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Fuel Performance Improvement Program

Quarterly Progress Report
January - March 1978

Consumers
Power
Company

EXXON NUCLEAR
COMPANY, Inc.

Pacific Northwest Laboratory

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FUEL PERFORMANCE IMPROVEMENT PROGRAM

**Quarterly Progress Report
January - March 1978**

Compiled by C. E. Crouthamel

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

April 1978

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Approved: _____

L. P. Bupp

L. P. Bupp
Program Office

SUMMARY

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

The final version of the Task 1 topical report "State-of-the-Technology Review of Fuel-Cladding Interaction" (C00-4066-2/PNL-2488) was completed and submitted for clearance approvals.

The major revision of the Task 1 topical report "Assessment of Fuel Concepts" (C00-4066-3/PNL-2490) was completed as scheduled. Conclusions and recommendations have been summarized in this quarterly report. Clearance approvals are expected in May, 1978.

Task 2 - Testing and Demonstration Program

Selection of fuel concepts for the first phase of the evaluations was completed and was based on (1) the results of Task 1 topical reports; (2) fuel design concepts that do not represent a significant departure from commercial technology; and (3) the existence of an adequate data base to facilitate commercial application.

The combination of the primary concepts to be included in fuel designs for initial screening tests in the Halden Boiling Water Reactor (HBWR) are as follows:

- reference fuel
- annular fuel
- coated-cladding/annular fuel
- packed-particle (vipac) fuel

The initial group of 144 rods to be irradiated in the Big Rock Point Reactor (BRPR) will also include about 24 segmented, full-length rods. The fuel segments, or rodlets, which are to be irradiated in BRPR and subsequently ramped in a test reactor, will be comprised of:

- reference fuel
- coated-cladding/reference fuel
- annular fuel

- coated-cladding/annular fuel
- packed-particle (vipac) fuel

Prepressurization has been evaluated to have a significant beneficial effect on reducing fuel-cladding-interaction effects and is not considered to be a major departure from current technology. Therefore, prepressurization with 4-6 atmos. of He in combination with the reference, annular pellet/coated-cladding, and packed-particle fuel designs is being recommended for evaluation in the BRPR Series 1 demonstration rods.

Task 2A - Out-of-Reactor Studies

Out-of-reactor studies were continued in support of the Halden Series H-1 test fuel fabrication. The major effort was in the evaluation of graphite coatings applied to the inside surface of cladding tubes. Advanced coating evaluations were also continued.

Also, mechanistic studies were continued to understand the relationship between pellet-cladding-interaction failures during irradiation and fuel performance code analysis results.

Task 2B - In-Reactor Experiments

In-reactor experiments are being conducted in the HBWR to study the detailed performance characteristics and limits of the fuel rod design concepts selected. Fabrication of the test rods is underway this quarter and the rods for IFA-518 will be completed and shipped during April 1978 to Halden, Norway. The screening phase test matrix for H-1 through H-3 at Halden include the various instrumentation designations (elongation sensor, pressure transducer, and fuel center thermocouple).

Task 2C - In-Reactor Demonstrations

This quarter in-reactor demonstrations are primarily involved with the fabrication of the Halden H-1 series test rods.

Design analysis of the BRPR Series 1 Big Rock Point Reactor demonstration fuel is continuing.

Fuel process development required the major manpower effort and was applied to the following: pellet process development for reference and annular pellets, packed particle fuel process development, graphite coated cladding development, welding process development for rods with low void volumes and special Halden hardware.

Task 3 - Fuel Performance Evaluation

Fuel performance evaluation serves as a focal point for the integration and evaluation of data from the FPIP and evaluates the results of other fuel clad interaction related programs. During this quarter the requirements for irradiation history data in HBWR and BRPR were defined and sources and methods of collecting the data are being checked.

Task 4 - Establishment of Technical Bases

Establishment of technical bases draft version for the Big Rock Point Reactor demonstration fuel has been prepared and circulated for internal project review.

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1.0 INTRODUCTION

C. E. Crouthamel (Exxon Nuclear Company)

1.1 Program Objectives

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

1.2 Background

The concerns in both BWR and PWR light water reactor fuel performance have been discussed in previous quarterly reports^(1.1, 1.2).

1.3 Program Description

The program description and history have been documented in the previous quarterly reports^(1.1, 1.2) which also give the task structures and brief descriptions of each task.

1.4 Program Organization

The performance of the program involves the direct participation of Consumers Power Company (CPC), Exxon Nuclear Company (ENC), and Pacific Northwest Laboratory (PNL). 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 PNL. The functional arrangement of the program organization by task is shown in Figure 1.1.

As prime contractor, CPC has overall responsibility for directing the program. This is accomplished through a Steering Committee under the chairmanship of the Program Director. CPC has designated ENC to manage a Program Office for the execution of the program subject to policy directions and review by the Steering Committee. In addition, ENC is responsible for the design, process development, and fabrication of fuel concepts under study.

PNL is primarily responsible for 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.

REFERENCES

- 1.1 "Fuel Performance Improvement Program, Quarterly/Annual Progress Report, April-September 1977," C00-4066-4, November, 1977.
- 1.2 "Fuel Performance Improvement Program, Quarterly Progress Report, October-December 1977," C00-4066-5, January, 1978.

FIGURE 1.1: Fuel Performance Improvement Program Organization

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

W. J. Bailey (Pacific Northwest Laboratory)

The objective of the Task 1 activity is to assess the state of the technology of fuel-cladding interaction and to identify fuel concepts with potential for improved performance capabilities in order to provide direction and technical support for the Fuel Performance Improvement Program. The state-of-the-technology review presents a summarization of postulated fuel-cladding-interaction mechanisms as reported in the open literature. Those concepts that will be investigated as part of this program and the technical bases which support their selection are presented in the assessment report.

2.1 State-of-the-Technology Evaluation

2.1.1 Status of Activity

The final version of the topical report entitled "State-of-the-Technology Review of Fuel-Cladding Interaction" (C00-4066-2/PNL-2488) was submitted for clearance approvals as scheduled. Distribution of the report is pending final release approval.

2.1.2 Conclusions^(a) in the Topical Report

The zircaloy cladding on irradiated UO₂ fuel is sometimes breached as a consequence of fuel-cladding interaction when the power is rapidly increased. *However, fuel temperature rather than power may be the key variable.* As indicated by Fuhrman et al. (EPRI NP-218), a fuel temperature increase resulting from a decrease in fuel-cladding heat transfer can be a major contributor to fuel-cladding interaction. *Reducing the temperature within the fuel has the effect of reducing the thermal-mechanical driving force for fuel-cladding interaction, as well as reducing the release of aggressive fission products*

(a) Interpretive comments and conclusions by the authors of the report are italicized.

from the fuel. The probability for a fuel-cladding-interaction (FCI) defect to occur appears to increase with power, amount of power increase, rate of power increase, dwell time at the high power, and burnup.

For fuel that fails due to fuel-cladding interaction during normal operation, including anticipated transients associated with power ramps, the principal mode of failure appears to be stress-corrosion cracking (SCC). Fuel-cladding interaction in Zircaloy-clad UO_2 fuel rods appears to be a mechanism for failure that is generic to both BWR and PWR fuel. In general, it appears that stress (a mechanical process) and fission-product availability (a chemical process) are reciprocal. It also appears that a pin-hole-type failure may be the result of a high fission-product concentration and a low stress in the cladding, while an axial-split-type cladding failure may be the result of high 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. If the threshold stress is low, the cracks tend to be straight with the predominant failure mode being transgranular cleavage. If the threshold stress is high, more crack branching and more fracture-surface fluting should be observed.

It has not yet been clearly shown experimentally which material parameter--stress, strain, 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 but there is doubt as to whether a critical strain is a prerequisite for SCC. Studies indicate that SCC of Zircaloy is strain-rate

dependent; studies on material other than Zircaloy indicate that SCC occurs within a narrow range of strain rates.

