABLE 6.3. Characteristics of DOE FPIP, Series 1
Full-Length Demonstration Fuel Rods
for Big Rock Point Reactor

Reference Fuel

Type	Solid Pellet
Material	UO ₂
Density, % Theoretical	93.5 $\pm$ 1.5
Diameter, cm (in)	0.9436 $\pm$ 0.0013 (0.3715 $\pm$ 0.0005)
Length, cm (in)	0.762 $\pm$ 0.064 (0.300 $\pm$ 0.025)
Dish	Both ends
Dish Volume, %	2.0 $\pm$ 0.5
Enrichment, wt% ²³⁵ U	2.52 $\pm$ 0.05

Annular Fuel

Type	Annular Pellet
Material	UO ₂
Density, % Theoretical	93.5 $\pm$ 1.5
Diameter, OD, cm (in)	0.9436 $\pm$ 0.0013 (0.3715 $\pm$ 0.0005)
ID, cm (in)	0.2985 $\pm$ 0.013 (0.1175 $\pm$ 0.005)
Length, cm (in)	0.762 $\pm$ 0.064 (0.300 $\pm$ 0.025)
Dish	None
Enrichment, wt% ²³⁵ U	2.69 $\pm$ 0.05

Packed-Powder Fuel

Type	Vibratory compacted particles
Material	UO ₂
Density, particles, % theoretical	>98
Diameter, cladding ID, cm (in)	0.9677 $\pm$ 0.0038 (0.3810 $\pm$ 0.0015)**
Enrichment, wt% ²³⁵ U	2.52 $\pm$ 0.05
Smear Density, % Theoretical	87.0 $\pm$ 1.0

TABLE 6.3. (Continued)

Insulator Disks

Material	Alumina
Density, g/cc	3.9
Diameter, cm (in)	0.919 (0.362)
Length, cm (in)	0.508 (0.200)
Number, location	2, one at each end of fuel column

Plenum Spring

Material	Inconel X-750
Spring force, nominal, Kg (lb)	5.0 (11)
Spring constant, nominal, Kg/cm (lb/in)	2.14 (12)
Spring volume, cm ³ (in ³)	0.628 (0.0383)

Fill Gas

Material	Helium
Purity, %	99.5
Fill pressure, nominal Pa (atm)	10 ⁵ (1)

Cladding

Material	Zircaloy-2
Condition	CW-Stress relieved
OD, cm (in)	1.15 $\pm$ 0.0038 (0.451 $\pm$ 0.0015*)
ID, cm (in)	0.968 $\pm$ 0.0038 (0.381 $\pm$ 0.0015**)
Wall thickness, minimum, cm (in)	0.0831 (0.0327)
Length, m (in)	1.9022 $\pm$ 0.00038 (74.889 $\pm$ 0.015)
Allowable bow, cm/m (in/ft)	0.125 (0.015)
Coating (present in some rods with annular fuel)	Dag 154
Coating thickness, mm (mil)	0.0635 $\pm$.0013 (0.25 $\pm$ 0.5)

TABLE 6.3. (Continued)

Fuel Rod

OD, cm (in)	1.140 ± 0.0051 ($0.449 \pm 0.002^{***}$)
Plenum length, cm (in)	9.909 (3.901)
Fuel column length, cm (in)	177.8 ± 0.64 (70.0 ± 0.25)
Allowable bow, cm/m (in/ft)	0.125 (0.015)

*Cladding to be selected to fall within a range of ± 0.0025 cm (± 0.001 inch).

**Cladding to be selected to fall within a range of ± 0.0025 cm (± 0.001 inch) except for nominally 0.0025 cm (0.0010 inch) greater ID for cladding tubes selected for coating.

