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**POWER REACTOR
HIGH PERFORMANCE UO_2 PROGRAM
PROGRESS REPORT NUMBER 9
OCTOBER 1, 1968 - MARCH 31, 1969**

COMPILED BY
S. Y. OGAWA

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3) - 189
PROJECT AGREEMENT 50

NUCLEAR FUELS DEPARTMENT
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA

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POWER REACTOR HIGH PERFORMANCE
UO₂ PROGRAM

Progress Report Number 9
October 1, 1968—March 31, 1969

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U. S. Atomic Energy Commission
Contract AT(04-3)-189
Project Agreement 50

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FOREWORD

The United States and the European Atomic Energy Community (EURATOM) on May 29, and June 18, 1958, signed an agreement which provides a basis for cooperation in programs for the advancement of the peaceful applications of atomic energy. This agreement, in part, provides for the establishment of a joint U.S. - EURATOM research and development program which is aimed at reactors to be constructed in Europe under the Joint Program.

The work described in this report represents the Joint U.S. - EURATOM effort which is in keeping with the spirit of cooperation in contributing to the common good by the sharing of scientific and technical information and minimizing the duplication of effort by the limited pool of technical talent available in Western Europe and the United States.

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ABSTRACT

The complete results of the hot-cell examinations performed on eight selected fuel rods, and of the thermal-hydraulic test performed on a 3×3 , 9-rod mock-up of the 7×7 Advanced Performance Fuel assembly corners are presented. The performance of the fuel rods from the standpoint of operation with UO_2 incipient and moderate central melting was very satisfactory. Specifically, it was shown that the volume expansion on UO_2 melting was adequately accommodated by the design, and that the degree of melting that occurred corroborated the design predictions very well.

The results of the examinations show that the failures appear to be strictly from causes external to the fuel rod and, especially, not a direct result of the centermelt operation. The evidence, particularly the incipient rod failure, suggests a localized cladding overheating with the high heat flux, crud deposition and unique thermal-hydraulic conditions downstream of the lateral spacing dimples on the corner angle all contributing in some manner not yet completely understood.

The out-of-pile loop tests designed to investigate the failure mechanism did reveal a local, spot overheating phenomenon that could account for the failures through accelerated cladding corrosion over extended periods of operation. At this time, it appears prudent to avoid a firm conclusion since the local, spot overheating did not occur at exactly the same positions in the loop test as the observed in-pile cladding deteriorations, and extrapolations must be made from the idealized loop test conditions to the actual reactor environment. Since relatively high heat flux is one of the governing factors leading to the failures in the rods located in the unique bundle corner positions, the straightforward corrective action to avoid similar failures is to lower the power of these rods. The plan is to place very low power-producing rods (containing depleted UO_2) in the critical positions by rearrangement of rods within each assembly. Eight new rods with lower enrichment (2.44%) will be inserted as part of the correction.

Fuel assembly D-50 (0.570 inch o.d., pellet UO_2 fuel) presently operating in the BRP core is a suspect failure, presumably from the same localized cladding overheating under the corner angle that affected the two Advanced Performance powder assemblies earlier. Examination of this assembly at the site is scheduled during the next refueling shutdown starting on April 21, 1969.

The information and data contained in this report were submitted in a separate report to the AEC-DRL in compliance with the license pursuant to obtaining authorization to resume the irradiation of the four Advanced Performance Fuel assemblies.

SECTION I

INTRODUCTION

A research and development program was initiated on September 1, 1966, under Project Agreement 50 of Contract AT(04-3)-189 to irradiate a significant number of Zircaloy-2-clad UO_2 fuel rods to varying degrees of central melting in an operating power reactor. This is a cooperative program involving the AEC, Euratom, Consumers Power Company, and the General Electric Company. The program can be thought of as an extension of an earlier AEC-Euratom high-performance

UO_2 program (Contract AT(04-3)-189, Project Agreement 17), which demonstrated that there is no intrinsic life-limiting mechanism that will prevent the attainment of high burnup in fuel rods operating with central melting. The AEC-Euratom portion of the program covers the detailed fuel design, including safeguards analysis and reactor license changes necessary to their irradiation, indirect fabrication supervision, irradiation following, and post-irradiation examination and reporting.

Six assemblies were planned for irradiation in the Consumers Big Rock Point reactor to an average exposure of 15,000 MWd/t.* Two of the assemblies were designated Intermediate Performance Fuel, and will operate with incipient melting in the central region of the UO_2 at rated power conditions. These assemblies might be considered the next step toward the elimination of the central melt criterion used as a limit in current fuel design practice. The remaining four assemblies were designated Advanced Performance Fuel, and will operate with definite but moderate central melting at rated power conditions. These advanced performance assemblies represent a further step in reactor technology wherein the hottest fuel rods in the core are operating with moderate amounts of the UO_2 in the molten state at rated power conditions.

The detailed description of the center-melt fuel design have been presented in previous reports. A summary of the design is shown in Appendix A.

This is the ninth progress report covering the period October 1, 1968 through March 31, 1969. Presented in this report are the following:

- a. The complete results of the hot-cell examinations performed on 8 fuel rods at the Vallecitos Nuclear Center.
- b. A description of the thermal-hydraulic loop tests performed on a 3×3 , 9-rod mock-up of the 7×7 Advanced Performance fuel assembly corner, and the results obtained from these tests.

The information and data contained in this report were generated during this reporting period and were submitted to the AEC-DRL through the Consumers Power Company,

Jackson, Michigan, in partial fulfillment of the requirements stipulated in the license which first permitted the start of the irradiation on the six centermelt test assemblies.