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

Crack nucleation in the Zircaloy cladding is the key step in fuel-cladding interaction failures. Some investigators indicate that once the oxide film on the inner surface of the cladding is fissured, small cracks [$\sim 10 \mu\text{m}$ (~ 0.00039 in.)] appear to develop in the metal at preferred sites (e.g., second-phase particles, slip-band intersections, slip-band grain boundary intersections where the orientation is unfavorable for slip across the grain boundary, or combinations thereof). The mechanism for the breakdown of the protective oxide film on the inner surface of the cladding is unknown. A critical stress is required for growth [to $20 \mu\text{m}$ (~ 0.00079 in.)] of the cracks. Once formed, the cracks propagate fairly rapidly, predominantly by transgranular cleavage. If the tensile stress or aggressive fission-product environment is not sustained, propagation of the cracks will proceed at a slow rate (intergranular characteristics are favored in this case) and the crack may arrest.

Hence, evaluating intergranular areas at the crack front might be employed as a means of determining whether fuel-cladding-interaction damage is cumulative.

Sensitivity of Zircaloy to embrittlement by fission products and impurities which deposit at the fuel-cladding interface of irradiated LWR fuel rods is generally unknown. The likely aggressive fission-product species of concern are iodine vapor (released from CsI), *cadmium vapor and/or* liquid cesium, in

that order. Characteristics of the cladding, fuel pellet, and fuel rod that may influence SCC are identified in this report.

At present, the detailed characteristics of the power changes that are normally required of reactors are not sufficiently well known to permit a fundamental understanding of the fuel-cladding-interaction phenomenon. There is also a distinct difference between test reactors and power reactors in the absolute heat rating conditions required to produce comparable power-ramping effects. The operating environments and fuel designs (e.g., length) utilized in the two reactor types are different; however, it has also been postulated that a contributing factor to this difference may be the disturbance of the fuel-cladding interface during manipulation of the fuel rods during the conduct of the irradiations in a test reactor.

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) of the zircaloy in the fuel-cladding interface region.*
- *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 employed to select the primary fuel concepts for evaluation as part of this program.

2.2 Fuel Concept Assessment

2.2.1 Status of Activity

The major revision of the draft of the topical report, "Assessment of Fuel Concepts" (C00-4066-3/PNL-2490), was completed as scheduled. Resulting comments are now being incorporated into the report prior to publication. A detailed check was made to verify that the information and/or references used are publicly available.

2.2.2 Conclusion and Recommendations in the Topical Report

The relative merits of various LWR UO_2 fuel concepts with the potential for improved power-ramping capability were qualitatively assessed. In the evaluation, it was determined that of the various concepts being considered, those that presently possess an adequately developed experience base include annular pellets, cladding coated with graphite on the inner surface, and packed-particle fuel. Therefore, these were selected for initial evaluation as part of the Fuel Improvement Program. For this program, graphite-coated cladding is being used in conjunction with annular pellet fuel as one of the concepts with the anticipation of gaining the advantage of the combined improvements.

Annular pellet fuel was selected because, under comparable heat ratings, it will operate at lower fuel temperatures than solid pellet fuel. Hence, the driving force (fuel temperature) for thermal-mechanical interaction and chemical reaction in annular pellet fuel is expected to be reduced because annular pellet fuel is expected to produce lower general cladding stresses; lower localized stresses at pellet-pellet interfaces; lower releases of fission products, hence a decrease in the likelihood of stress-corrosion cracking; and improved safety margins in comparison to solid pellet fuel. The existing experience base indicates that annular pellet fuel has

performed entirely satisfactorily. Efforts to develop, fabricate, and license annular pellet fuel should be minimal because of the extent to which this has already been done.

Coated cladding was selected because of the favorable experience with fuel rods that had a graphite coating on the inside surface of the cladding. In power-ramping tests in support of the Canada Deuterium Uranium (CANDU) reactor system, such fuel rods exhibited improved behavior in comparison to fuel rods with uncoated cladding. Though the mechanisms for behavioral improvement are not fully understood, it is suspected that the graphite acts either as a lubricant, an adsorber or absorber of active fission products, or a barrier to certain fission products or, most likely, a combination of these effects. Commercial fuel rod designs with graphite-coated cladding are currently in use in Canadian power reactors (e.g., Bruce and Pickering). Graphite-coated cladding should represent a minor deviation from current practice; however, material selection, application, outgassing, and handling technology will need to be developed. Licensing efforts should be minimal. The effect of high burnup on fuel rods with coated cladding in BWR and PWR environments needs to be studied.

Packed-particle fuel was selected because localized cladding deformation is virtually absent for this fuel type and of its potential as an FCI remedy. The performance of a significant amount of packed-particle fuel has generally been demonstrated to be satisfactory. No inherent performance limitations are evident. Packed-particle fuel has been shown to perform equally well as or, in some cases, superior to pellet fuel. Most behavioral problems that have occurred with Zircaloy-clad, packed-particle fuel can be attributed to the entrapment of excessive quantities of hydrogenous impurities (e.g., moisture) within the fuel rods. However, this fabrication-technology-related problem is not unique to this fuel type. By implementing

sound fuel fabrication methods and procedures, it has been demonstrated in the past that the problem can be alleviated. It will be important to convincingly demonstrate that contaminants in the packed-particle fuel can indeed be kept at an acceptably low level under manufacturing conditions. The development of commercial fabrication capabilities for vipac and sphere-pac fuel may be expected to be significant. The effort to license reload quantities of vipac and sphere-pac fuel may be expected to be significant; however, packed-particle fuel has been licensed previously for use in two United States commercial BWRs (Big Rock Point and Dresden-1). Safety-related considerations [e.g., loss of cooling accident (LOCA)] associated with packed-particle fuel are to be evaluated as part of this program.

Prepressurization with helium has been reported to have a beneficial effect by reducing fuel-cladding interaction in both BWR and PWR fuel designs. Because it is not considered to be a major departure from current technology, prepressurization in combination with the annular, coated-cladding, and packed-particle fuel designs is being considered for evaluation in this program as a fuel-cladding-interaction remedy.

Several other fuel concepts had attributes which were considered to have significant potential for improving power-ramping behavior. However, it was judged that many of those concepts require further evaluation of their effect on power-ramping behavior or further out-of-reactor testing before performing in-reactor evaluations as part of this program. Other concepts with potential for improving fuel-cladding-interaction behavior are being evaluated in other programs. Among those concepts are: chamfered pellets, variable (radially) enrichment fuel, barriers, cladding with noncylindrical inside surface, pellet spacers, getters and other means for control of fission products,

and several concepts involving fuel and cladding property variations. Results from these other programs will be monitored. Several possible concepts were deleted from further consideration because they had a low probability of being developed, and/or they represented a major deviation from current technology, and/or the degree of anticipated behavioral improvement was minor.

3.0 TASK 2 - TESTING AND DEMONSTRATION PROGRAM

M. D. Freshley (Pacific Northwest Laboratory)

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 performance capabilities. The program includes Task 2A, Out-of-Reactor Studies; Task 2B, In-Reactor Experiments; and Task 2C, In-Reactor Demonstrations.

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

Annular pellet, coated-cladding, and packed-particle or vipac fuel are the basic fuel concepts deemed to have potential for improved power-ramping capability that have been selected for initial investigation. Selection of these concepts for the first phase of the evaluations is based on: 1) the results from the Task 1 state-of-the-technology evaluation; 2) concepts that do not represent a significant departure from commercial technology; and 3) existence of an adequate data base to facilitate application.

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

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

In addition to the fuel concepts listed above, the initial 144-rod group of rods to be irradiated in BRP will include coated-cladding/reference fuel. Also included in the initial BRP loading will be 24 segmented, full-length rods. The fuel segments, or rodlets, which are to be irradiated in BRP and subsequently ramped in a test reactor, will be comprised of:

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

Prepressurization has been reported to have a significant beneficial effect on reducing fuel-cladding-interaction effects and is not considered to be a major departure from current technology. Therefore, prepressurization with 5-10 atmos. of He in combination with the annular coated-cladding, reference, and packed-particle fuel designs is being considered for evaluation as a fuel-cladding-interaction remedy.

Coordination of sub-task activities is continuing in order to maintain scheduled progress. Resolution of problems associated with producing the HBWR test rods is currently the primary activity. Progress in addressing the critical items is being made.

