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Fuel Performance Improvement Program

Quarterly Progress Report April - June, 1977

Consumers
Power
Company

EXXON NUCLEAR
COMPANY, Inc.

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FUEL PERFORMANCE IMPROVEMENT PROGRAM

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Quarterly Progress Report
April - June, 1977

Compiled by D. S. Rowe

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July 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 is being compared to a reference fuel of contemporary pellet design through test reactor experiments. Initial screening tests are being planned for the Halden Boiling Water Reactor. Fuel design and process development has started 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 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. While the program was described as part of the Electric Power Research Institute (EPRI) contractors' meeting in Dallas, Texas, April, 1977, this quarterly report is the first written presentation of the program. 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 the operation of both PWR's and BWR's.

- 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. 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 the nuclear fuel industry 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 cannot be too large 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 identical to 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. 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 reactor experiments are subjected to 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 (FCI) STATE-OF-THE TECHNOLOGY AND POTENTIAL IMPROVED CONCEPTS

W. J. Bailey (BNW)

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 performance capabilities. Two activities were initiated and are currently underway. One involves a state-of-the-technology review of fuel-cladding interaction. The other activity is concerned with the qualitative assessment of the relative merits of fuel concepts. The status of these two activities is described below.

2.1 Literature Survey

A literature search for information on postulated fuel-cladding interaction mechanisms and associated supportive data was initiated and completed. Over 100 reports and papers plus several patents have been reviewed. The reports and papers were examined and the findings summarized. Some of the references deal with on-going programs, e.g., EPRI, Canadian, Halden, Studsvik, that are closely related to this program. A draft of the state-of-the-technology report^(2.1) is currently being prepared and is expected to be issued late in 1977.

2.2 Fuel Concept Assessment

Efforts to identify fuel concepts that have potential for improved performance capability and to perform a qualitative assessment of the relative merits of those concepts were initiated and are continuing. The list and descriptions of candidate fuel concepts are being developed together with the criteria for selection and evaluation of concepts. The initial effort was devoted to developing criteria that address the defect mechanisms(s).

Preliminary results of Task 1 have identified the following fuel concepts for initial consideration in the program: 1) annular-pellets; 2) cladding internally coated with graphite; and, 3) packed-particle fuels.

These concepts have been identified on the basis of: 1) there is an adequate existing technology base for implementation; 2) they do not

represent a significant departure from current commercial technology; and 3) they are not included as primary concepts for evaluation in other FCI programs. It is expected that other concepts will be identified as the program progresses.

A draft of the assesement report^(2.2) is currently being prepared and is expected to be issued late in 1977.

References

- 2.1 W. J. Bailey, C. L. Wilson, L. J. MacGowan and P. J. Pankaskie, "State-of-the-Technology Review of Fuel-Cladding Interaction", COO-4066-2. (To be published)
- 2.2 W. J. Bailey, J. O. Barner, and P. J. Pankaskie, "Assessment of Fuel Concepts," COO-4066-3. (To be published)

3.0 TASK 2 - TESTING AND DEMONSTRATION PROGRAM

M. D. Freshley (BNW)

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 fuel concepts with improved performance capabilities.

Initial 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 Studies, Task 2B - In-Reactor Experiments, and Task 2C - In-Reactor Demonstrations.

The activities in progress in Task 2A are concentrating 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 are involved in evaluating graphite coated cladding and the characteristics of packed-particle fuel.

Activities in Task 2B have concentrated 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 power cycling highly instrumented test rods in the HBWR followed by power ramping of individual rods in the HBWR ramping rig. For the primary tests, rods will be irradiated to significant burnups in the BRP reactor followed by power 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 are concentrating 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 overpower ramp testing following their irradiation in BRP.

3.2 Test Program Criteria

The objective of the testing and demonstration program is to conduct a comprehensive, well balanced program to provide the data needed for establishing the technical design bases for fuel concepts with improved power ramping capabilities. In order to achieve this objective, several criteria have been established pertaining to the basic precepts of the program:

- The testing and demonstration program is to be remedy-oriented.
- The testing and demonstration is to evaluate fuel remedies which have the potential to operate within the technical specification limits of the nuclear steam supply system.*
- The testing and demonstration program is to emphasize those concepts that offer promise of significant improvement.
- The various testing and demonstration program activities are to be mutually supportive and complementary.
- The testing and demonstration program is to use a reference pellet fuel design to help define the degree of improvement of selected concepts.
- The testing and demonstration program is to provide data needed to facilitate commercial application of the most promising fuel concept(s).