***Rods to be selectively etched to fall within a range of 0.0038 cm (± 0.0015 inch).

TABLE 6.4. Characteristics of DOE, Series 1 Segmented Fuel
Rods for Big Rock Point Reactor

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, cm (in)	0.805 (0.317)
Coil OD, at working length, cm (in)	0.864 ± 0.038 (0.340 ± 0.015)
Free length, cm (in)	5.690 ± 0.076 (2.240 ± 0.030)
Spring force, Kg (lb)	0.885 (1.95)
Spring constant, Kg/cm (lb/in)	0.370 (2.07)
Spring volume, cm ³ (in ³)	0.292 (0.0178)

TABLE 6.4 (Continued)

Cladding

Material	Same as demonstration rods
Condition	Same as demonstration rods
OD	Same as demonstration rods
ID	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	Same as demonstration rods
Length, upper segment, cm (in)	66.624 $\pm$ 0.013 (26.230 $\pm$ 0.005)
Length, mid and lower segment, cm (in)	60.018 $\pm$ 0.013 (23.629 $\pm$ 0.005)

Fuel Rod

Number of segments	3
OD	Same as demonstration rods
Allowable bow	Same as demonstration rods
Plenum length, upper plenum	Same as demonstration rods
Plenum length, mid and lower plenum, cm (in)	3.302 (1.300)
Fuel column length, upper segment, cm (in)	54.445 $\pm$ 0.406 (21.435 $\pm$ 0.16)
Fuel column length, middle segment, cm (in)	54.684 $\pm$ 0.203 (21.529 $\pm$ 0.08)
Fuel column length, lower segment, cm (in)	54.445 $\pm$ 0.203 (21.435 $\pm$ 0.08)
Axial distance between fuel columns, cm (in)	7.112 (2.800)
Fuel material, upper segment	Reference fuel
Fuel material, mid and lower segment	Reference fuel Annual fuel Pack-powder fuel

compactions were made. The resulting densities were all above 98% TD. Based on these results the packed-particle process was selected as the reference process to produce the high-density feed material for vipac fuel.

Design was started to install a vipac loading station in the ENC process development laboratory. Installation is scheduled for completion in mid-December.

In the interim period, a temporary loading station will be utilized to test compaction densities and techniques.

6.3.4 Materials

The cladding for the Halden test pins was hand sorted from commercial tube lot. Fifty tubes were selected based upon the analyses of the ultrasonic ID, OD wall thickness chart traces. The tubes were segregated into 2 categories:

- 1) tube ID, cm (in) 1.237 $\pm$ 0.0025 (0.487 $\pm$ 0.001), and
2) tube OD, cm (in) 1.237 $\pm$ 0.00125 (0.487 $\pm$ 0.0005).

The 1.237 $\pm$ 0.00125 (0.487 $\pm$ 0.0005 tubes will be used for the fuel pins. The others for process development.

The uranium requirement for the Halden fuel pins and the BRP demonstration rods was specified.

Halden. Fuel pins - 11% ^{235}U (45 kg UO_2)
Process development - depleted UO_2 (25 kg UO_2)

BRP Demonstration Rods.

Reference and packed-particle rods	- 2.52%	^{235}U
		(50 kg U as UF_6)
Annular rods (including segmented)	- 2.69%	^{235}U
		(156 kg U as UF_6)
Replacement rods	- 2.52%	^{235}U
		(164 kg U as UF_6)

Delivery of the 11% enriched UO_2 is scheduled for January 1978.

7.0 TASK 3 - FUEL PERFORMANCE EVALUATION

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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 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 behavior comparisons. A master list of preirradiation information requirements for the Task 2B screening test (Table 7.1) was compiled showing the measurement precision to be employed, the status of technique development, and the justification or use to which the data will be put.

Careful attention to an evaluation of the conditions to which the experiments are exposed in the reactor during irradiation is equally important. Reliable information of this type is essential in order to develop the technical bases to support the required predictive fuel performance modeling tools. Activities were initiated to define data requirements to ensure that they are adequately addressed in developing test plans and designs for Tasks 2A, 2B, and 2C.

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. Examination of incipient failure regions in the cladding will provide an improved understanding of the cladding failure mechanisms of

concern. Examination of failed cladding regions will also be performed to elucidate in-reactor failure mechanisms for comparison with out-of-reactor laboratory studies.