The initial operation of centermelt assemblies in the Big Rock Point reactor was authorized by Amendment No. 1 to the BRP Facility Operating License No. DPR-6. The first authorization was for a relatively short operating period equivalent to exposure accumulation of approximately 15 percent of the fuel lifetime, or $\sim 3,000$ MWd/T. After this period of operation, the assemblies were to be nondestructively inspected at the site. The two Intermediate Performance fuel assemblies* were permitted to be returned to the core for continued irradiation if they were found to be in satisfactory condition. The remaining four Advanced Performance fuel assemblies* were to be left out of the core until destructive, post-irradiation examinations could be completed on selected fuel rods to verify that the very high-power fuel rods could be operated further with safety. The occurrence of two unexpected and unique fuel rod failures during the first operating cycle imposed additional examination and analysis requirements to determine the cause of failure. A satisfactory explanation for the failures was a prerequisite to developing a technically sound fix to the fuel assemblies and re-establishing the feasibility of continued irradiation.

The authorization to resume the irradiation of the four Advanced Performance bundles was reserved to the AEC-Division of Reactor Licensing. Their decision is to be based on a review of data and performance evaluations supplied by General Electric and Consumers Power Company responsive to their expressed concerns regarding the safety in continued operation. This document

* Megawatt days per short ton of uranium metal

* See Appendix for a summary of the fuel design.

contains the aforementioned examination data and performance evaluations and was submitted pursuant to obtaining authorization to resume the Advanced Performance bundle irradiations. Two specific areas are covered: a) performance evaluation of the fuel rods to show that the degree of UO_2 central melting and the behavior of the molten UO_2 was in conformance to the fuel design, and b) analysis and interpretation of the unique failures that occurred. In addition, the results from a loop thermal-hydraulic test and other supplemental analyses pertinent to the failure mechanism are discussed. A rearrangement of rods in the fuel assemblies to prevent the same kind of failures from occurring is planned so that the assemblies are indeed ready for further irradiation.

Prior reports issued to the Commission under this contract have included the following:

1. GEAP-5402, December 1966, Progress Report No. 1, July 1 through October 31, 1966, W. H. Arlt, et al.
2. GEAP-5431, February 1967, Progress Report No. 2, November 1 through December 31, 1966, E. A. Lees.
3. GEAP-5487, April 1967, Progress Report No. 3, January 1 through March 31, 1967, E. A. Lees.
4. GEAP-5521, July 1967, Progress Report No. 4, April 1 through June 30, 1967, E. A. Lees.
5. GEAP-5585, January 1968, Progress Report No. 5, July 1 through December 31, 1967, compiled by S. Y. Ogawa.
6. GEAP-5591, January 1968, Fuel Design Summary and Program Status, S. Y. Ogawa, E. A. Lees, M. F. Lyons.
7. GEAP-5605, April 1968, Progress Report No. 6, January 1 through March 31, 1968, compiled by S. Y. Ogawa.
8. GEAP-5661, July 1968, Progress Report No. 7, April 1 through June 30, 1968, compiled by S. Y. Ogawa.
9. GEAP-5751, October 1968, Progress Report No. 8, July 1—September 30, 1968, compiled by S. Y. Ogawa.

SECTION II

SUMMARY

Detailed post-irradiation hot-cell examinations were performed on eight centermelt fuel rods returned to the Vallecitos Nuclear Center. The scope of the examinations (Table 1) was designed for two purposes: (1) to show that the degree of UO_2 central melting and the behavior of the molten UO_2 was predictable, thereby confirming the fuel design calculations, and (2) to determine the reason for the two rod failures that did occur.

A fix has been determined for the centermelt assemblies to avoid any similar

failures in the corner positions. By a combination of new rod additions and position rearrangements, the power producing rods under the corner angle are eliminated, putting the assemblies in a condition that insures reliable continued irradiation.

2.1 DESIGN CONFIRMATION

The results of the examinations showed that the performance of the fuel rods from the standpoint of operation with UO_2 central melting was very satisfactory. Specifically, it was shown that the volume expansion on UO_2

melting was adequately accommodated by the design, and that the degree of melting that occurred corroborated the design predictions very well. These conclusions are based on the following experimental observations:

- Measurements of the fuel rod diameters by profilometry showed no interaction between the UO_2 and the cladding.
- The molten radii determined from metallographic examination of fuel rod cross sections were very close to those predicted by design (Table 5, Figures 21, 22, 23, 24, 25). This good correlation was also shown by a comparison of the heat fluxes calculated using the experimental data with heat fluxes calculated from reactor operating conditions (subsection 8.1 and Figure 33). The heat fluxes were in good agreement, within 5 percent.
- Full length neutron radiographs (Figures 5 to 9) showed the void distributions in the UO_2 and the condition of the cladding. Thermal calculations were performed for the highest power pellet and powder Advanced Performance fuel rods to reconstruct the behavior of the UO_2 during actual rod operation which led to the void distribution in the UO_2 fuel column observed in the neutron radiographs, (subsection 8.2). The void distribution and fuel redistribution were found to be reasonably predictable, using a relatively simple model and the calculational methods used for the design. (Refer to the gamma scans in Figures 16 and 17, and to Figures 41 and 42.) The cladding of the intact fuel rods showed no signs of hydriding in the neutron radiographs; i. e., no evidence of deterioration from internal attack mechanisms.