The particle density of the Dynapaked feed material for the Halden vipac rods was low (97.9% T.D.) and a post-Dynapaking sintering operation did not increase the density (perhaps due to heating too rapidly). This resulted in low (85-86% T.D.) vipac fuel densities. Consequently, the material was re-Dynapaked by adding UO₂ to adjust stoichiometry and after sintering, particle densities of +99% T.D. were achieved. A vipac test with this material resulted in an acceptable smear density of 86-87% T.D.

4.0 TASK 2A. OUT-OF-REACTOR STUDIES

L. R. Bunnell and P. E. Hart (Pacific Northwest Laboratory)

The purpose of this task is to conduct application-oriented, out-of-reactor studies in support of the development and demonstration of fuel concepts that will minimize fuel-cladding interaction (FCI) failures during reactor ramping. During the second quarter of fiscal year 1978, activities included further refinement of the process for applying graphite coatings, characterization of these coatings, and, in cooperation with ENC, preparation of the coated cladding for the Halden H1 test.

4.1 Outgas Studies

As reported previously,^(4.1) graphite coatings contain significant quantities of H_2 and H_2O following outgassing. In the process of developing an outgas procedure which would produce acceptable levels of hydrogen sources in the coating, the dag was processed under optimum conditions, i.e., a small quantity of dag was outgassed in a large, well-pumped vacuum furnace at low pressure, $<1.3 \times 10^{-3}$ Pa, the temperature of $450^\circ C$ was the maximum allowable for the cladding, and the time of 24 hr was the longest practical. During actual production, the pressures will probably be higher, the temperatures lower to allow a control safety margin, and it would be convenient to outgas for a shorter time. Toward this end, a series of outgas runs was made to investigate the effect of these processing variables. Water content was determined by a combustion technique in which water and all hydrogen released to $800^\circ C$ were measured. Results are summarized in Table 4.1. From these results, the following observations can be made:

1. The first four measurements are consistent in that higher temperature and lower pressure produced the lowest moisture content, while the lower temperature and higher pressure resulted in the highest water content. The other two conditions were intermediate and equivalent.

TABLE 4.1. Water and (Calculated) Hydrogen Contents of Dag 154, Outgassed Under Various Conditions

<u>Outgas Conditions</u>		<u>Pressure, Pa</u>	<u>Water Content, ppm</u>	<u>Fuel Wt. Basis*, ppm</u>	
<u>Temperature, °C</u>	<u>Time, hr.</u>			<u>H₂</u>	<u>H₂O</u>
450	24	1.3×10^{-3}	3,400	0.20	1.6
400	24	1.3×10^{-3}	7,300	0.40	3.5
450	24	$1.3 \times 10^{-1\dagger}$	6,600	0.40	3.1
400	24	1.3×10^{-1}	10,000	0.50	4.8
450	4	5.2×10^{-3}	7,400	0.40	3.5
400	4	1.3×10^{-3}	5,300	0.30	2.5

* For fuel-weight basis conversion, the ratio: weight fuel/weight coating = 2100, for a 7.6 μm (0.3 mil) coating.

† Pressures higher than 1.3×10^{-3} Pa were obtained by use of a controlled Ar leak.

2. Outgassing for a period of 4 hr produces acceptable results, though it appears that the pressure should be low.
3. All of the outgassing conditions listed in Table 4.1 will produce graphite coatings of acceptable moisture content.

During preparation of the graphite-coated tubes for the Halden H1 irradiation, outgassing was done at $425 \pm 25^\circ\text{C}$, 1.3×10^{-2} Pa, for 24 hr. To obtain 95/95 statistics, analysis for water and outgas were obtained in quintuplicate. The average water content was 3.1 ppm, $\sigma = 0.4$ ppm on a fuel weight basis, and the gas content to 1600°C was 14 cc/g; the gas was 67% hydrogen. On a fuel weight basis, this gas would add 0.4 ppm hydrogen to the system. As long as the coated tubes are stored in a dry atmosphere after outgassing, the coating will remain acceptably dry. Graphite scraped from a reject tube stored 18 days in a glove box containing 1 ppm H_2O was analyzed to contain 2.5 ppm H_2O , on a fuel weight basis.

4.2 Binder Decomposition

Dag 154 contains a thermoplastic binder which bonds the graphite platelets to each other and to the cladding. The outgassing process decomposes the binder to carbon; the details of the process were studied by TGA. The binder was obtained in as pure a state as possible by centrifuging Dag 154 for 16 hr at 200,000 G. Even this did not produce completely clear liquid, but it was nearly so. The supernate was dried to eliminate isopropanol, then heated in a thermogravimetric analyzer in a Zr-gettered Ar atmosphere at 100°C/hr. The decomposition becomes rapid at 300°C and is complete at 400°C (Figure 4.1). Thus, outgassing of the coated tubing at 425°C is sufficient to decompose the binder and produce a coating with an acceptably low hydrogen content.

4.3 Pellet Loading

Dag 154 is quite durable as applied. After outgassing, however, only a small amount of carbon from the decomposed binder holds the platelets together and bonds them to the cladding. Pellet loading is quite punishing to this almost binderless material, especially if pellet chips are produced. This was illustrated by an experiment in which 200 Halden-sized depleted UO_2 pellets were passed through a Halden-sized tube 51 cm-long with a 6 mm-thick graphite coating that had been outgassed 24 hr at 450°C, $P = 1 \times 10^{-3}$ Pa. Pellets were loaded as a group of five, making no special effort to perform the loading carefully. Though the amount of coating lost was small ($\sim 0.1\%$), the coating was removed to bare metal at the pellet contact point, especially in the first 20 cm where the pellets were not well-aligned. Auger spectroscopy confirmed that the coating was completely removed. The damaged inside surface area was examined (Figure 4.2) by SEM.

Due to the amount of graphite lost during the above test, additional simulated pellet loadings were performed in a more careful manner.

FIGURE 4.1. TGA Results Obtained from Dag 154 Binder

100 μ m

FIGURE 4.2. Inside Surface of Zircaloy Cladding After 200 Pellet Passages, in a High Damage Area. Light areas are bare metal.

FIGURE 4.3. Method of Testing Coating Adherence and Ductility by Deforming Coated Cladding

The tubes were then cut lengthwise and examined visually. Procedures and observations are summarized in Table 4.2.

TABLE 4.2 Conditions and Results of Pellet Loading Test

A. Tube inclined at 13° (13° is just below the angle at which pellets would slide spontaneously, implying a static friction coefficient of ca. 0.23), 200 pellets loaded as groups of five, tube rotated 180° about longitudinal axis after loading a 100 pellet increment.	Dag well-burnished, but contained only a few isolated scratches to base metal.
B. 50 pellets, more representative of Halden pellet number, tube horizontal, 180° rotation after 25 pellets, pellets loaded individually.	Coating burnished at high places, few isolated scratches to base metal.
C. 200 pellets, tubing horizontal, rotated 90° after each 50 pellet increment, pellets loaded as groups of five.	Coating burnished uniformly around circumference, like B expect fewer scratches.

Clearly, damage to the coating can be minimized by proper attention to loading technique. It would be very difficult, however, to avoid any scratches through the coating by anything approaching a normal pellet loading operation. The Big Rock Point fuel rods will require the most care because of the larger number of pellets which must be inserted. Efforts are under way to identify a dag which would be more durable, yet not contain hydrogen-bearing or hydrophilic binders.

4.4 Adhesion Testing

Much of the adherence between Dag 154 and cladding is lost when the coating is outgassed. Characterization of the bond is difficult because the interflake bonding is weaker than the coating-to-cladding bond. Cellophane tape, for instance, will strip away many (~10) consecutive layers of graphite flake without reaching base metal. So any

normal method of contact with the dag to test its adherence to the Zircaloy will result in failure in the coating.

Another way to characterize adherence is to deform a coated tube, measuring the strain which can be accommodated before the coating cracks and noting whether the coating remained adherent after cracking. Figure 4.3 shows the testing arrangement schematically. Examination of the coating by SEM indicates that outgassed Dag 154, applied to a thickness of 6 μm , will undergo about 1% tensile strain before fracture (Figure 4.4). Strain is determined by measuring the ovality of the specimen and calculating strain as a function of angular location. Although the coating does not become detached spontaneously after cracking, it will peel to base metal in one layer when tested with cellophane tape. Other candidate coating materials will be evaluated by the above test for purposes of subjective comparison.