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.

A tape containing the data base information used for the FRAP-S Code was obtained. The tape consists of two files: the first file contains information on 352 rods catalogued in a format used by EG&G; the second file contains the same information but is organized in a modified format. The tape was accessed and the information on both files was outputted. A copy of the information from both files is being used to evaluate data requirements. The data and format are also expected to be useful in the development of the performance evaluation function.

Preliminary assessment was made of feasible poolside and hot-cell post-irradiation examination (PIE) studies.

**TABLE 7.1. Preliminary Preirradiation Characterization Requirements
for FPIP Task 2B Fuel Rods (Screening Tests in Halden)
Revision 2, October 28, 1977**

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
1. Rod Identification	Mark on end plug.	Inspection records provided on hard copy.	Accepted. Circumferential grooves will be provided in plugs for axial location markers for axial measurements. A single axial scribe mark on lower end plug to be provided for rotational orientation.	A unique i.d. must be provided for each rod built in order to maintain traceability, prevent mixup of reject rods, for control and proper placement of rods during irradiation and for PIE. Would be preferable to engrave or stamp identification, as opposed to vibratool. (Crud may obliterate vibratool).
2. Rod Type	Refer to as Reference, Annular, vipac, or Annular-coated.	Provide inspection record relating type of rod to identification number on end plug.	Accepted	Traceability
3. Rod a. Length	Precision ± 0.002 in. corrected to 68°F	Provide data sheet.	Accepted	The rod length must be measured accurately in order to determine axial strain during PIE. Measurement must be corrected for thermal expansion (e.g. 68° versus 95°F would cause almost 1/2 mil difference). 0.002 accuracy is an upper limit. If can be measured better would be helpful.
b. Weight	Precision $\pm 0.1g$	Provide data sheet.	Accepted	Measurement of weight to high precision is often useful in determining if a rod has failed in service. Small weight change may be due to absorption of water or loss of fission products. Obtaining total rod weight by summing components accumulates too much uncertainty.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
3. Rod (Continued) c. Diameter	<p>Backup diametral measurements will be taken to support PIE measurements.</p> <p>Preirradiation: Precision ± 0.001 in. 0° and 90° starting at lower end cap circumference groove. A profilometer yielding a continuous axial scan will be used. Most of the rods will have bulky instrumentation leads coming from the end plugs.</p> <p>Postirradiation: Nominal axial profilometer traces, 45° apart, axial, circumferential reference on rod, precision ± 0.0001 in.</p>	Provide data sheet.	<p>Use of eddy current technique in PIE may be hampered if weld overhang > 5 mil on radius. Won't change weld process, but will consider machining off weld overhang if greater than 5 mil prior to PIE.</p> <p>Accepted. Commercial practice on weld overhang i.e., ring gages.</p>	As-fabricated rod diameters are required to determine diametral strain and ridding during PIE. Strains are often on the order of 0.001 in. Hence, much greater accuracy is required in preirradiation measurement in order to obtain meaningful PIE determination of strain.
d. Leak Check	$\leq 10^{-8}$ std. atm cc/sec criterion	Provide statement that criterion was met.	Accepted. Equipment to be set as near 10^{-9} as possible. Instrument leads sealed during leak test.	Integrity check: Objective to minimize chance of manufacturing defects.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
3. Rod (Continued)				
e. Component Locations	Three full length radiographs, 60° apart (overlapping segments). Include in the X rays a standard in the form of a stepped bar with dimensions measured to ± 0.001 in. The 2 to 3 steps should bracket the pellet and rod diameters. The standard will assist in compensating for distortion of the X ray. Resolution capability should be sufficient to detect a large, out-of-place chip of fuel, approximately 30 mil thick (viewed edgewise), and roughly 150 mil in the other dimensions.	Provide radiographs.	Develop appropriate technique. Standard to contain step diameters of 0.40, 0.45, 0.45 in. Step length approximately 2 in. Material: steel.	The location of the rod components, axial and radial gaps, location of chips, etc., must be known prior to irradiation in order to compare with PIE results. Chips in gaps could cause additional PCI.
f. Fuel Isotopic (As-loaded)	Nondestructive assay technique should be able to verify correct position of the two enrichments in each rod. Also want to document any vipac axial density variation as well as show low enrichment guard pellets at ends of vipac column.	Provide charts.	Present neutron activation technique not adequate. Use of neutron radiographic facility to be investigated.	The rods contain two different enrichments. Misplacement of the two types could cause power peaking and/or unusual stress conditions in the cladding. Therefore, location must be verified. Axial variation in vipac density must be known to extrapolate to operating powers. Neutron radiography would show internal pellet defects and location of thermocouples, neither of which are visible in X rays.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