* Both PWR and BWR operating conditions are of interest. Local rod power ramps due to BWR rod notching is beyond the scope of the program.

3.3 Primary Fuel Concept Selection

Annular pellet, cladding internally coated with graphite, and packed-particle 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 Power Improvement Program. Selection of these concepts for the first phase of the evaluations is based on: 1) preliminary results from the Task 1 - state-of-the-technology evaluation; 2) they do not represent a significant departure from current commercial technology; 3) there is an adequate data base to permit ease of application; and 4) the data base indicates that the concepts have potential for improved FCI behavior. It is expected that the state-of-the-technology evaluation will identify other fuel concepts that may be evaluated as the program progresses. Another consideration is to combine features of the primary concepts, e.g., annular pellets with coated cladding, to take advantage of possible synergistic effects. The design characteristics selected for the primary fuel concepts are summarized in Table 3.1.

Supporting information was reviewed in response to key design considerations related to the conceptual designs. The supporting information needed for process development and detailed design for annular pellet fuel included the basis for the size of the annular hole, the performance experience of annular pellet fuel, and the performance characteristics of annular pellet fuel such as cracking, chipping, swelling, fission product migration (radially and axially), fuel relocation or spalling of the I.D. surface, fuel relocation, fission gas release, and fuel temperature comparisons. For packed-particle fuel, the evaluation included fabrication experience, thermal conductivity, gap conductance, fission gas release, internal rod pressure, and fuel relocation. Supporting information for graphite coated cladding is being developed from laboratory experiments in Task 2A.

3.4 Reference Fuel Design

The relative power ramp performance of the primary fuel concepts are to be compared with the performance of a standard or "reference" fuel design. As recommended by the Steering Committee and agreed to by all parties concerned, the conceptual design for the reference fuel is the same as the ENC standard design. It was further decided to use the ENC 11x11 design for the BRP reactor as the reference fuel design.

The reference design thus meets the following criteria:

- It is relatable to contemporary commercial fuel designs.
- It is relatable to operating experience with power reactor fuels.
- It is amenable to fabrication by conventional commercial processes.

TABLE 3.1

Conceptual Design Characteristics of the
Primary Fuel Concepts

<u>Fuel</u>	Reference	Annular	Coated- Cladding	Packed- Particle
Form	Pellet	Pellet	Pellet	Particles ⁽¹⁾
Diameter, mm (in.)	9.44 (0.3715)	9.44 (0.3715)	9.44 (0.3715) ⁽²⁾	--
Length, mm (in.)	7.62 (0.300)	7.62 (0.300)	7.62 (0.300)	--
L/D ratio	0.81	0.81	0.81	--
Density (% TD)	93.5	93.5	93.5	≥ 98 ⁽³⁾
Dish	Yes (2%)	No	Yes (2%)	--
Chamfer	No	No	No	--
Annulus dia., mm (in.)	--	3.00 (0.118)	--	--
Technical requirements	Std. ENC	Std. ENC	Std. ENC	Std. ENC

TABLE 3.1 (Continued)

Conceptual Design Characteristics of the
Primary Fuel Concepts

	Reference	Annular	Coated- Cladding	Packed- Particle
<u>Cladding</u>				
Material	Zr-2	Zr-2	Zr-2	Zr-2
Treatment	CW/SR	CW/SR	CW/SR	CW/SR
O.D., mm (in.)	11.41 (0.449)	11.41 (0.449)	11.41 (0.449)	11.41 (0.449)
I.D., mm (in.)	9.68 (0.381)	9.68 (0.381)	9.68 (0.381)	9.68 (0.381)
Wall Thickness, mm (in.)	0.864 (0.034)	0.864 (0.034)	0.864 (0.034)	0.864 (0.034)
R/T ratio	6.60	6.60	6.60	6.60
Technical requirements	Std. ENC	Std. ENC	Std. ENC	Std. ENC

TABLE 3.1 (Continued)

Conceptual Design Characteristics of the
Primary Fuel Concepts

	Reference	Annular	Coated- Cladding	Packed- Particle
<u>Fuel Rod</u>				
Diametral gap, mm (in.)	0.241 (0.0095)	0.241 (0.0095)	0.241 (0.0095)	--
G/D ratio	0.0249	0.0249	0.0249	--
Plenum length, mm (in.)	99.1 (3.90)	99.1 (3.90)	99.1 (3.90)	99.1 (3.90)
Plenum Spring	Inconel X-750	Inconel X-750	Inconel X-750	Inconel X-750
Insulator pellets	Alumina	Alumina	Alumina	Alumina
Fill gas	Helium	Helium	Helium	Helium
Internal pres.	1 atm.	1 atm.	1 atm.	1 atm.
Active fuel length ⁽⁴⁾ , m (in.)	1.778 (70.0)	1.778 (70.0)	1.778 (70.0)	1.778 (70.0)
Technical requirements	Std. ENC	Std. ENC	Std. ENC	Std. ENC