4.5 Coating Thickness Measurement

The main method used for measuring coating thickness has been air gauging. The noncontact nature and sensitivity of air gauging makes it ideal for measuring a thin graphite layer without applying forces which would cause damage. Although standard air-gauging practices are followed and gauge blocks are used regularly, the thickness of an actual graphite layer had not been measured directly, i.e., by a metallographic technique. Thus, a coated tube was carefully air gauged and sectioned at the gauging site. Standard metallographic techniques for polishing graphite dictate vacuum-impregnation by a mobile resin to provide support during polishing. When this was done, the thickness of the dag layer was about three times that determined by air gauging; examination of the graphite showed that the resin had penetrated the coating and caused swelling. Another approach was to hand grind an unmounted specimen with SiC paper and observe the edge of the coating in the SEM. This technique only works when the coating is broken back a short distance from the Zircaloy; otherwise it is

50 μ m

FIGURE 4.4. SEM Micrograph of Coating Fractured in Adherence Test. The area pictured was strained ca. 2%.

5 μ m

FIGURE 4.5. SEM Micrograph Showing Graphite Coating Edge-on for Thickness Measurement.

obscured by flowed metal. Figure 4.5 was obtained using the above technique; the coating thickness can be measured as 7.2 μm , in reasonable agreement with the 6.4 μm obtained by air gauging.

4.6 Friction Testing

Initial friction testing, as reported previously,^(4.2) was done in an air atmosphere at room temperature. Testing by the same method, in which an oversized pellet is forced through a section of cladding and the force measured, was also performed at 400°C in a Zr-gettered Ar atmosphere. As before, the Dag 154-coated tubes had been out-gassed 24 hr, 450°C, at 1.3×10^{-3} Pa. Results are plotted in Figure 4.6. The results show the variability expected in a test where dimensions are critical, but they clearly indicate a sizable reduction in friction force associated with the coated tubes. The difference is greater than in the case of the room-temperature tests, even though the interference between pellet and cladding is greater because of thermal expansion differences between cladding and pellet. Because the graphite should be less lubricating in the dry environment of the 400°C test, galling would be expected to be more severe with the softened uncoated Zircaloy. Examination of the inner surface indicates that the galling was at least as severe as for the room-temperature tests.

4.7 Advanced Coatings

The search for improved coating materials continues. If MoS_2 is to be a candidate, a coating system containing no undesirable binders must be identified and the possibility of undesirable reactions with sulfur in a reactor environment must be addressed.

The possible use of alternate types of dag and siloxanes is being addressed; all pertinent information is being assembled.

4.8 Mechanistic Studies

An approach has been developed to study the relationship between pellet-cladding-interaction failures during irradiation and fuel performance

FIGURE 4.6. Plot showing Results of 400°C Dry Inert Gas Push-through Friction Tests. Both coated and uncoated tubes are shown. Shaded areas bounded all values measured, but each symbol type refers to a separate tube and pellet.

code analysis results. This approach involves determining the cladding failure threshold conditions and evaluating the relationships between the local stress-strain-time-history threshold and the corresponding states of stress and strain present during pellet-cladding interaction.

Current work is concentrating on evaluation of reported experimental results on SCC mechanisms to determine the stress-strain-time relationship for cladding failure. These results will be compared with analytical prediction of stress-strain conditions that exist at the fuel pellet-cladding interface to evaluate the applicability of these results in predicting cladding failure.

REFERENCES

- 4.1 "Fuel Performance Improvement Program, Quarter/Annual Progress Report, April-September 1977," C00-4066-4, November 1977.
- 4.2 "Fuel Performance Improvement Program, Quarterly Progress Report, October-December 1977," C00-4066-5, January 1978.

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5.0 TASK 2B - IN-REACTOR EXPERIMENTS

J. O. Barner and S. R. Wagoner (Pacific Northwest Laboratory)

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. Details of the general testing criteria for the screening phase, the bases used for selecting the irradiation facilities, and the detailed test descriptions were presented in References 5.1 and 5.2.

5.1 Screening Phase

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

Fabrication of the test rods was initiated during the quarter. General development of the fabrication equipment and techniques, as well as the status of fabrication, is summarized in Section 6 of this report. The development and fabrication of the nonstandard components was completed as part of Task 2B. All the end plugs for the test rods, both instrumented and noninstrumented, were fabricated by the Halden Project. Seven upper end plugs with W-3% Re/W-25% Re thermocouples, 10 lower end plugs with pressure sensors, 18 lower end plugs with combination elongation/failure sensors, and 24 upper standard noninstrumented end plugs were fabricated. Except for the upper end plug with thermocouples, these amounts include several spare end plugs of each type. Difficulties with procurement of a satisfactory hermetic seal for the thermocouple-type plug allowed fabrication of only one spare plug of this type.

A method of electron-beam welding short 0.5 mm diameter W-26% Re spacer wires to the 2.4 mm diameter W-22% Re thermocouple well was developed. The thermocouple well will be used to contain the fuel centerline thermocouple for the packed-particle fuel and the spacer wires center

TABLE 5.1. Task 2B: Screening Phase Test Matrix

ROD TYPE	H1LC	H1UC			TOTAL TESTED	TOTAL TESTED BURNUP, MWd/Kg
		H2LC	H2UC			
			H3LC	H3UC		
REFERENCE					6	3/10 3/20
ANNULAR					6	3/10 3/20
ANNULAR - COATED					6	3/10 3/20
PACKED-PARTICLE					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

the thermocouple well in the fuel column. A centered thermocouple well will assure reliable temperature measurement during operation and an adequate compaction capability during fabrication. Several of these thermocouple wells were fabricated for use in the packed-particle fueled rods for the initial loading of IFA-518.

A tandem Zircaloy-2 porous frit material was originally designed for the IFA-518 test rods. The upper section was to be a nominal 10 μm filter and the lower section was to be a nominal 5 μm filter. The filters were intended to prevent fuel particle or chip relocation into the instrumented region of the end plugs, yet allow gas flow from the fueled region of the rods to the end plug region. After receipt of the two frit materials, flow tests were performed on the individual filter and the tandem filter. The results of tests indicated that the tandem filter excessively impeded gas flow. The rod design was changed to incorporate a single 10 μm filter. The required filters were fabricated.

All of the test rods incorporate alumina insulator disks and medium-enrichment UO_2 pellets at each end of the fuel column to prevent power peaking. Central holes were required in some of the pellets to accommodate centerline thermocouples. A method of drilling these holes in the sintered pellets, using diamond drills, was developed and the required pellets were fabricated satisfactorily.

In preparation for shipment of the test rods to Norway, three DOT-6M shipping containers (110 gallon size) were purchased. Polyethylene inserts were designed to contain the rods and prevent damage during shipment. Three of the inserts were fabricated. An export declaration for the fuel material was received from the appropriate government agencies.

5.2 Performance Predictions

The presence of thermocouples in the fueled region of selected test rods will cause localized perturbations in peak fuel temperatures when

compared with rods without thermocouples. A computer study was initiated to determine the magnitude of two of these perturbations (gamma heating in the thermocouple and axial heat conduction up the thermocouple) on the recorded temperature. At the present time all fuel types, except packed-particle fuel, have been analyzed.

The code used in the analysis was the three-dimensional asymmetric heat transfer code; TRUMP^(5.3). To determine the effect of gamma heating and axial conduction on the fuel centerline temperature, a series of calculations were made for each fuel type. For the reference, annular and annular-coated fuel types, each of the following cases were evaluated:

- 1) Fuel without a thermocouple (for the reference fuel, the fuel was modeled both with and without a thermocouple hole);
- 2) Fuel with a thermocouple; and
- 3) Fuel with a thermocouple and gamma heating.

For the study, the rods were modeled from the top of the Al_2O_3 insulator pellet to the bottom of the thermocouple. The model included the low enrichment (2.78% ^{235}U) pellet located at the top of the fuel stack. The model without a thermocouple in the fuel stack assumed the top boundary of the Al_2O_3 insulator pellet to be adiabatic.