3. Rod (Continued)				
g. Weld Conditions	Radiograph thickness less voids, inclusions. Lack of penetration $\geq 80\%$ of wall thickness.	Provide radiographs.	Accepted	Weld Integrity
h. Instrumentation	Verify instrumented rods in working order. Check for gross shipping damage to end plugs prior to assembly and electrical continuity before and after welding into rods.	Provide calibration data and X rays of end plugs taken prior to assembly.	Accepted. Procedure to be provided when available from Halden.	Tests largely depend on in-pile instruments.
i. Fill gas	Certify $>99.9\%$ He.	Provide data sheet containing gas analysis.	Neutral pressure across gloves in box acceptable proof of ± 1 psi at 1 atm.	Determine fission gas release. Helium must be pure to prevent degradation of gas conductivity. The amount loaded must be known in order to model F.G. release and gas conductivity. ± 1 psi uncertainty is $\sim \pm 7\%$ for 1 atm, which would be acceptable.
j. Loading procedure	General description of technique.	Describe	Accepted	Required for summary reports. Need assessment of effects of loading on lubrication integrity, likelihood of non-He gas contamination, likelihood of H_2O contamination or steps contained to prevent contamination. Concern is greatest for vipac.
k. Surface treatment (Exterior)	General description: cleaning, abrading, etching, autoclaving.	Describe	Instrumentation during surface treatment.	Part of standard procedure to control corrosion resistance of cladding.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
4. Cladding				
a. Purchase Order	Reference	On file.	Accepted	Traceability
b. Fabricator Certs	Records of ingot, trex, tube, as available in commercial practice	Provide copies of records.	Accepted	Traceability and characterization of cladding properties for modeling.
c. Overcheck Results	As available in commercial practice	Provide data sheets.	Accepted	Same as 4.b.
d. Surface Treatment	Describe and characterize result, not only routine treatment of inner and outer surfaces, but also special preparation in vicinity of welds, especially in conjunction with coated cladding.	Provide description.	Accepted	Need to characterize and trace surface treatment, especially with lubricated cladding, in case of clad failures that may be related to surface conditions.
e. Actual tube Identity	Tubes shall be kept in individual, clean plastic bags marked with identification.	Provide description of control procedure.	Accepted. Develop as needed.	Traceability to lot and archive sample.
f. Actual Dimensions	ID, OD, wall to precision ± 0.0002 inc. Maintain identity and location of section of original tube.	Provide data sheet or charts, as available.	UT traces giving ID, OD, wall will be retained after extracting average, minimum and maximum dimensions. (Whole tube, not 20 in. section.) Test specified to detect spiraling of minimum wall. Complete report in preparation.	Pre-measurement and PIE will be used to determine strain in cladding and ridging. See comments on pellet dimensions. ID will be used to determine gap. Wall thickness variations will be used in conjunction with initial ovality correlations. Minimum wall spiraling is related to rod bow.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
4. Cladding (Continued)				
g. Coating	Summary of process procedure. Measure average thickness of coating on each cladding. Characterize thickness variability on samples. Characterize particle orientation and coating density. Characterize defects before and after landing. Chemically analyze processed coating for retained oxygen and hydrogen sources.		(Typical leave (g) blank)	Characterization
5. Fuel (both enrichments)				
a. Material Purchase Order	Reference	On file	Accepted	Traceability
b. Processing Procedure	General description of technique	Describe	Accepted	Required for reports. Need information on contamination from unwanted trace elements, e.g., stainless steel ball milling, copper screens, trapped gases, et
c. "Label" each pellet so its location in the stack is known.	Must follow behavior of each pellet.	Provide record showing identity of each pellet and its location in each rod.	Accepted. Traceability	
d. Dimensions	Each pellet, diameter to precision ± 0.0002 in. (set equipment to ± 0.0001 in.), length to precision ± 0.0001 in. Record end out-of-squareness if greater than 0.002 inc. Provide nominal values all other	Provide data sheet.	Commercial practice is to accept pellet diameters to a wider tolerance than the measurement precision required for this program. Consequently, a special measuring setup will be used. The choice is between an optical scanner or a micrometer. Surface roughness and lobing of the	Combination of pellet diameter and cladding ID determine gap which in time effect cladding strain, ridging, fuel temperature fission-gas release, etc. Must be known accurately to model. 1 mil variations in gap can have significant effects. Hence, must know variability (in dimensions)