- (1) Initially, sintered and crushed or Dynapak angular particles. Recommend three size fractions, i.e., 50 to 65%, +30 mesh; 10 to 35%, -30 + 100 mesh; and 15 to 30%, -100 mesh. BNW experience was 67.5%, -6 + 20 mesh; 12.5%, -35 + 100 mesh; and 20%, -200 mesh. The precise distribution and particle size need to achieve maximum density vary depending on the type of feed material, type of vibratory, and fuel rod geometry.
- (2) Pellet diameter may have to be adjusted smaller to accommodate graphite coating thickness.
- (3) Smear density >86% TD.
- (4) Selected rods of each type to be segmented to facilitate ramping in test reactors.

4.0 TASK 2A - OUT-OF-REACTOR STUDIES

P. E. Hart (BNW)

The purpose of this task is to conduct applications-oriented out-of-reactor studies to support the implementation and demonstration of the fuel concepts which will minimize the adverse effects of FCI. During fiscal year 1977, major emphasis is being placed on the development of methods to apply a graphite coating on the inside of the cladding and to develop additional data on packed-particle fuel concepts which will be evaluated in the first irradiations.

4.1 Coated-Cladding Studies

Graphite coated on the inside of the cladding has been selected as a fuel concept for evaluation in this program. Graphite coatings have been used in avoiding defects associated with power ramps.^(4.1) Other coating materials will be considered for future evaluation.

Evaluation of three candidate graphite dags supplied by Acheson Colloid of Port Huron, Michigan was initiated. Spectrochemical analysis results (Table 4.1) indicate that two of the dags contain less than 1000 ppm cation impurities, which are principally Ca, Fe, and Na; while the other is clearly a sodium silicate bonded system, which contains about 15,000 ppm Si and 3000 ppm Na. Initial moisture analyses at 800°C indicated that greater than 1% by weight of water remained in bulk deposits of all dags after being heated in air at 125°C for 12 hours. Thermogravimetric studies have been initiated to define outgassing conditions required to produce moisture and volatile free coatings and to define moisture pickup during storage and handling of graphite coated cladding.

Graphite application methods are under development; the simplest means of achieving full coverage of cladding appears to be a flood coating technique, followed by heating in either a vertical or horizontal position to remove the carrier. A diagram of the flood coating method currently being used is shown in Figure 4.1. A measured quantity of dag is poured into the top of the tube and the plug gradually withdrawn leaving a coat of dag on the inner surface.

Methods for measuring coating thickness are also under evaluation. The tube geometry hampers use of the high-precision profilometers. A lower-precision ($\pm 0.08 \mu\text{m}$) optical gauge is being used for evaluating test specimens. Air gauging is also presently being considered for measuring thickness of the coatings applied to full length cladding tubes.

A number of criteria have been identified for judging the relative merits of the graphite coatings under evaluation. Among these is adherence of the coating to the cladding at room temperature and during thermal cycling. The adherence of the coatings during thermal cycling is under evaluation. A layer of each coating was applied on the inside of Zircaloy-2 tubing at a thickness of $\sim 25 \mu\text{m}$; the specimens were thermally cycled to 500°C (932°F) eleven times in gettered argon, at heating and cooling rates of $16^\circ\text{C}/\text{min}$. The two coatings containing no sodium silicate binder debonded during the cycling. The sodium silicate bonded system was strongly adherent to the cladding following the cycling. Additional tests are in progress at lower temperatures and at lower heating rates.

4.2 Packed-Particle Fuel Studies

In support of the packed-particle fuel concept, a high temperature, recording semi-microbalance system will be used to determine the moisture desorption and resorption behavior. Resulting data will be used to define the conditions required to remove moisture from fuel particles and the atmosphere control that will be required to prevent moisture pickup during powder storage, rod loading, and welding. This experimental effort will be initiated during the first month of the following quarter.

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, May 9-11, 1977, St. Charles, IL.

TABLE 4.1

A Spectrochemical Impurity Analysis for
Graphite Dags Under Investigation

Element	Dag Designation		
	Dag 154	Aquadag	RD-5
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, good to $\pm$ factor of two.