2.2 EXAMINATION OF THE FUEL ROD WITH A SURFACE PROTRUSION FROM BUNDLE D-51

The examination on this rod (CP-409, a 0.570-inch o.d. compacted powder UO_2 rod from Assembly D-51) confirmed that the surface protrusion observed at the axial mid-point was actually a piece of tramp material caught beneath the spacer spring. The neutron radiograph (Figure 7) showed no cladding deterioration whatsoever at the location of the deposit.

2.3 FAILURE EXAMINATIONS

2.3.1 Visual

Examination of the bundles at the reactor site revealed two definitely failed rods and one incipient failure. All three rods were from the Advanced Performance powder fuel assemblies, with the two definitely failed rods being from the identical local position in each of the two powder rod bundles. This local position was in the hot corner of the bundle on the side closest to the center of the reactor core and shrouded by the bundle corner angle support. The incipient rod failure was also found in the hot corner of bundle D-54, in the other rod position shrouded by the corner angle on a diagonal from the failed rod.

The exact locations of the cladding defects, the areas of crud spalling and cladding corrosion, and the incipient failure were determined during the detailed examinations of the failed rods. These features were consistently located slightly downstream from the lateral spacing dimples embossed in the corner angle, on the side of the fuel rod facing the corner angle (Figures 2, 3, and 4).

2.3.2 Destructive

Cross sections were made through the defects in the two definitely failed rods

(0.700-inch o.d. powder UO_2 rods CM-401 and CM-412) and through the incipient failure region of the other rod (powder rod CM-409).

The defect in rod CM-401 was associated with a localized cladding bulging. This bulging was found to be the result of the formation of zirconium oxide on the inside cladding surface (Figures 26 and 28). The cladding was cracked in this region, but the exact nature or sequence of events leading to this defected condition could not be determined. The UO_2 in this section appeared to be unaltered.

The main defect in rod CM-412 was associated with excessive corrosion resulting in a hole of a diameter equal to approximately one-sixth of the cladding circumferences (Figure 27). Despite this opening, the UO_2 remained relatively intact, with a maximum loss of from 0.5 to 1 gram (estimated) of UO_2 from the region immediately adjacent to the hole in the cladding. The UO_2 in this cross section had been altered considerably (absorbed oxygen) (Figure 28b). Although the UO_2 structure does not show a distinct molten boundary, it suggests that melting had extended very close to the UO_2 surfaces facing the cladding defect. This conclusion takes into consideration the fact that at the given maximum heat flux, both the thermal conductivity and melting point of the UO_2 were lowered by its high oxygen content. No drastic effects are apparent, although grain growth in the cladding around this defect indicates high cladding temperatures. The neutron radiographs of these two defected rods (Figures 10, 11) showed extensive cladding hydriding from the inside surface in the lower regions of both rods. The resulting embrittlement probably contributed to the secondary defect observed in rod CM-412 near the lower end plug. The hydriding undoubtedly resulted from water entering the lower part of the rod after the defects further up on the rods had occurred. Hydriding from the outside was observed on

the upper fuel rod surfaces. The regions affected in this manner coincided with the crud spalled and corroded areas observed during the overall visual examination.

The cross section through the incipient failure (rod CM-409, Figure 29) showed that the cladding wall had been thinned locally by accelerated corrosion. The grain growth (from ASTM 11-12 to ASTM 9-10) in the Zircaloy-2 cladding structure showed that unusually high cladding temperatures were experienced. The temperature in this region was estimated to be 1400°F . Since no beta transformation structure was seen, the cladding temperature definitely did not exceed 1500°F .

The neutron-radiograph of this incipient failure rod (Figure 12) showed that it was otherwise in satisfactory condition. The void distribution in the UO_2 was similar to the other intact powder UO_2 rod examined and the cladding was free from hydriding, except for a tiny external hydriding zone at the incipient failure region.

2.3.3 Failure Interpretation

The destructive examination of the fuel rods essentially verified the tentative conclusions regarding the failure mechanism developed earlier from the reactor site exams. The failures appear to be strictly from causes external to the fuel rod and, especially, not a direct result of the center-melt operation. The evidence, particularly the incipient rod failure, suggests a localized cladding overheating with the high heat flux, crud deposition and unique thermal-hydraulic conditions downstream of the corner angle lateral spacing dimples all contributing in some manner not yet completely understood.

2.4 THERMAL-HYDRAULIC LOOP TEST

A loop test of a 3×3 , 9-rod mock-up of the 7×7 Advanced Performance Fuel

Assembly corner was performed to investigate the thermal-hydraulic characteristics of this unique bundle configuration. An exact duplicate of the bundle corner geometry was used. This was a refinement in the test simulation that was not possible when the original loop thermal-hydraulic tests were performed at the beginning of the program, as part of the bundle design effort.

The local conditions that prevailed at the point of failure during actual reactor operation; i.e., below the middle of the rod facing the corner angle, were duplicated as closely as possible in the specific test section of the mock-up. Details on the thermal-hydraulic test are provided in Section VII.

Critical heat flux (CHF) conditions were not obtained at the duplicated rated reactor conditions. Normal CHF could be induced, however, at much higher heat fluxes and qualities. The test results, interpreted by a multi-channel hydraulic analysis and extrapolated to the actual reactor operating condition, showed that the minimum critical heat flux ratio (MCHFR) in the reactor was about 1.9* (Figure 33).

* Relative to experimental data obtained in the 3 x 3, 9-rod loop test rather than referenced to the usual data correlation lines.