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
5. Fuel (Continued) d. Dimensions (Continued)	<p>dimensions, such as chamfer, dish, land, taper, as applicable. Demonstrate, if possible, center hole ID of annular pellet (and hole taper) not to vary ≥ 0.0002 in. from nominal. If this criteria is met, report nominal dimension. For all pellet types: Demonstrate, if possible, lobing not a concern. Criteria: difference between min. and max. Less than 0.0002 in. Demonstrations will be to 95/95 on a single lot of pellets for Task 28 screening tests. Results will apply throughout program.</p> <p>Photograph vipac particles on SEM for surface condition and size distribution. No enrichment mark is necessary on annular pellets.</p>		<p>pellets may also be detected with a profilometer. Diamond wheel grinder will be used on both 2B and 2C pellets.</p>	<p>to better than 1 mil. Surface roughness, non-cylindrical shape can also have effects. Properties of shape of vipac fuel should be known to have idea of surface area, sharpness of particle surfaces etc. Pellets for tests should have same surface properties as "reference" fuel.</p>

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
e. Chemical Content	Analyze for total U, O/U ratio, N, Fe, F, Cl, H ₂ O, B, Cd, others as commercial practice dictates. Perform isotopic analysis of U for masses 234, 235, 236, 238. Results to be standard commercial precision.	Provide copies of analysis results.	Statistical sampling procedures for the unique lot and batch size of Task 2B screening test rods to be determined.	O/U ratio affects O ₂ potential, fission-product chemistry, strength properties. Must be known to model & interpret results. Trace materials in fuel can affect cladding, e.g., hydriding, (H ₂ O), grain boundary corrosion (F, Cl), general embrittlement (Fe, Si, Cd), and neutron capture (Cd, B). Must be known to prevent adverse effects and interpret results.
f. Metallographic Analysis.	Photos adequate to determine grain size, porosity size and distribution, and the presence of second phases. These will probably be 200X and 500X, as-polished and as-etched.	Provide photos with proper identification.	Same as for chemical content	Grain size affects creep properties. Porosity contains trapped gases. Porosity distribution affects in-pile densification. Second phases could contain adverse contaminants.
g. Resinter Test for Densification Properties	Commercial test results subject to sample size considerations.	Provide data sheets.	Three pellets per sinter run until engineering judgment shows one pellet per sinter run is adequate.	Test characterizes susceptibility for densification which in turn affects fuel temperature, fission-gas release, etc.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
5. Fuel (Continued)				
h. Absorbed gas and Water Vapor	Analyze for CH ₄ , Co, CO ₂ , N ₂ , O ₂ , H ₂ O, total H, and total gas. Standard commercial precision.	Provide data sheets.	Demonstrate before production runs begin that SST/rubber plug and seal is adequate for vipac, coated cladding, annular as well as standard pellets. For annular and standard, records are adequate proof. Concern is that there is minimum moisture pickup between loading and welding vipac rods.	These contaminants affect gas-gap conductivity and fuel temperature and hydriding of cladding. Low impurities needed to prevent adverse effects and exact analysis needed for modeling.
i. Batch Identification	Each rod	Reference in inspection records.	Accepted. Some rods may have more than one batch. The location in the rod must be known.	Traceability
j. Fuel Density	Immersion density each batch. Geometric density each pellet to commercial precision.	Provide data sheet.	Precision of immersion density for vipac not set. Vipac density requirements to be set.	Affects conductivity, strength, creep properties, fission product release. Needed for modeling.