Information used to model the fuel rods was obtained from several sources. The radial gap size was matched to the gap sizes calculated by GAPCON-THERMAL-III.^(5.4) The flux depression was calculated from equations used in the GAPCON-THERMAL-III code. The thermal conductivity values for the UO_2 were obtained from characterization of other Halden experiments.^(5.5) Thermal conductivity of the insulator pellets and the thermocouple materials were taken from Touloukian.^(5.6) Gamma heating values in the wire thermocouple were assumed to be 0.86 kW/kg. Due to the expected relative flatness of the axial flux profile in the area of the thermocouple during the experiment, a constant linear power of 43.6 kW/m was used for the fuel length containing the thermocouple.

The design parameters used in the model of the fuel are summarized in Table 5.2.

Using the above information, the temperature distribution in the rods was calculated. The centerline temperature calculated at the tip of the thermocouple, as well as the corresponding radial hot gap, are shown in Table 5.3.

The centerline temperature calculations predict that the axial heat conduction of the thermocouple does not cause any significant temperature perturbation at the thermocouple junction. However, the addition of the thermocouple slightly decreased the fuel temperatures near the top of fuel stack.

The results also predict that gamma heating in the thermocouple will be significant. The heat generation from gamma heating causes only a small increase in the centerline temperature calculated at the thermocouple.

TABLE 5.2. Design Parameters for the Fuel Types

	Reference Fuel	Reference Fuel with Thermocouple Hole	Annular Fuel	Annular Coated Fuel
Clad ID (mm)	10.52	10.52	10.52	10.52
Clad OD (mm)	12.29	12.29	12.29	12.29
Fuel OD (cold) (mm)	10.26	10.26	10.26	10.25
Fuel OD (hot) (mm)	10.43	10.43	10.39	10.38
Center Hole Diameter (mm)	-	1.02	1.63	1.63
Coating Thickness (mm)	-	-	-	0.0064
Nodes in Fuel	16	14	13	13
Nodes in Clad	3	3	3	1
Nodes in Gap	1	1	1	1
Nodes in Coating	-	-	-	1
Axial Segments	10	10	10	10
Length of Axial Segments in Fuel (mm)	8.12	8.12	8.12	8.12
Length of Axial Segments in Al ₂ O ₃ (mm)	5.08	5.08	5.08	5.08
Coolant Temp. (°C)	245	245	245	245
Thermocouple Hole Length (mm)	78.23	78.23	78.25	78.23
Fill Gas	He	He	He	He

The thermocouple was modeled with the following design parameters:

Thermocouple Design Parameters

Nodes in Sheath	1
Sheath Material	W-Re
Sheath OD (mm)	1.58
Sheath ID (mm)	1.07
Insulator Materials	BeO
Nodes in Insulator	2

TABLE 5.3. TRUMP Predictions for Centerline Temperature
for Three Fuel Types

<u>Case</u>	<u>Centerline Temperature, °C</u>	<u>Hot Radial Gap, mm</u>
Reference	1614	0.043
Reference with Center Hole	1401	0.043
Reference with Thermocouple	1401	0.043
Reference with Thermocouple and Gamma Heating	1405	0.043
Annular	1425	0.062
Annular with Thermocouple	1425	0.062
Annular with Thermocouple and Gamma Heating	1427	0.062
Annular-Coated	1437	0.062
Annular-Coated with Thermocouple	1439	0.062
Annular-Coated with Thermocouple and Gamma Heating	1440	0.062

REFERENCES

- 5.1. Fuel Performance Improvement Program, Quarterly/Annual Progress Report, April-September, 1977, C00-4066-4, November 1977.
- 5.2. Fuel Performance Improvement Program, Quarterly Progress Report, October-December, 1977, C00-4066-6, January 1978.
- 5.3. A. L. Edwards, TRUMP: A Computer Program for Transient and Steady State Temperature Distributions in Multidimensional Systems. UCRL-14754, Rev. 3, Lawrence Livermore Laboratory, Livermore, CA 94550, September 1, 1972.
- 5.4. D. D. Lanning, C. L. Mohr, F. E. Panisko and K. B. Stewart, GAPCON-THERMAL-3 Code Description, PNL-2434, January 1978.
- 5.5. C. R. Hann et al., Test Design, Pre-Characterization and Fuel Assembly Fabrication for Instrumented Fuel Assemblies IFA-431 and IFA-432. BNWL-1988, Battelle, Pacific Northwest Laboratories, Richland, WA 99352, November 1977.
- 5.6. Y. S. Touloukian, Thermal Properties of High Temperature Solid Materials, Volume 1 and 4, MacMillan Co., New York, NY, 1967.

6.0 TASK 2C - IN-REACTOR DEMONSTRATIONS

R. K. Welty (Exxon Nuclear Company)

The objective of this task is to demonstrate commercial acceptability of fuel concepts which have the greatest potential for improved power ramping capability. This will be accomplished through irradiation of a 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 Design and Analysis

6.1.1 Halden Series H-1

The design package (parts list, drawings, and specifications) was issued for Task 2B Halden Series H-1 test fuel rods. Fuel rod assembly drawings appear in Figures 6.1, 6.2, 6.3, 6.4, 6.5, 6.6, 6.7, and 6.8. A parts list revision has been drafted and the final design review meeting has been scheduled to approve the revisions arising during fabrication.

Information obtained during the fabrication of Halden test rods will permit an evaluation of the acceptability of the cladding coating specification. This specification has not yet been

QUANTITY	ITEM	DESCRIPTION	REMARKS
1	1	ASSEMBLY	
2	2	ASSEMBLY	
AR	AR	3 FUEL PELLETS (HIGH ENRICHMENT)	303,123-1
2	2	4 END PELLETS (LOW ENRICHMENT)	303,123-2
1	1	5 CLADDING	303,126-2
1	1	6 LOWER END CAP, ELONGATION SENSOR	303,132-1
1	1	7 LOWER END CAP, PRESSURE SENSOR	303,132-2
1	1	8 UPPER END CAP, STANDARD	303,110
1	1	9 PLENUM SPRING	303,129
2	2	10 INSULATOR DISC	303,165
1	1	11 LOWER END PLUG ASSEMBLY	303,112

- 1 ASSEMBLY, ELONGATION SENSOR
2 ASSEMBLY, PRESSURE SENSOR

General Notes: UNLESS OTHERWISE SPECIFIED

- DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
- NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
- WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
- STAMP SERIAL NUMBER IN .06 (1,52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2,29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4,37) FROM LOWER END OF PART 11.
- PLUG WELD, 1 PLACE, MIN. COVER .02 (0,51), MAX. OVERHANG .005 (0,13).

FIGURE 6.1

APPROVED	REV	REVISIONS	SCALE	DATE	BY
			NONE		

DESIGNED BY: <i>J. J. Ball</i> 11/1/78 COMPONENTS: <i>R. G. Ball</i> 1-7-78 DRAWING CODE: <i>10/1/78</i> RESPONSIBLE DESIGNER: <i>10/1/78</i> CHECKED BY: <i>10/1/78</i> DRAWN BY: <i>10/1/78</i>		EXXON NUCLEAR COMPANY, Inc. HICKLAND, MASSACHUSETTS 01901 REFERENCE FUEL ROD ASSEMBLY XN-NF-303,115
DATE: 11/1/78 SCALE: NONE		

QUANTITY	ITEM	DESCRIPTION	REMARKS
1	1	ASSEMBLY	
2	2	ASSEMBLY	
AR	AR	3 ANNULAR FUEL PELLET (HIGH ENRICHMENT)	303,125-1
2	2	4 END PELLET (LOW ENRICHMENT)	303,123-2
1	1	5 CLADDING	303,126-2
1	1	6 LOWER END CAP, ELONGATION SENSOR	303,132-1
1	1	7 LOWER END CAP, PRESSURE SENSOR	303,132-2
1	1	8 UPPER END CAP, STANDARD	303,110
1	1	9 PLENUM SPRING	303,129
2	2	10 INSULATOR DISC	303,165
1	1	11 LOWER END PLUG ASSEMBLY	303,112

General Notes: UNLESS OTHERWISE SPECIFIED

1. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
2. NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
3. WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
4. STAMP SERIAL NUMBER IN .06 (1,52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2,29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4,37) FROM LOWER END OF PART 11.
5. PLUG WELD, 1 PLACE, MIN. COVER .02 (0,51), MAX. OVERHANG .005 (0,13).