FIGURE 4.1 FLOOD COATING APPLICATION DEVICE
FOR FLOOD COATING WITH MINIMUM
QUANTITY OF DAG

5.0 TASK 2B - IN-REACTOR EXPERIMENTS

J. O. Barner (BNW)

The objective of the in-reactor parameter testing task is to conduct irradiation tests designed to study the detailed behavioral characteristics and performance limits of those fuel rod concepts which have potential for improved power ramping capabilities.

Selected 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 minimize currently imposed operating restrictions that are intended to prevent fuel-cladding interactions (FCI) will be emphasized.

The following discussion describes the status of the in-reactor parameter test planning and design activities. The discussion proceeds logically from determination of the testing criteria and general testing plan, to selection of the testing facilities, and to 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 parameter testing task were established as follows:

a) Testing Scheme Should Address the Defect Mechanism

This criteria is based upon a presumed defect mechanism. The defect mechanism is presumed to be a localized fuel-cladding mechanical interaction accentuated by fission product chemical attack. The mechanism may occur during any period after fuel-cladding contact, 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, together with fission

products previously released, may result in an accentuated propensity for stress corrosion cracking. Therefore, the testing scheme should address strain concentrations, as well as the general strain concentrations in the cladding, the propensity for stress corrosion cracking, and the differential expansion between fuel and cladding that is associated with a power increase.

b) Operating Conditions Relatable to Commercial Conditions

This does not mean that the test conditions should be exactly the same as that expected in a commercial reactor. This would be impossible, because every rod in every commercial reactor is exposed to different operating environments and histories. The test conditions should approximate or be more severe than the conditions that are anticipated to cause fuel rod failures in commercial reactors. In this way behavioral differences between possible improved concepts may be accentuated. The operating conditions of interest are the rates of power increase defined by the technical specifications of the nuclear steam supply system of both PWR's and BWR's.

c) Testing Phases Should be Comparable

The results from testing phases should be relatable and comparable within themselves and to a reference fuel design. A reference design should be included in all phases so that results can be compared 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 test should provide supportive information for data and design analysis, as well as information suitable for any

operational safety uncertainties. Supportive information is especially important to establish the technical bases for design.

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 Scheme

It may be anticipated that several possible improved concepts may be selected for inclusion in the in-reactor parameter testing task, e.g., coated cladding, annular fuel and packed-particle fuel. Most efficient utilization of the available resources for the irradiation testing of the improved concepts implies a reduction in the number of concepts as testing proceeds. Therefore, the irradiation testing scheme is separated into a screening and primary phase.

The screening phase is intended to define which concept(s) have the greatest potential for improved power ramp performance as compared to the reference design. This phase will use highly instrumented fresh rods irradiated to approximately 10 and 20 MWD/kgU. The results will help to define concepts to be included in the primary phase testing.

The primary phase testing will concentrate on defining the degree of performance improvement of a selected concept(s) as compared to the reference design. Rod segments commercially irradiated in the Big Rock Point reactor will be power ramped in a test reactor for the primary phase testing.

5.1.3 General Test Plan Characteristics - Screening Phase

The general testing criteria suggest several general screening phase test plan characteristics:

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. Burnup levels in the range from 8-20 MWD/kgU were selected.

During power cycling operation, the steady-state power levels should be maintained for periods adequate to promote a significant degree of stress equilibration, "reconditioning," between the fragmented fuel and the cladding. No more than two cycles per 10 MWD/kgU are anticipated.

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

Several reactor maneuvers can result in power increases in fuel rods, e.g., a general total reactor power increase after extended operation at a fractional 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. Selection of a typical differential power for cycling operation must be somewhat arbitrary. Selection of a very low differential power would be expected to produce insignificant results. Selection of a very high differential power would also be expected to produce insignificant results because the original power from which the cycle started would have been so low that no meaningful fuel-cladding interaction of "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 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 overpower condition after the cycling operation will also start from this level. 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, it should be high, because test reactor results typically overpredict actual results in commercial reactors. Therefore, a differential ramping power of approximately 100 percent has tentatively been selected. In order to provide adequate testing flexibility, the maximum capability for ramping differential power will be designed for 125 percent after a burnup of 20 MWD/kgU. Final selection of actual differential ramp powers will be based upon modeling analyses, cycling results, and interim examination results, and will be determined just prior to the actual ramp testing.

As mentioned previously, the primary goal of the program is to minimize restrictions on power maneuvering rates. Therefore, the peak increase rate dictated by the entire commercial reactor system would be the upper limit for power increase rates. This rate, approximately 10 percent of peak power per minute will be used during the preramp irradiation and ramp testing.