k. Fuel Weight	±0.1 g each rod.	Provide data sheet.	Accepted. Measure pellet weight to 3 decimal places. Accumulated uncertainty less than ±0.1/g rod.	Required to be reported to test reactor for neutronic calculations and for transfer of SNM to foreign country.
l. Pellet Defects	Minimize defects on cylindrical surface and defects that may produce small chips. Suggested visual standards: Not accept flakes on cylindrical surface that are >5 mil deep or >15 mil extent.	Provide photos of visual standards.	To be negotiated.	Defects on cylindrical surface affects heat transfer. Chips could promote PCI. Defects on flats can affect stacking geometry. Internal defects could affect fragment relocation. Neutron radiography would be nice for internal defects.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
6. End Pellets, which included insulator pellets and powder seals			Powder seal may be sintered metal, porous, force-fit plug. Guard pellets are low enrichment pellets and Al_2O_3 insulator pellets.	
a. Dimensions	Diameter to ± 0.0005 in., length to ± 0.005 in.	Provide data sheet.	Accepted. Same level of detail as more highly enriched fuel.	Same as for fuel
b. Material	General Chemical analysis	Provide data sheet.	Accepted	Characterization
c. Absorbed gas	Typical gases and values	Provide data sheet.	Accepted	Same as for fuel
d. Density	Typical Immersion	Provide data sheet	Accepted	Same as for fuel
7. Miscellaneous				
a. Springs	Surface cleanliness Contributions to internal volume	Describe, commercial practice.	Accepted	Do not want spring to contribute to degradation.
b. End Plugs	Contribution to internal volume	Drawings	Halden will supply all end plugs. ENC will provide adapter for lower end plug.	Traceability: prevent stringers part in tube from contribute to void volume in rod. Dimensions affect weldability, assembly and fit-up test rig.
c. Instrumentation	To be provided by Halden	Descriptions and drawings	See 7b.	A lot of test information will come from instruments. Requirements will be specified later. Accumulation of fuel fragment around bellows must be prevented.
d. Welding Procedure	General description of assembly procedure	Describe	Accepted	Required for reports. Traceability to qualified procedure.

TABLE 7.1. (Continued)

CHARACTERISTIC	REQUIREMENT	FORM OF INFORMATION	STATUS	JUSTIFICATION
8. Archives				
a. Cladding	One 12-in. sample from each tube used for fuel rods	Retain	Develop plan for retention of all archive samples.	May be required for PIE comparisons.
b. Fuel	Five pellets from each batch or 40 g vipac fuel	Retain	See 8a.	May be required for PIE comparisons.
c. End pellets	5 end pellets from each pellet lot	Retain	See 8a.	May be required for PIE comparisons
d. End Plug Material	6 in. of bar stock from each lot	Retain	See 8a. Also, one or two upper instrumented end plugs will be retained.	May be required for PIE comparisons.
e. Springs	6 from each wire lot	Retain	See 8a.	May be required for PIE comparisons.
9. Drawings				
Assembly and details	Complete information on dimensions, specifications, procedures should be available as needed on each rod type.	Provide drawings	Accepted	Must supply drawings to test reactor. Other information to assist in PIE interpretation.

8.0 TASK 4 - ESTABLISHMENT OF TECHNICAL BASES

J. F. Patterson and K. R. Merckx (Exxon Nuclear Company)

The objective of Task 4 is to interpret program results to facilitate commercial application by establishing technical bases for design and design criteria. Technical bases for design provide the foundation and best state-of-the-art for product design. They include technology which has been established through research and development studies from operating experience and from licensing requirements. They also include the analytical methods, behavior models, and computer codes which have been accepted for design use.