FIGURE 6.2

APPROVED	REV	ADDED PLUG WELD AND NOTE 5	REVISIONS	NONE	DATE	11/1/77	BY	11/1/77
----------	-----	----------------------------	-----------	------	------	---------	----	---------

DESIGNED BY J. E. BULLMAN 11/1/77 CHECKED BY R. E. BULLMAN 11/1/77 DESIGNED BY R. E. BULLMAN 11/1/77 CHECKED BY J. E. BULLMAN 11/1/77 DATE 11/1/77		EXXON NUCLEAR COMPANY, INC. ROCKLAND, MASSACHUSETTS 01866	
DRAWING NO. XN-NF-303,116		SHEET NO. 11

QUANTITY	ITEM	DESCRIPTION	REMARKS
▼	▼	1 ASSEMBLY	_____
		2 ASSEMBLY	_____
AR	AR	3 ANNULAR FUEL PELLET (HIGH ENRICHMENT)	303,125-2
2	2	4 END PELLET (LOW ENRICHMENT)	303,123-2
1	1	5 CLADDING	303,126-1
	1	6 LOWER END CAP, ELONGATION SENSOR	303,132-1
1	7	7 LOWER END CAP, PRESSURE SENSOR	303,132-2
1	1	8 UPPER END CAP, STANDARD	303,110
1	1	9 PLENUM SPRING	303,129
2	2	10 INSULATOR DISC	303,165
1	1	11 LOWER END PLUG ASSEMBLY	303,112

General Notes: UNLESS OTHERWISE SPECIFIED

1. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
2. NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS. TO BE USED FOR REFERENCE ONLY.
3. WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
4. STAMP SERIAL NUMBER IN .06 (1.52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2.29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4.37) FROM LOWER END OF PART II.
5. PLUG WELD, 1 PLACE, MIN. COVER .02 (0.51), MAX. OVERHANG .005 (0.13).

FIGURE 6.3

FIGURE 6.3 1 RIGID PLUG WELD AND NOTE 3	NEW DESIGN & TEST DESIGN <i>1986/11/18</i>	EXXON NUCLEAR COMPANY, Inc. ROCKLAND WASHINGTON 98003	ANNULAR PELLET COATED FUEL ROD ASSEMBLY
	COMPONENT ENGINEER <i>REDAID 1-9-78</i>		
	DE SIGN CHECKED <i>1/6/78</i>		
	RESPONSIBLE DESIGNER <i>1/6/78</i>		
	CHECKED BY <i>1/6/78</i>		
DRAWN BY <i>1/6/78</i>	DATE <i>1-21-78</i>	EXXON NF-303,117	DRAWING NO. NONE
APPROVED <i>1/18/78</i>	REV REVISIONS	DATE NONE	BY NONE

QUANTITY	ITEM	DESCRIPTION	REMARKS
1	1	ASSEMBLY	
2	2	ASSEMBLY	
AR	AR	POWDERED FUEL (HIGH ENRICHMENT)	
2	2	END PELLETT (LOW ENRICHMENT)	303,123-2
1	1	CLADDING	303,126-2
1	1	LOWER END CAP, ELONGATION SENSOR	303,132-1
1	1	LOWER END CAP, PRESSURE SENSOR	303,132-2
1	1	UPPER END CAP, STANDARD	303,110
1	1	PLENUM SPRING	303,129
2	2	INSULATOR DISC	303,165
1	1	LOWER END PLUG ASSEMBLY	303,112

- 1 ASSEMBLY, ELONGATION SENSOR
2 ASSEMBLY, PRESSURE SENSOR

General Notes: UNLESS OTHERWISE SPECIFIED

- DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
- NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
- WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
- STAMP SERIAL NO IN .06 (1.52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2.29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4.37) FROM LOWER END OF PART 11.
- PLUG WELD, 1 PLACE, MIN. COVER .02 (0.51), MAX. OVERHANG .005 (0.13).

FIGURE 6.4

APPROVED	REV	REVISIONS	SCALE	DATE	BY
		1	NONE	8-15-77	ADD 2 PLUG WELDS AND NOTE 5

DESIGNED BY: <i>R. J. Sullivan</i> 1/15/77 CHECKED BY: <i>R. J. Sullivan</i> 1/15/77 DATE: 1/15/77 SCALE: 1/15/77		EXXON NUCLEAR COMPANY, Inc. ROCKLAND, MASSACHUSETTS 01866
PACKED POWDER FUEL ROD ASSEMBLY		
DRAWING NO.: XN-NF-303,118		SHEET NO.: 1 OF 1

QUANTITY	ITEM	DESCRIPTION	DWG. NO. XN-
1	1	ASSEMBLY	
2	2	ASSEMBLY	
10	3	INSTRUMENTED FUEL PELLETS (HIGH ENRICHMENT)	303,124-1
1	4	END PELLETS (LOW ENRICHMENT)	303,123-2
1	5	INSTRUMENTED END PELLETS (LOW ENRICHMENT)	303,124-2
AR	6	FUEL PELLETS (HIGH ENRICHMENT)	303,123-1
1	7	LOWER END CAP, ELONGATION SENSOR	303,132-1
1	8	LOWER END CAP, PRESSURE SENSOR	303,132-2
1	9	UPPER END CAP, THERMOCOUPLE	303,111
1	10	INSULATOR DISC	303,165
1	11	INSTRUMENTED INSULATOR DISC	303,127
1	12	PLENUM SPRING	303,129
1	13	CLADDING	303,126-2
1	14	LOWER END PLUG ASSEMBLY	303,112

- 1 ASSEMBLY, ELONGATION SENSOR AND THERMOCOUPLE
- 2 ASSEMBLY, PRESSURE SENSOR AND THERMOCOUPLE

General Notes: UNLESS OTHERWISE SPECIFIED

- DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
- NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
- WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
- UPPER END AND INSTRUMENT LEADS TO BE MASKED DURING ETCHING FROM CIRCUMFERENCE GROOVE UPWARD (ZONE '2').
- STAMP SERIAL NO IN .06 (1.52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2.29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4.37) FROM LOWER END OF PART 14.

FIGURE 6.5

NAME: <i>Robertson</i> / <i>1/11</i> COMPANY: <i>EXXON</i> / <i>1-5-77</i> DESIGN: <i>Robertson</i> / <i>1/11</i> RESPONSIBLE DESIGNER: <i>Robertson</i> / <i>1/11</i> CHECKED BY: <i>D.J. B. MD</i> / <i>1/11</i> DATE: <i>1-10-77</i>		EXXON NUCLEAR COMPANY, Inc. RICHMOND, VIRGINIA 23261	
SCALE: NONE		REFERENCE FUEL ROD ASSEMBLY (THERMOCOUPLE) XN-NF-303,119	
APPROVED	REV	REVISIONS	DATE

QUANTITY	ITEM	DESCRIPTION	DWG. NO.
1	1	ASSEMBLY	
1	2	ASSEMBLY	
1	3	LOWER END PLUG ASSEMBLY	303,112
1	4	END PELLETT (LOW ENRICHMENT)	303,123-2
1	5	INSTRUMENTED END PELLETT (LOW ENRICHMENT)	303,124-2
AR	6	ANNULAR FUEL PELLETT (HIGH ENRICHMENT)	303,125-1
1	7	LOWER END CAP, ELONGATION SENSOR	303,132-1
1	8	LOWER END CAP, PRESSURE SENSOR	303,132-2
1	9	UPPER END CAP, THERMOCOUPLE	303,111
1	10	INSULATOR DISC	303,165
1	11	INSTRUMENTED INSULATOR DISC	303,127
1	12	PLENUM SPRING	303,129
1	13	CLADDING	303,126-2

- 1 ASSEMBLY, ELONGATION SENSOR
AND THERMOCOUPLE
- 2 ASSEMBLY, PRESSURE SENSOR
AND THERMOCOUPLE

General Notes: UNLESS OTHERWISE SPECIFIED

- DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
- NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
- WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
- UPPER END AND INSTRUMENT LEADS TO BE MASKED DURING ETCHING FROM CIRCUMF. GROOVE UPWARD (ZONE '2').
- STAMP SERIAL N9 IN .06 (1,52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2,29) LONG PRIOR TO WELDING. NOT LESS THAN .172 (4,37) FROM LOWER END OF PART 3