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 concepts to each other and to the reference rod design. The tested rods or groups of rods should operate under similar conditions. This might best 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 feedback of test results for evaluation and inclusion in the decision making process for subsequent experimentation is vital to achieving the objective of the test program. Because irradiation tests cover long periods of time, utilization of on-line instrumentation, interim examinations 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 figure clearly illustrates the need for information flow from earlier tests to later tests.

c) Supportive Information

The bulk of direct supportive information as functions of irradiation time, e.g., fuel centerline temperatures, cladding extension and internal rod pressures, must necessarily be generated in the screening phase. These are the only rods which can be instrumented. Indirect supportive information in the form of implied fission gas release, fuel temperature,

FIGURE 5.1: Schedule and Information Flow
Task 2B - In-Reactor Parameter Testing

cladding strain, etc., will be generated during the post-irradiation examination, and, to a lesser extent, 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 to be tested in the primary phase.

5.1.4 General Test Plan Characteristics - Primary Phase

Several of the items discussed 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 history and cycling operation, actual rod power level and direct supportive information. The general power cycling and rod powers 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 reactor. Figure 5.1 illustrates the primary phase scheduling and information flow. The source of the rods for the primary phase testing is from the BRP demonstrations.

Rods are available from BRP once per year in mid-summer. A series of five primary phase test groups is tentatively planned. The first test, B1, has been scheduled at a burnup of only 6 MWD/kgU. It is anticipated that this test may be delayed to a later date or the rods assigned among the remaining tests. After power ramp testing, an optimal reconditioning and reramping is illustrated; however, these options are not within currently planned program scope. Selection of the improved concepts to be tested will be based upon the screening phase results.

The test result indices, method of measurement, and period 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 handling a moderate testing load. 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. 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 a compromise must be made in selection.

5.2.1 Screening Phase Tests

The discussion of the general testing plan, Section 5.1, indicated a need for a high degree of instrumentation in the screening phase tests. Those tests are to be comparative with a goal of obtaining a simple ranking of improved power cycling behavior and not an exact 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 screening the concepts. Based upon the developed

TABLE 5.1 Test Indices

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

TABLE 5.2 Comparison of Testing Facilities

Item	HBWR (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.

and proven instrumentation capabilities and cost considerations, the HBWR has been selected for the screening phase tests.

A draft of an agreement to conduct parameter irradiation tests (Task 2B) in the HBWR was composed and is currently in the review and approval process. 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 Nuclear Regulatory Commission and IFA.

5.2.2 Primary Phase Tests

Primary phase testing is intended to define the degree of power cycling improvement of one, or two, concepts as compared to the reference fuel. 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 accepting fuel rods or segments previously irradiated in BRP. An important requirement is that the test reactor accept the enrichment of BRP fuel rods. Clearly the HBWR cannot provide this environment and another test reactor will be used for the primary phase testing. The selection of a reactor is being deferred pending further development in BRP demonstration fuel rod design.

5.3 Testing Plan Development-Screening Phase

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 a 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) individual power ramping of the six rods from the lower cluster in the HBWR 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,

- d)' repetition of step (a),
- e) repetition of steps (b) and (c), with the lower rods ramped at a peak burnup of 15-20 MWD/kgU,
- f) repetition of (d) and (e), and
- g) individually ramping the last six upper cluster rods after a burnup of 8-10 MWD/kgU.

The physical layout and power distributions of the rods during cycling and ramping are illustrated in Figure 5.2. Three-fourths of the rods will be ramped from the lower cycling position. The highest power in rods from the lower position is 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 more consistent gauge-length.

As a result of the above description, six rods from the lower cluster and six rods from the upper cluster will be ramp tested after 10 MWD/kgU burnup. Twelve rods from the lower cluster will be ramp tested after 20 MWD/kgU burnup.

The general cycling/ramping schemes are illustrated in Figures 5.3a and 3b. Figure 5.3a illustrates the cycling/ramping scheme for the initial lower six-rod cluster. Figure 5.3b illustrates the cycling/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/kgU, the midpoint burnup shown in Figure 5.3b. The burnup periods illustrated should be considered idealized. Final selection of burnup periods 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.

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

FIGURE 5.3. General Cycling-Ramping Schemes

The screening phase test rig and rod characteristics are listed in Table 5.4. Power control during 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 period at the previous peak power in the cycling rig prior to ramping will be completed to reduce any uncertainties in 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 detectors to determine axial power variation and radial flux tilts, and a failure detector that measures fission products in the coolant.