The establishment of technical bases in this program is concerned with the development of an appropriate data base and selection of analytical procedures which are applicable for use in the design of fuel with improved power cycling capability, and assessment of performance in a commercial power reactor.

8.1 Technical Bases

One of the main program objectives is to establish technical bases for the fuel types which have improved power cycling capability. Much of the data required for establishing these bases will be collected and generated during the program; hence, complete technical bases for the candidate fuel concepts can not be expected during the initial phases of the program. The initial activity in generating technical bases was to write a set of bases for the reference fuel based on existing LWR technology for oxide pellet fuel with zircaloy cladding. These bases have been restricted to the fuel rod and general criteria required to establish the fuel rod design and fabrication. Particular attention is devoted to the thermal-mechanical interaction bases that are associated with the main objectives of this program. The organization of technical bases has been arranged so that treatment of the various candidate

fuel concepts can be included by adding additional sections. The initial draft version of the resultant bases for the reference fuel was completed and circulated for internal review.

Early formulation of technical bases is helping to focus on data, performance models, and design and fabrication criteria requirements for the candidate fuel concepts. The initial technical bases were written using the FRAP-S performance model and MATPRO material property package as representative of existing LWR-pellet fuel technology. This selection was made because the reference material is in the public domain and documentation is available. Fabrication standards were selected wherever feasible on established standards so that the initial bases have general industry acceptance. The existing Exxon Nuclear Company (ENC) was reformulated to conform to the bases FPIP Technical Bases package. This exercise has brought attention to special considerations that occur when adapting general technology to integrated bases for specific products.

8.2 Data Requirements

Two types of data requirements are being addressed in establishing the technical bases. One data source is that which has been generated by past and ongoing programs conducted throughout the technical community. The second data source is the specific data developed within the program to extend the data base in areas of concept performance especially as related to fuel-cladding interaction.

8.2.1 General Data Sources

Initial efforts to establish technical bases for the candidate fuel concepts pointed out the need for a directed review of the literature associated with the technical bases requirements. A general file system was developed that will be compatible with computer search techniques, rapid updating, and flexibility for refinement. Initial entries have been made

into this file as part of establishing technical bases for the program-selected concepts. File extension will be accomplished in conjunction with establishing the detailed technical bases. A specific type of data requirement that plays a major role in bases for performance prediction is a file of measured performance of fuel rods under well-defined fabrication and operational conditions. The FRAP-S verification file was activated and its data storage format altered to provide ease of including the specialized data that will result from tests of the candidate fuel concepts. A file of collected data has been initiated with some initial fuel rod data for packed-powder and annular pellet fuels. Collection of initial data has required that the extended flexibility of selected file format be utilized.

8.2.2 Program-Generated Data

Although general data requirements are being treated by Task 3, special attention has been given the generation of detailed operational histories by Task 4. Such records are required for code verification which is central in establishing performance data bases. Existing interpretational methods for fuel management core following calculations have been adapted for the control rod configurations encountered in Big Rock Point (BRP). The additional requirements of detailed power history for a fuel rod have been incorporated into the sequence of file treatment programs. This extended capability will provide a bundle history card image record along with the bounding control rod data. This data file is required to generate fuel rod local power generations when the appropriate local power factors are input into an interpretation program. A sample calculation for one BRP fuel irradiation cycle was performed using the modified programs including the interpretation program.

8.3 Fuel Performance Code

The proposed technical bases specify deformation and stress design criteria and the calculational methods and property models are to evaluate these criteria. Initial FPIP technical bases have assumed that the FRAP-S performance model and MATPRO property models will be the starting point for developing the bases. The FRAP-S code has been activated and the code test problem has been run. Documentation of FRAP-S and MATPRO has been related to the technical bases. Implementation of a program-based performance code is still in its formative stage.

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