FIGURE 6.6

DESIGNED BY <i>P. Ballou</i> CHECKED BY <i>P. Ballou</i> DATE <i>1/2/78</i>		EXXON NUCLEAR COMPANY, Inc. RICHMOND, VIRGINIA 23261
APPROVED BY <i>P. Ballou</i> DATE <i>1/2/78</i>		
DRAWING NO. NONE		XN-NF-303,120

QUANTITY	ITEM	DESCRIPTION	
1	1	ASSEMBLY	
1	2	ASSEMBLY	
1	3	END PELLETT (LOW ENRICHMENT)	303, 123-2
1	4	INSTRUMENTED END PELLETT (LOW ENRICHMENT)	303, 124-2
AR	5	ANNULAR FUEL PELLETT (HIGH ENRICHMENT)	303, 125-2
1	6	LOWER END CAP, ELONGATION SENSOR	303, 132-1
1	7	LOWER END CAP, PRESSURE SENSOR	303, 132-2
1	8	UPPER END CAP, THERMOCOUPLE	303, 111
1	9	INSULATOR DISC	303, 165
1	10	INSTRUMENTED INSULATOR DISC	303, 127
1	11	CLADDING	303, 126-1
1	12	PLENUM SPRING	303, 129
1	13	LOWER END PLUG ASSEMBLY	303, 112

- ① ASSEMBLY, ELONGATION SENSOR AND THERMOCOUPLE
- ② ASSEMBLY, PRESSURE SENSOR AND THERMOCOUPLE

General Notes: UNLESS OTHERWISE SPECIFIED

- DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
- NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
- WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
- UPPER END AND INSTRUMENT LEADS TO BE MASKED DURING ETCHING FROM CIRCUMF. GROOVE UPWARD (ZONE '2').
- STAMP SERIAL NO IN .06 (1,52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2,29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4,37) FROM LOWER END OF PART 13.

FIGURE 6.7

DESIGNED BY: <i>J. B. Ballman</i> 11/1/78 CHECKED BY: <i>J. B. Ballman</i> 1-7-78 DESIGN CODE: <i>303, 124</i> RESPONSIBLE DESIGNER: <i>J. B. Ballman</i> 1/6/78 CHECKED BY: <i>J. B. Ballman</i> 1/6/78 DRAWN BY: <i>J. B. Ballman</i> 1/6/78 DATE: 1/6/78		EXXON NUCLEAR COMPANY, Inc. RICHMOND, VIRGINIA 23261	
NONE		ANNULAR PELLETT (COATED) FUEL ROD ASSEMBLY (THERMOCOUPLE)	
APPROVED: _____ REV: _____ REVISIONS: _____		XN-NF-303,121 1 0

QUANTITY	ITEM	DESCRIPTION	DWG. NO. XN-
1	1	ASSEMBLY	
1	2	ASSEMBLY	
AR	AR	3 PACKED POWDER (HIGH ENRICHMENT)	
1	4	END PELLET (LOW ENRICHMENT)	303,123-2
1	5	INSTRUMENTED END PELLET (LOW ENRICHMENT)	303,124-2
1	6	THERMOCOUPLE WELL	303,109
1	7	LOWER END CAP, ELONGATION SENSOR	303,132-1
1	8	LOWER END CAP, PRESSURE SENSOR	303,132-2
1	9	UPPER END CAP, THERMOCOUPLE	303,111
1	10	INSULATOR DISC	303,165
1	11	INSTRUMENTED INSULATOR DISC	303,127
1	12	PLENUM SPRING	303,129
1	13	CLADDING	303,126-2
1	14	LOWER END PLUG ASSEMBLY	303,112

- 1 ASSEMBLY, ELONGATION SENSOR
AND THERMOCOUPLE
- 2 ASSEMBLY, PRESSURE SENSOR
AND THERMOCOUPLE

General Notes: UNLESS OTHERWISE SPECIFIED

1. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ANSI Y14.5, 1973.
2. NUMBERS SHOWN IN PARENTHESIS ARE MILLIMETERS, TO BE USED FOR REFERENCE ONLY.
3. WELD MAY BE DRESSED TO MEET OVERHANG REQUIREMENT.
4. UPPER END AND INSTRUMENT LEADS TO BE MASKED DURING ETCHING FROM CIRCUMFERENCE GROOVE UPWARD (ZONE "2").
5. STAMP SERIAL NR IN .06 (1,52) HIGH CHARACTERS AND ORIENTATION MARK .09 (2,29) LONG PRIOR TO WELDING, NOT LESS THAN .172 (4,37) FROM LOWER END OF PART 14.
6. PLUG WELD, 1 PLACE, MIN. COVER .02 (0,51), MAX. OVERHANG .005 (0,13).

FIGURE 6.8

APPROVED	REV	REVISIONS	SCALE	NONE	DRAWING NO.	XN-NF-303,122	REV	1	1
----------	-----	-----------	-------	------	-------------	---------------	-----	---	---

DESIGNED BY <i>J. J. O'Brien</i> 11/1/78 COMPONENT ENGINEER R. G. Hall 11-2-78 CHECKED BY <i>J. J. O'Brien</i> 1/6/79 RESPONSIBLE DESIGNER DATE 1/6-81 1/5/79 DRAWN BY <i>J. J. O'Brien</i> 8-10-77		EXXON NUCLEAR COMPANY, INC. BOSTON AND WASHINGTON, D.C.	
Packed Powder Fuel Rod Assembly (Thermocouple)		XN-NF-303,122	

incorporated into the design package for the Big Rock Point Reactor Series S-1 demonstration fuel rods.

Several minor design changes were incorporated into Halden Series H-1 test rods including:

- 1) A vent hole was added to the lower end plug, XN-NF-303,113, to relieve pressure buildup during end cap welding of the vipac rods. The hole will be sealed with a plug weld after lower end cap welding.
- 2) A vent hole was added to the upper standard end cap, XN-NF-303,110, for the same reason as item 1 above.
- 3) The thermocouple well in vipac rods was shortened from 139.7 mm to 97.5 mm (5.5 to 3.84 inches) length. Sufficient Halden thermocouple upper end caps were not received with a hole which would allow mating with the well. The upper end of the well will now reside about 25 mm (1 inch) below the upper end cap.

6.1.2 Big Rock Point Series S-1

Design analyses are continuing to support licensing of FPIP fuel rods in the G-4 reload batch for Big Rock Point Reactor. Monte Carlo simulations of the various fuel rod configurations were conducted to determine local power peaking associated with segmented rods and the various fuel forms. Results will now be used in safety analyses and thermal-hydraulic design calculations.

A parts list revision for Big Rock Point Reactor G-4 was drafted, approved, issued, and sent to CPC for approval. Appendix B of the revised parts list contains details for fabrication and loading of Task 2C demonstration fuel rods into four G-2 assemblies.

6.2 Fuel Process Development

6.2.1 Pellet Process Development

The UO_2 for Halden Series H-1 arrived from ORNL in three sublots of varying U-235 enrichments, 10.86%, 10.95%, and 11.15%. Originally it was planned to use the 10.86% enrichment for reference pellets and the 11.15% enrichment for annular pellets. However, due to powder variability within the sublots, a master blend of the two was made with a resulting enrichment of 11.023% U-235. This lot, FPIP-001, was used to prepare reference and annular pellets.

Fabrication of reference and annular pellets was completed. The reference pellets were accepted through Q.C. release with no difficulties, but the first lot of annular pellets experienced a high frequency of cracking during outgassing. The annular pellets were divided into two lots, Rev. 0 and Rev. 1. The Rev. 1 pellets were sintered in two stages, 5.5 hours at 1700°C and then 11 hours at 1700°C . The Rev. 0 pellets were sintered once for 11 hours at 1700°C . Essentially all of the cracked pellets were associated with the single Rev. 0 sintering cycle. Another lot was made to replace the broken pellets and sintered in the two step sintering cycle (Rev. 1). No broken pellets occurred during outgassing.