The design basis for screening phase rods is of current ENC design. The fueled length for all rods will be 500 mm. This length is the maximum length that can be accommodated in the ramping rig. Zircaloy-2 tubing (12.29 mm OD x 10.26 mm ID) used in ENC designs will be used for all test rods. Tubing from the same cladding lot will be used in all test rods. All fuel types will utilize the same fueled length and plenum length.

Differences in power due to changing smear densities will be partially compensated by appropriate placement in the anticipated flux tilt. An initial enrichment of 10.0% U-235 will allow ramping powers of 65 kW/m (19.8 kW/ft) after 20 MWD/kgU burnup. 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 approximately two rods from the initial lower cluster. The combination elongation-failure monitors and the pressure monitors will be operative during both the cycling and ramping portions of the tests. The thermocouples will be operative only during the cycling portion.

TABLE 5.4
Screening Phase Test Characteristics

Item	Description
<u>Test Reactor</u>	Halden Boiling Water Reactor
<u>Rig Characteristics</u> (IFA-518)	
Geometry-Cycling	Two six-rod clusters, one above the other
Cooling Mode-Cycling	Forced cooling
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	Natural Convection
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	ENC design
Rod Diameters, mm	12.29 x 10.26
Fueled Length, mm	500
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	Approximately four rods from initial lower cluster
Rod Pressure Monitors	Approximately two rods from initial lower cluster
Axial End Power Peaking	Prevented by use of intermediate enrichment pellets on ends of fuel column

5.4. Testing Plan Development-Primary Phase

Based upon the results of the screening phase tests, the best improved concepts will be chosen from the rods available in the Task 2C demonstration irradiations for further ramp testing in the primary phase.

As listed in Figure 5.1, approximately 15 primary phase ramping tests will be carried out. Rods from the various BRP series will be selected. At least three burnup levels will be tested. The general primary testing scheme is illustrated in Figure 5.4. As mentioned in the previous sections, neither the test reactor nor the ramping power have been determined. In order to: (1) maintain the flexibility of choice of the available reactors during the primary phase 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, it is expected that some of the BRP demonstration rods will be segmented into shorter lengths. Segmentation of the fuel is currently under consideration for design.

FIGURE 5.4. General Primary Testing Scheme

6.0 TASK 2C - IN-REACTOR DEMONSTRATIONS

R. K. Welty (ENC)

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

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

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

6.1 Series-0 Design and Fabrication

Initial Series-0 fuel rods have been fabricated and will be inserted into the Big Rock Point reactor for irradiation. This series consists of 28 reference fuel rods, four rods with undished chamfered pellets and eight rods of annealed cladding. The pellet chamfer was nominally 50° measured from the pellet axis and a length of 0.13 mm (0.005 in.) to 0.38 mm (0.015 in.) as measured along the axis. The cladding was annealed at approximately 513°C (960°F) was to achieve the following properties:

Ultimate Tensile Strength

MPa (psi)

483 (70,000)

262 (38,000)

Temperature

°C (°F)

21 (70)

382 (720)

Yield Stress, 2% Offset

Mpa (psi)

414 (60,000)

207 (30,000)

21 (70)

382 (720)

Elongation in 2 inches, %

Min. Max.

24 34

32 44

21 (70)

382 (720)

The Series 0 rods were fabricated and contributed to the program at its early stages of development as part of ENC's cost sharing contribution. The rods were felt to have potential value in the program prior to a full assessment and selection of fuel concepts. The rods have been delivered to the Big Rock Point reactor for irradiation; however, a decision concerning their ultimate use in the program will be deferred pending further definition of concepts to be evaluated. Series 0 irradiations are also being used as a prototype irradiation to develop data acquisition procedures.

All fuel rods were specially characterized with regard to pellet and clad dimensions. Table 6.1 lists the characteristics of the rods. 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.

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

TABLE 6.1

SERIES-0 RODS

Fuel Bundle No. G3-14

<u>Rod Ser. No.</u>	<u>Quan.</u>	<u>Rod Process</u>	<u>Special Clad Treat.</u>	<u>Clad Lot Number</u>	<u>Pellet Process</u>	<u>Enri.</u>	<u>Pellet Lot #</u>	<u>Archive Sample</u>
IL400001 IL400002	2	Strd.	Annealed	0254-0000-5479	Char.	3.82	99-2	5 pellets from lot 99-2
IN400001 IN400002	2	Unaut. as etched	Annealed	0254-0000-5479	Char.	3.82	99-2	5 pellets from lot 100-1
IO400001 IO400002	2	Unat. as grd.	Annealed	0254-0000-5479	Char.	3.82	99-2	5 pellets from lot SR-26-1 R1
IP400001 IP400002	2	Anodized	Annealed	0254-0000-5479	Char.	3.82	99-2	2 full-length clads from lot 0254-2000-5479
IZ300001 thru IZ300004	4	Strd.	None	0254-2000-5027	Undisch Cham.	2.52	SR26-1R1	1 fuel rod serial serial no. 1H300059A
IH400061A thru IH400070A	10	Strd.	None	0254-2000-5027	Char.	3.82	99-2	
IH300042A thru IH300058A IH300060A	18	Strd.	None	0254-2000-5027	Char.	2.52	100-1	