Further, the test results showed that the multi-channel hydraulic analysis computer code reliably predicted flow and quality distribution for very high relative rod powers, and thus the code is expected to accurately predict CHF for normal geometries where standard correlations apply. However, the multi-channel model does not account for very localized perturbations (such as lateral spacer dimples).

Small areas ($1/64$ inch²) of discoloration were observed on the heater surfaces immediately downstream from the corner angle dimples. This discoloration indicated very local overheating of the surface. A thermocouple located in one of these areas detected a temperature oscillation of 15°F. However, thermocouple transient response studies indicate that at the heater surface directly above the thermocouple, the amplitude of the temperature oscillations may be as high as 75°F. Temperatures much higher than that had most assuredly been reached (as indicated by the dark color of the spots) but not detected, due to the thermocouple being offset laterally about 50 mils from the center of the spot, and also possibly due to the slight movement of the hot spot with different conditions. The long term effect of this local over-temperature phenomenon on Zircaloy-2 corrosion could be an explanation for the observed in-pile failures.

SECTION III

CONCLUSIONS

The performance of the centermelt fuel rods has been demonstrated to be completely satisfactory for operation with incipient and moderate UO₂ central melting.

The detailed post-irradiation examinations have confirmed a high degree of accuracy

for: (a) the design calculations that predicted the amount of central melting and the behavior of the molten UO₂, and (b) the physics predictions of relative and absolute rod powers.

Complete design accommodation of the UO₂ phase change volume expansion on melting

was demonstrated. Power cycles from melting-to-solid back melting were experienced with no adverse effects.

Thus, with respect to centermelt concerns, per se, the fuel met design expectations and its irradiation can be safely continued to achieve the original high fuel exposure targets.

However, two unexpected and unique fuel failures occurred during the first irradiation cycle. The cause of the failures is now attributed to localized accelerated corrosion of the Zircaloy-2 cladding. The driving force for this corrosion is local cladding overheating caused by very local flow disturbances associated with the dimples on the corner angle, a configuration unique to the centermelt assemblies. The exact thermal-hydraulic mechanism involved in the perturbation is not known. It is clear that the failures are not caused by exceeding critical heat flux (CHF) limits, as defined in the usual sense, since this would cause much higher cladding temperatures and very rapid cladding destruction.* The overheating observed is more nearly characterized by the concept of local flow stagnation or the existence of an insulating blanket at all heat fluxes rather than the sudden CHF destruction of the cooling film which is normally regarded as a threshold phenomenon.

The out-of-pile loop tests designed to investigate the failure mechanism did reveal a local, spot overheating phenomenon that could account for the failures through accelerated cladding corrosion over extended periods of operation. At this time, it appears prudent to avoid a firm conclusion

since the local, spot overheating did not occur at exactly the same positions in the loop test as the observed in-pile cladding deteriorations and extrapolations must be made from the idealized loop test conditions to the actual reactor environment. For example, the loop test runs were short (< 1 hour for each run) compared to the actual fuel rod operating time at full power before failure (13 days) and more highly corrosion-resistant (at elevated temperatures) nickel sheathing was used on the heater rods rather than Zircaloy-2. Thus, an accelerated corrosion mechanism related to local thermal-hydraulic conditions might well not have been manifested in the loop. In addition, the loop environment was extremely clean so there was no possibility of aggravation from local crud build-up; e.g., by "ski-jump" deposits such as observed in-pile. Keeping in mind these limitations, the occurrence and detection of the spot overheating is still considered very significant and the phenomenon is probably involved in the failures.

Since relatively high heat flux is one of the governing factors leading to the failures in the rods located in the unique bundle corner positions, the straightforward corrective action to avoid similar failures is to lower the power of these rods. The plan is to place very low power-producing rods (containing depleted UO_2) in the critical positions by rearrangement of rods within each assembly. Eight new rods with lower enrichment (2.44 percent) will be inserted as part of the correction (Section IX).

Analyses were made for the bundles with the revised rod arrangement and confirmed that the bundles can be operated within reactor license limits. There is ample grounds for confidence in the ability of the multi-channel hydraulic analysis code to accurately predict the local (subchannel) flow and quality for the new configuration. This ability was demonstrated

* Normal CHF is defined as the heat flux at which the heat transfer mechanism breaks down with temperature oscillation beyond normal operating levels. This specifically excludes time dependent processes.

by the accurate predictions obtained from the hydraulic model for the loop test results measured on the 9-rod mock-up.

To recapitulate, the destructive rod examinations and actual in-pile failure experience have alleviated concerns regarding continued safe operation with central melting. This evidence combined with the proposed correction to the rod arrangement to avoid additional corner rod failures provides good confidence that the bundle irradiations can be resumed with high

expectations of reliable operation. Authorization was solicited from AEC-DRL to so proceed, with final approval being made contingent on the results from the site examination of bundle D-50. Bundle D-50 is presently operating in the BRP core and is a suspect failure, presumably from the same localized cladding overheating under the corner angle that afflicted the two Advanced Performance powder bundles earlier. If bundle D-50 is failed at some other location or for clearly different cause, irradiations will be further deferred until bundle D-50 can be carefully evaluated.

SECTION IV

SUMMARY OF OPERATION

This section covering the first three month's operation was part of the document submitted to the AEC-DRL, was covered in detail in the previous progress report (No. 8), and therefore is not repeated here.