6.2.2 Packed Particle (Vipac) Fuel Process Development

The 10.95% U-235 enriched UO_2 subplot was Dynapaked for use as feed material for the vipac fuel rods. After Dynapaking, the fuel was sintered using a sintering cycle of 24 hours at 1700°C . The fuel particles from this run had low density (approximately 97% T.D.) and were very porous in nature. As a result, this material was re-Dynapaked.

Sixty-nine packed particle fuel rods were prepared in developing loading techniques to achieve uniform and suitably high average fuel

smear densities. The most successful technique was a continuous feed at low vibrational energies followed by a brief vibrational period at high energies with the follower rod in place. Typical rod smear densities were:

#66	85.33% T.D.
#67	86.00% T.D.
#68	85.65% T.D.
#69	86.10% T.D.

The gamma densitometer has been set up to check density uniformity along the length of the rods. The best estimate at this time is that the rods can be loaded to a standard deviation of $\pm 2\%$ of the nominal density. The standards for the gamma densitometer were prepared and used for calibration of the instrument.

Cladding strain measurements taken during test compaction of vipac fuel columns indicated high stresses. Process acceptance criteria in Product Specification XN-S30685 were revised to allow acceptance based on tests with a conservative safety factor of total cycles during compaction. No cladding, end cap or weld cracks occurred after an order of magnitude more compaction cycles than permitted during rod fabrication.

6.2.3 Graphite-Coated Cladding Process Development

Tests were completed which were designed to evaluate the effect of graphite coating moisture levels on cladding performance. Short length test fuel rods were fabricated with either graphite-coated or standard, uncoated cladding and loaded with a glass ampule that contained various amounts of moisture (Table 6.1). The rods were heated to 450°C for 24 hours and then pinched against two mandrels to deform the cladding and rupture the internal film (Figure 6.1). The test rods were reheated to 450°C for over 100 hours to determine if hydriding of the cladding would occur.

TABLE 6.1
TEST FUEL ROD MAKEUP

<u>Test Rod. No.</u>	<u>Coating Thickness (μm)</u>	<u>Water Added (mg)</u>	<u>Fuel Weight (gm)</u>	<u>Void Volume (cc)</u>
<u>Coated</u>				
14	10-13	--	117.7	1.964
16	11-12	0.58	118.7	1.531
13	10-13	0.63	123.0	1.151
21	14-16.5	2.83	122.2	1.484
9	19-20	14.37	117.9	1.554
12	15.5-19	14.51	112.7	1.108
<u>Uncoated</u>				
14	--	--	120.4	1.169
27	--	0.81	118.6	1.688
30	--	0.87	119.7	1.553
28	--	3.17	124.4	1.250
29	--	15.67	121.8	2.207
22	--	15.87	118.4	1.696

Hydriding was observed (Table 6.2) on only the two test rods with the highest amount of moisture (Figures 6.9, 6.10, 6.11, 6.12, and 6.13). These rods contained over 100 ppm moisture on a fuel weight basis. A test rod with about 15 ppm moisture (50% above current specification limits) showed no hydriding. Other test rods were not hydrided, but may have failed mechanically during the pinching operation and released their hydrogen. Apparent through-thickness cracks were seen in some metallographic specimens.

The inside surface of cladding tubes was coated with graphite and the thickness was measured with the air gauge system set up in the Process Engineering Laboratory. The test results were acceptable and the production tubes were coated and are ready for vacuum outgassing. The outgas system was checked with several coated tubes to determine the extent of zirconium oxide film formation under the graphite coating during outgassing and total residual moisture in the graphite. Results were acceptable, and tubes with optimum film thickness were selected for loading.

6.2.4 Welding Process Development

Weld acceptance was based on radiography. A minimum weld cover equal to 80% of nominal wall thickness and a 2.5 mm (0.010 inch) limit on void size was allowed. Welding difficulties continue and require careful analysis of the radiographic indications to assure joint integrity. A fatigue crack growth analysis indicates that current acceptance criteria are quite conservative.

The lower end plugs and caps were welded on all 18 fuel pins except for the vipac tubes, which only have the lower end plug welded on. The upper end caps were welded on the five reference fuel pins with radiographic and metallographic analyses being performed on all welds. Several weld defects were dispositioned. Pin H1LCR002 was rejected (weld gap) and a new

TABLE 6.2
TEST FUEL ROD RESULTS

<u>Test Rod Number</u>	<u>Rod Moisture Content (mg/cc cold void)</u>	<u>Equivalent Fuel Moisture Content (ppm)</u>	<u>Hydride Indications</u>
<u>Coated</u>			
14	0.62	10.3	None*
16	1.14	15.2	None*
13	1.62	15.2	None*
21	2.73	33.2	None*
9	10.03	132.3	None
12	14.19	139.5	Slight
<u>Uncoated</u>			
19	<0.21	<2.0	None*
27	0.62	8.8	None*
30	0.71	9.3	None*
28	2.74	27.5	None*
29	7.21	130.7	Extensive
22	10.08	144.4	Leaked after deformation

*May have failed mechanically during deformation, allowing hydrogen to escape.

FIGURE 6.9 Test Rod Configuration (Rod No. 29)

FIGURE 6.10 Hydride Layer in Deformed Region of Test Rod No. 29

FIGURE 6.11 Hydride Layer in Crack in Test Rod No. 29

FIGURE 6.12 Electrolytically Charged Hydride Layer on Zircaloy Cladding (6 hr at 100 mA/cm² in 0.1 N H₂SO₄)

FIGURE 6.13 Slight Evidence of Hydride Layer in Test Rod No. 12

assembly was built. Pins H1LCA007, H1UCR003, and H1UCR004 were accepted after a fatigue crack analysis showed that voids (cracks) will not penetrate 0.64 (0.025 inch) in 1000 thermal cycles.

6.2.5 Documentation

The Process Test Authorization-Halden (PTA) was reviewed, approved, and issued. This document is the controlling document for fabrication of the fuel. The design characteristics for the Halden rods are summarized in Table 6.3.

TABLE 6.3

DESIGN CHARACTERISTICS OF TEST PINS

PTA-230

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.	83.6	80.3	87.0	80.5
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.

7.0 TASK 3 - FUEL PERFORMANCE EVALUATION

M. D. Freshley and R. E. Schreiber (Pacific Northwest Laboratory)

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

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

An evaluation of the operating history requirements for the BRP demonstration irradiations is being made. It is planned to supplement the existing computerized data collection system at BRP with additional data that will most likely have to be obtained by reviewing the operational records and expanding the normal fuel management calculations. The additional information can include such things as:

- Once a week for thermal power if plant is operating at essentially constant power.
- Up ramps of power reported according to category of slow, moderate, or fast ramp rate, unless they exceed 3%/hour, in which case actual rates are required.
- Down ramps of power, including scrams, reported according to category of slow, moderate, or fast.
- Core flux maps at least weekly, if feasible.
- Offgas activity increase reported in terms of number of estimated rods with pin hole defects, if possible.

- Coolant flow and temperature data for FPIP channels reported at least as frequently as axial flux profiles (core maps).
- Water chemistry and coolant activity reported at least twice a week if offgas indicates suspected defects. Once a week if no defects.
- Report any abnormal occurrences affecting core operation.

8.0 TASK 4 - ESTABLISHMENT OF TECHNICAL BASES

K. R. Merckx (Exxon Nuclear Company)

A preliminary version of the Technical Bases for the Big Rock Point FPIP Fuel has been distributed within the project for comment. This version includes sections on annular, packed particle, and graphite lubricated fuel as well as the reference fuel. The sections on the new fuel concepts proposed design criteria were not supported with the appropriate technical bases in some cases; hence, revision and addition of supporting bases can still be expected before general distribution will be approved.

8.1 Data Requirements

The performance file for the new concept fuels has been expanded to include information generated specifically from the Big Rock Point FPIP fuel irradiations. A proposed irradiation history data content has been included in this file structure which is to be generated by computer automated file treatment programs using fuel management information as the main input data source. The programs generating this data have been written and a report describing their content and operation has been submitted for project distribution. Actual performance data from the literature must now be collected to serve as existing bases for code verification of the performance models for the new fuel concepts.

APPENDIX A
SI UNIT CONVERSION TABLE

CONVERSION TABLE FOR SI UNITS^(a)

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

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

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

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

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

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