TABLE 6.2

PELLET CHARACTERIZATION DATA

LOT NO. 99-2, SAMPLE SIZE 100

	Average	Standard Deviation	Upper	95/95 Range Lower
Density, % TD	93.79	.2156	94.27	93.31
Diameter, mm (in.)	9.4325 (.37136)	.0077 (.0003036)	9.4498 (.37204)	9.4153 (.37068)
Length, mm (in.)	7.5651 (.29784)	.0783 (.003082)	7.7399 (.30472)	7.3904 (.29096)
Dish Volume, %	1.869	.06239	2.008	1.730

LOT NO. 100-1, SAMPLE SIZE 100

Density, % TD	93.81	.2309	94.33	93.29
Diameter, mm (in.)	9.4338 (.37141)	.0063 (.0002479)	9.4480 (.37197)	9.4198 (.37086)
Length, mm (in.)	7.5418 (.29692)	.0842 (.003316)	7.7297 (.30432)	7.3536 (.28951)
Dish Volume, %	1.908	.06842	2.060	1.755

LOT NO. SR 26-1*, SAMPLE SIZE 50

Density, % TD	93.15	.2908	93.84	92.46
Diameter, mm (in.)	9.4348 (.37145)	.0030 (.0001182)	9.4419 (.37173)	9.4277 (.37117)
Length, mm (in.)	7.5725 (.29813)	.0901 (.003549)	7.7869 (.30675)	7.3579 (.28969)
Chamfer Angle, °	52.76	2.65	58.7	46.8

* Rev. 1

FIGURE 6.1
 FUEL ROD POSITION MAP
 BIG ROCK POINT RELOAD G-3
 FUEL BUNDLE NO. G3-14

			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

6.2 Series-1 Design and Analysis

Conceptual design parameters and preliminary information supporting the technical bases for Series-1 fuel rods were reviewed for use in fuel design for BRP reload G-4. Thermal-hydraulic scoping calculations of centerline fuel temperatures were performed using conservative assumptions with regard to the initial pellet-to-cladding gap and subsequent fuel cracking and densification. Results showed the fuel temperature to be within current licensing criteria and without restrictions on current operational limits in the BRP reactor.

Reference drawings and specifications have been started for Series-1 test rods. Preliminary fuel assembly design calls for four identical test assemblies, each containing twenty-four Series-1 rods consisting of six each of the following medium enrichment fuel rod designs: reference, graphite-coated cladding, annular-pellet and packed-particle fuels. Preliminary neutronic design of the test assemblies indicates a beginning of life local power distribution essentially identical to standard ENC G-4 fuel assemblies.

Reference fuel rods will be a standard, medium enrichment, ENC G-4 fuel rod design with the rod and pellet dimensions fully characterized.

Graphite coated cladding fuel rods will be a standard, medium enrichment, ENC G-4 fuel rod design with a thin layer of graphite lubricant applied to the inner surface of the cladding in the active fuel region.

Annular pellet fuel rods will contain fuel pellets with a 10 volume percent hole in the center of the pellet. The pellets will have standard ENC G-4 outer dimensions. The pellet enrichment will be adjusted upward to provide equal Series-1 rod power.

Packed-particle fuel rods will contain high density particle ($>98\%$) UO_2 . The fuel column will be vibratory compacted to $87 \pm 1\%$ theoretical density in the active fuel region.

6.3 Fuel Process Development

6.3.1 Annular Pellet Process Development

Conceptional design of single station punch and die tooling has begun for annular pellet fabrication.

6.3.2 Coated-Cladding Process Development

Laboratory experimental work has begun at BNW in support of coated cladding fuel development. The objective of the initial investigation is to provide information for the near term needs in design analysis and graphite coating application methods.

6.3.3 Packed-Particle Fuel Process Development

Extensive tests were run with British Nuclear Fuels, Ltd. UO_2 powder in an attempt to obtain high density UO_2 pellets which would then be crushed for feed material for vibratory compaction. BNFL powder was used as it is reported to be a very active powder. The effects of lubricants added to the powder was investigated to reduce porosity within the pellet matrix. When lubricants are omitted from the powder, very small length to diameter ratio pellets must be pressed to prevent end capping and laminations.