Assembly D-50 was the only one returned to the core for continued irradiation. The operation of assembly D-50 (0.570-inch o.d., pellet fuel) was resumed when the BRP reactor was started up on July 16, 1968, after a refueling shutdown. At full reactor power (75 MWe), the peak heat flux on D-50 was 425,000 Btu/h-ft². It was operated essentially at this power level till October 24, 1968, when significant increase in the

stack off-gas activity again signalled the occurrence of fuel failure. Reductions in the power from the core region in which D-50 is located were accompanied by reductions in the stack off-gas activities. In November, the power in D-50 was reduced about 15 percent.

In mid January 1969, an increase in off-gas activity at reduced reactor power (60 MWe) occurred. As a result, the control rod next to D-50 was inserted slightly more than half-way, and a sharp decrease in the released activity by a factor of about four resulted. The assembly is presently operating with a peak heat flux of ~ 250,000 Btu/h-ft².

SECTION V

SITE EXAMINATION

This section was also a part of the document submitted to the AEC-DRL and was covered in detail in the previous progress report (No. 8), and therefore is not repeated here.

The shipments of the fuel rods selected for the post-irradiation examinations from the Big Rock Point reactor site to the Vallecitos Nuclear Center were completed on November 11 (700 EX cask) and on November 14 (TREAT-2 cask), 1968.

SECTION VI

POST-IRRADIATION EXAMINATIONS

6.1 SCOPE OF WORK

The scope of work performed for each of the eight fuel rods is shown in Table 1. The results from each of the types of examination are discussed in the following sections.

6.2 VISUAL INSPECTION

Each fuel rod was visually inspected over the entire surface, using the hot cell periscope. The intact fuel rods CM-224, CM-505, CP-511 and CP-307 had no unusual surface features, and were in very good condition.

The surface protrusion on rod CP-409 had been partially removed when viewed at this time. This was the result of the large amount of handling that the rod received since leaving the reactor fuel storage pool. A comparison of the feature in the pool and in the hot cell is shown in Figure 1. The rod was otherwise in very good condition.

The examination of the two failed rods, CM-401 and CM-412, and the incipient failure rod, CM-409, established the axial locations of the discrete crud spalled areas and the specific areas of severe cladding deterioration in relation to the dimples in the corner angle piece (see Appendix for locations of dimples). The photographs in Figures 2, 3, and 4 show the overall condition of the rod surfaces. The locations of the corner angle dimples and the spacers have been included. It should be noted that only the rod surfaces that faced the corner angle were affected. The position where cross sections were prepared and examined are noted for reference.

Specific features to be observed on each of the rods follows.

a. Rod CM-401, (D-55) (Figure 2)

- Location and extent of typical crud spalled areas downstream from corner angle dimples.
- Cladding deterioration at 32 inches (bulge) and at 45-1/4 inches (severe corrosion).

b. Rod CM-412, (D-54) (Figure 3)

- Crud spalling characteristics identical to rod CM-401.
- Cladding penetration at 45-1/4 inches (severe corrosion).
- Cladding penetration and bulging in the lower region of the rod. The characteristics of this penetration are different from those at 45-1/4 inches and indicate localized cladding embrittlement. These features are considered to be representative of a secondary defect.

c. Rod CM-409, (D-54) (Figure 4)

- Incipient failure region at 42 inches.
- Local crud build-up downstream from dimples at 46 inches and 49-5/8 inches.

6.3 NEUTRON RADIOGRAPHY

Full length neutron radiographs of all eight rods are shown in Figures 5, 6, 7, 8, 9, 10, 11, and 12. There are two prints included for each rod. Both were made from the same negative, but using different printing exposures for the purpose of emphasizing the UO_2 fuel in one case and the cladding in the other. There are two scales alongside

TABLE 1

SUMMARY OF POST-IRRADIATION EXAMINATION

<u>Type Examination</u>	<u>Rod Serial Numbers</u>							
	<u>0.570 in. o.d.</u>			<u>0.700 in. o.d.</u>				
	<u>CP-307</u>	<u>CP-409</u>	<u>CP-511</u>	<u>CM-224</u>	<u>CM-505</u>	<u>CM-401</u>	<u>CM-409</u>	<u>CM-412</u>
	<u>Pt-I</u>	<u>Pr-I</u>	<u>Pr-I</u>	<u>Pt-I</u>	<u>Pr-I</u>	<u>Pr-F</u>	<u>Pr-F</u>	<u>Pr-F</u>
1. Visual	x	x	x	x	x	x	x	x
2. Measurements								
a. Profilometry	x*	-	x	x*	x	x	x	x
b. Length	x	-	x	x	x	-	x	-
3. Neutron Radiography	x	x	x	x	x	x	x	x
4. Gamma Scan	x	x	x	x	x	x	x	x
5. Burn-up				x	x			
6. Metallography	x	x	x	x	x	x	x	x

Pt = pellet UO₂Pr = powder UO₂

I = intact

F = failed

* = profiled also after decrudding

the neutrographs: one shows the distance in inches from the top end of the rod. This scale is convenient for matching features on the photographs obtained during the visual examinations (see Figures 2, 3, and 4). The other scale shows 4-inch interval marks (or nodes) which are convenient for matching with the respective gamma scans. These marks are numbered consecutively. The locations where cross sections were prepared metallographically are pointed out by reference to specific figures that show the results of the microscopic examinations.

The neutrographs emphasizing the UO_2 show the distribution and size of the central voids. The central voids are the result of either fuel densification (or migration of included voids to the center under a thermal gradient during operation) or fuel relocation or slumping associated with the fall in the level of the molten UO_2 column due to lowering of the rod power or to complete solidification of the UO_2 .