Table 6.3 is a summary of the tests run to date. The highest consistent density achieved was 98% T.D. which is not high enough based on BNW's experience involving the relationship between compaction efficiencies and fuel rod smear densities for various particle densities. BNW's experience indicates that 98% T.D. is the lowest fuel particle density that can be used for 87% fuel rod smear densities. The conclusion from these tests is that sintering pellets is not an optimum method to achieve high density fuel particles for vibratory compaction.

As an alternate, a decision has been made to use a Dynapak machine located at BNW to densify the UO_2 powder for packed particle fuel rods. The Dynapak has been used in the past to produce fuel particle density in excess of 98% T.D. for the PRTR.

TABLE 6.3
SINTERED PELLET DATA FOR PACKED-PARTICLE FUEL

<u>Method of Prep.</u>	<u>Sintering Time and Temperature</u>	<u>Sintered Density (T.D.)</u>
A) Pressed directly	4 hrs. 1730°C	87.0%
B) Slugged - granulated - pressed	4 hrs. 1730°C	93.4%
C) Slugged - granulated - pressed	5-1/2 hrs. 1750°C	96.8%
D) Resintered (c)	5-1/2 hrs. 1730°C	97.2%
E) Slugged granules	13-1/2 hrs. 1730°C	74.7%
F) Slugged (0.1% lubricant added) grannulated - pressed	24 hrs. 1730°C	97.0%

7.0 TASK 3 - FUEL PERFORMANCE EVALUATION

M. D. Freshley (BNW)

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

The Performance Evaluation task includes the establishment of the data requirements for preirradiation characterization, postirradiation evaluations, data analysis. Evaluation of the fuel concepts under study will use the information developed in the various testing phases of this program as well as results obtained from on-going 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 pre- and postirradiation behavior comparison. Equally important is careful attention to an evaluation of the conditions to which the experiments are exposed in the reactor during irradiation. 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 and analysis.

8.0 TASK 4 - ESTABLISHMENT OF TECHNICAL BASES

K. R. Merckx (ENC)

The objective of this task is to transfer program results to commercial application through the establishment of technical bases for design. 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 and computer codes which have been accepted for design use.

The establishment of design criteria in this program is concerned with the selection of appropriate data and analysis procedures which can be used in the design of fuel with improved power cycling capability and to assess its performance in a commercial power reactor.

Initial activity of this task has concentrated on identifying data requirements for establishment of technical bases and activating a fuel performance code to help assess the fuel concepts under evaluation.

8.1. Data Requirements

Initial activity has concentrated on assisting Task 3 with the establishment of a computerized data file. The existing data base and computer file^(8.1) developed as part of the NRC sponsored program for LWR fuels has been adopted for use in this program. The file has been restructured to allow a consistent data input format and to allow inclusion of more data. This initial restructuring, conversion to the CDC-6600 and writing of a compatible read program has been completed. The data in the present file are essentially that of the original file. Further expansion of the file contents is expected to continue with contributions from the NRC sponsored program^(8.1) for LWR fuel and from the fuel performance evaluation efforts of this program.

Specification of general data requirements for the characterization of the commercial demonstration fuel has been outlined. A general plan has been developed which is structured to store data in computer files as it

is obtained from manufacturing, irradiation, and examination. Files are being formulated to accept information similar to that specified for other characterized test programs^(8.2, 8.3). Efforts are being directed toward using similar data requirements and data filing techniques as are employed in the Power Shape Monitoring Program^(8.3) for the fuel characterization. Some of the fabrication data have been obtained and their general applicability for inclusion is under study. Specific requirements are also being established for the data acquisition from poolside examination. ENC fuel management computer codes are planned to be used to obtain fuel rod histories. These programs require modification for application to Big Rock Point because of this reactor's particular control rod configuration. In summary, general data and computer file requirements have been outlined and areas requiring further development have been noted.

8.2 Fuel Performance Code

It is expected that evaluation and assessment of fuel concepts will require the use of a fuel performance computer code. The establishment of technical bases will also require the selection of appropriate calculational models to support the design analysis of fuel concepts. To satisfy this need a fuel performance code(s) is being activated for use in all tasks of the program. Currently the NRC sponsored development code FRAP-S has been obtained and activated on a CDC-6600 computer. This code has been selected because it is in the public domain and is expected to receive continued developmental support. Alternate codes meeting these requirements are also being sought.

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