It should be pointed out that three rods, CP-409, CM-401, and CM-412, were sheared into halves at the site for cask shipment. The sheared ends were brought together for the neutron radiography using short plastic tubes. Some UO_2 loss due to the shearing action and to subsequent rod handling is the reason for the discontinuity in the fuel column at the midpoints of these three rods.

6.3.1 Intermediate Performance Fuel

Fuel rods CP-307 (pellet) and CP-511 (powder) represent this performance category. Incipient melting was predicted to have occurred in these rods during the operation at the maximum power achieved. The neutrographs, Figures 5 and 6, show a continuous central void over the middle region in both rods. Evidence of the small amount of fuel relocation is clearly visible in rod CP-307, where the pellet dishes are partially filled. The location of the termini of the voids

correspond to the location at which the heat fluxes cause the central temperature to be about 1800°C , assumed to be the densification temperature. The cladding of these rods was free of any evidence of hydriding.

Fuel rod CP-409 had the surface protrusion at the midregion. The neutrograph of this rod (Figure 7) showed no evidence of cladding damage nor any unusual UO_2 behavior at the location of this feature. From this, it was concluded that the rod was in satisfactory condition, and that the surface protrusion was actually an extraneous deposit trapped beneath a spacer spring, as had been deduced earlier. No further examination was performed on this fuel rod.

6.3.2 Advanced Performance Fuel

6.3.2.1 Intact Rods

Fuel rods CM-224 (pellet) and CM-505 (powder) represent the Advanced Performance fuel category. Moderate central melting was predicted to have occurred in these rods.

The neutrographs (Figures 8 and 9) show that fuel redistribution occurred in both fuel rods. Interruptions in the central void are visible where early freezing at certain elevations caused UO_2 bridges to form. The fact that the fuel rods with moderate central melting had undergone power cycles from the molten-to-solid-to-molten conditions without any visible or measurable effects on the fuel-rod diameter indicate that such bridges caused no problems. The cladding of these rods was free of any evidence of hydriding.

6.3.2.2 Failed Rods

Rods CM-401 (Figure 10) and CM-412 (Figure 11) have essentially identical UO_2 and cladding features. In general, the overall internal void distribution in these two fuel rods are similar to that in the intact powder rods of the same size; viz., rod CM-505 (Figure 9).

Significant features to be noted are:

- No signs of large UO_2 losses through the cladding defects are observed. The region of the cladding crack in rod CM-401 is at position 35 inches from the top end of the rod. The region of the corrosion penetration in rod CM-412 is at position 45-1/4 inches. Subsequent cross sectioning through this location confirmed that the losses were small and confined to the region in the immediate vicinity of the opening in the cladding.
- As suspected, the lower regions of both rods are heavily hydrided from the inside cladding surfaces. The secondary defect in rod CM-412, located 6 inches from the bottom end is associated with the resulting cladding embrittlement. The hydriding occurred after the primary defect further up the rod, which permitted water to enter the fuel rod. The hydrogen generated from the corrosion reaction between the cladding inside surfaces and the water was absorbed by the cladding, thus precipitating the hydrides.
- Hydriding of the cladding from the outside surfaces are visible over the upper region of the fuel rods. The patterns correspond to the crud spalled and corroded areas observed during the visual inspections.

6.3.2.3 Incipient Failure Rod

The neutron radiograph for rod CM-409 (Figure 12) shows the UO_2 and the cladding to be in satisfactory condition, except for the region of the incipient failure. The void distribution in the UO_2 column is similar to that observed in the hottest powder rod of the same size. A small hydride spot is visible where the "incipient failure" or accelerated

corrosion of the cladding occurred on this rod. Otherwise, the rod is in entirely satisfactory condition.

6.4 GAMMA SCANS

All eight fuel rods were scanned for their gross axial gamma activity. The traces obtained are shown in Figures 13 through 20. Each figure has a scale along the bottom margin calibrated with nodes 0 to 18 representing 4-inch increments along the rod length. These nodes correspond to those shown in the neutron-radiographs. Above the gamma scan is a schematic representation of the fuel rod and the central void distribution taken from the neutrographs. It should be noted that depleted UO_2 pellets were placed at the ends of the active fuel column as insulators. Also, the large increase in gamma activity in the plenum of the fuel rod is due to tantalum metal used as a neutron flux suppressor. The need for this absorber arose because the plenum length in the center-melt rods had to be longer by design and the poison was needed to keep the upper ends of the rods in the surrounding bundles from power peaking.

Evidence of a pattern for the UO_2 redistribution is obvious from the scans obtained from rods CM-505 (Figure 17) and CM-409 (Figure 18). The gamma scans of rods in which extensive central melting occurred can in general be divided into the four regions as follows:

- Region 1, starting at the top of the rod, is a region of fuel densification. In this region, the gamma activity increases proportionally to the wire scan pattern. Also, a small uniform void is formed due to the UO_2 densification.
- Region 2 is a region of fuel melting and from which molten UO_2 has redistributed. The gamma activity is fairly constant corresponding to a large central void.

- Region 3 is a region where UO_2 has moved from regions above. A sharp increase in gamma activity corresponds to this density increase. Little or no voids are present.
- Region 4 is similar to the first region, but the small central void resulting from the densification may be filled with UO_2 that seeped down from a pool of molten UO_2 . As mentioned earlier, bridges may form, so that the regions are not as distinct as described above.

The Intermediate Performance Fuel, in which only incipient melting occurred, showed a gamma activity distribution that followed the core wire scan pattern closely. The overall activity scans for these rods are fairly continuous and smooth. It is interesting to note that the dishes at the ends of the pellets in rod CP-307 (Figure 13) are very distinctly recorded in the scan for this rod. Further, the periodic dips in the scan become less distinct over the high power region of the rod where some relocation of UO_2 into the dish void had occurred (also see respective neutrograph, Figure 5, for this evidence).

It is important to note that these gamma activity measurements represent the non-operating state of the rods. This distinction is very important when considering the scan for the rods that sustained moderate amounts of central melting during operation. For these rods, the gamma scans do not represent the axial power distribution, because the UO_2 during operation would be redistributed into the voids in the upper region by the volume expansion on melting.

The scans are very valuable, however, for determining relative gross power outputs among the rods. Relative powers between specific rods can be determined using the areas beneath the gamma scan traces. These

ratios can then be compared with those obtained by using calculated rod local power factors.

The areas beneath each of the gamma scans, as measured with a planimeter, are shown in Table 2.

Comparisons of the relative power ratios among specific rods of interest are shown in Table 3.

It can be seen that the ratios from measured data compare very favorably with physics design predictions. It is especially important to note that the powers of the Advanced Performance (0.700-inch o.d. rods) powder and pellet fuel were almost the same as predicted, with the powder fuel being slightly higher; i.e., 3 percent by physics predictions, and 1.2 percent by gamma scan data.

6.5 PROFILOMETER MEASUREMENTS

The diameters of seven fuel rods were measured. Three traces were obtained on each rod: 0-degree and 90-degree orientations and a helical trace. Two of these rods were measured additionally after decrudding (0.700- and 0.570-inch o.d. pellet rods). Pertinent results were as follows:

- There were no indications of rod diameter increase caused by interaction between fuel and cladding.
- No changes in ovality after irradiation were measured.
- There was no circumferential ridging at pellet interfaces.

These results together with those obtained at the site confirmed that the volume expansion on melting of the UO_2 had been satisfactorily accommodated in the design,

TABLE 2

AREAS BENEATH GAMMA SCANS

<u>Rod</u>	<u>Calculated Local Factor</u>	<u>Area under Gamma Scan Trace (in. 2)</u>	<u>Assembly</u>	<u>Type</u>
CM-505	1.89	144.4	D-55	Powder
CM-401	1.81	133.1	D-55	Powder
CM-412	1.81	141.8	D-54	Powder
CM-409	1.78	144.3	D-54	Powder
CM-224	1.89	142.7	D-52	Pellet
CP-409	1.43	104.5	D-51	Powder
CP-511	1.85	127.0	D-51	Powder
CP-307	1.83	128.5	D-50	Pellet

TABLE 3

COMPARISONS OF RELATIVE POWER RATIOS

<u>Rods Compared</u>	<u>Physics Ratio of Local Factors</u>	<u>Gamma Scan Ratio of Areas</u>	<u>Comparison</u>
CM-505 CM-224	1.00 (1.03)*	1.012	Highest Power Powder to Highest Power Pellet (7 × 7)
CM-409 CM-412	0.984	1.025	Locals in Assembly D-54
CM-505 CM-401	1.044	1.085	Locals in Assembly D-55
CP-511 CP-307	1.011	0.990	Pellet to Powder (8 × 8)

* Core Physics calculations showed powder fuel to be operating at 3% higher power than the pellet fuel.

and furthermore, no adverse effects were experienced even after several power cycles involving melt-solid-melt transitions.

6.6 LENGTH MEASUREMENTS

Overall length measurements were performed on five of the rods returned intact. Increases were measured on all of these rods. The results are shown in Table 4.

6.7 METALLOGRAPHIC EXAMINATIONS OF CROSS SECTIONS

Cross sections were taken from specific locations of the rods for detailed microscopic examination. The sections taken and the scope of the examinations were divided into two groups according to the specific purpose.

The purpose for Group I was to verify the degree of melting, by examining the UO_2 boundary corresponding to the last day's operation. The rods examined in this group included:

CM-224 (pellet UO_2)	} Advanced Per- formance Fuel 0.700-inch o.d.
CM-505 (powder UO_2)	

CP-307 (pellet UO_2)	} Intermediate Per- formance Fuel 0.570-inch o.d.
CP-511 (powder UO_2)	

The purpose for Group II was to examine the UO_2 and cladding for evidences pointing to the cause of failure. The rods examined in this group included:

CM-401 Definite failure	} All powder UO_2 Advanced Per- formance Fuel 0.700-inch o.d.
CM-412 Definite failure	
CM-409 Incipient failure	

6.7.1 Group I Examinations

The cross sections of the rods in Group I are shown in Figures 21 through 25. The sections were made in the high power region of the rods (41-1/4 inches from the top), indicated

by the wire scan data. An additional cross section was made for rod CM-224, at location 45 inches, when the first section showed a structure very close to melting, but no definite molten boundary. The molten boundary is defined as the sharp demarcation between a high density region and a region containing porosity. It is the terminus of the porous region. This boundary is clearly visible in the photo-mosaics of the UO_2 in Figures 21, 22, and 23b. The UO_2 structure showed no molten boundary for the Intermediate Performance powder and pellet rods, as shown in Figures 24 and 25.

A comparison of the melt radii determined from these examinations with those calculated from design curves is shown in Table 5. Greater degrees of melting were calculated for all rods at the highest power achieved during the cycle, but no experimental verification was possible.

As mentioned earlier, the final 7-day operation at a reduced power level erased all evidence of the prior molten boundary. The UO_2 microstructures observed reflect only the last day's operation.

The design confirmation of the melt radii in the large diameter fuel rods by the experimental data is exceptionally good. A further discussion on this matter is presented in a later section.

6.7.2 Group II Rods

The two defected fuel rods CM-401 and CM-412 were sectioned at 34-7/8 inches and 45-1/4 inches, respectively, from the top end. The cross-sectional planes examined cut through each of the defects.

The section of rod CM-401 is shown in Figure 26. The reason for the cladding bulging locally is clearly evident in the photo-mosaic of the defect region at high magnification. Localized corrosion of the

TABLE 4

RESULTS OF LENGTH MEASUREMENTS

Rod Serial No.	0.570-in. o.d.		0.700-in. o.d.		
	<u>CP-307</u>	<u>CP-511</u>	<u>CM-224</u>	<u>CM-409</u>	<u>CM-505</u>
	Pellet	Powder	Pellet	Powder	Powder
Post-Irradiation Length (a) *	74.807	74.735	74.841	74.752	74.753
	74.809	74.746	74.843	-	74.758
Pre-Irradiation Length (b)	74.763	74.722	74.781	74.730	74.731
(c)	74.768	74.727	74.786	74.735	74.736
(d)	74.774	74.733	74.792	74.741	74.742
ΔL (a) - (d)	+0.033/0.035	+0.002/0.013	+0.049/0.051	+0.011	+0.011/0.016

Notes: (a) Fuel rod cladding temperature: 110 to 120°F

(b) Rod temperature: 70°F

(c) Rod temperature adjusted to 90°F by calculation

(d) Rod temperature adjusted to 115°F by calculation

* A dummy fuel rod was used as a standard for calibrating the apparatus

TABLE 5

COMPARISON OF MELT RADII

	Intermediate Performance		Advanced Performance	
	0.570-in. o.d.		0.700-in. o.d.	
	CP-511 Powder	CP-307 Pellet	CM-505 Powder	CM-224 Pellet
Q/A, Btu/h-ft ² (last day)	437,000	432,000	472,000	460,000
Melt radius from sample r/R:	None	None	0.52	0.12
Calculated melt radius r/R:	0.08	None	0.525	0.29
Q/A, Btu/h-ft ² (highest power achieved)	507,000	500,000	525,000	525,000
Calculated melt radius r/R:	0.41	0.23	0.61	0.52

cladding inside surface resulted in the zirconium oxide which, due to its volume expansion, forced the cladding outward. The cladding is penetrated by cracks. However, it cannot be determined whether the crack occurred first and permitted the entry of water, or whether the attack occurred first and caused the bulging and attendant cracking.

The UO₂ appears to be satisfactory. Only a small area adjacent to the failure region is visibly oxidized. No loss of UO₂ through the cladding cracks is evident. The structure of the UO₂ across a diameter through the defect region is shown in Figure 28a. The molten boundary is visible. The melt radius from this section was also included in the thermal analysis, since the UO₂ appeared unaltered. The resulting correlation with design prediction was quite good.

The section of rod CM-412 is shown in Figure 27. The damage is extensive with approximately one-sixth of the cladding circumference lost. The UO₂ has remained remarkably intact, with only a very small loss (estimated to be 0.5 to 1.0 gram) from the region immediately adjacent to the hole in the cladding.

The cladding damage is the result of excessive corrosion. This can be seen by the massive zirconium oxide present, and by the structure of the cladding that borders the hole showing high oxygen and hydrogen contents. Evidence of high cladding temperatures causing the accelerated corrosion is seen in the grain growth in the cladding on the right hand side of the picture. This characteristic is more clearly evident in the incipient failure discussed later.

The UO₂ is oxidized throughout the entire cross section. No molten boundary is evident. However, the structure suggests that the melting in this region had reached close to the fuel surface of the UO₂ bordering the cladding defect during the highest power operation, without serious effects. It is reasonable to expect that the degree of melting becomes somewhat higher than that predicted for the given operating power level. In a defected condition, the fuel-cladding gap resistance increases so that the fuel will operate hotter. In addition, the higher oxygen content of the fuel will lower its thermal conductivity and also its melting point so that a greater amount of melting can be expected with the combination of

these factors. The structure of the fuel across a diameter is shown in Figure 27b.

The section through the incipient failure region of rod CM-409 is shown in Figure 29. It was taken at a position 45-1/4 inches from the top of the rod. The important characteristics are recorded in the photo-mosaics of the cladding in the region of the incipient failure. The as-polished and etched surfaces are shown. Significant observations were as follows:

- The cladding was not collapsed as was presumed earlier. The depressions were actually the result of metal corroded away. The original curvature of the inside circumference of the cladding was retained.
- The evidence of localized cladding overheating is very clearly visible by the grain growth and the limited area exhibiting this cladding behavior. The grains have grown from an original of ASTM 11-12 to ASTM 9-10 in the corroded region.

Based upon the results from independent grain growth studies, the temperature reached to promote the grain growth observed was estimated to be very close to 1400°F. The absence of beta quenched or acicular structure is proof that the temperature did not exceed 1500°F (transformation temperature for Zircaloy-2). A further support of the 1400°F estimate is given by calculations to determine the amount of metal that would

react in a Zircaloy-water reaction at 1400°F for a time period of 16 days (384 hours). This time period corresponds to the period during which the centermelt bundles were operating at the maximum power, and is considered to be a reasonable upper limit. The calculated equivalent metal thickness of 10 mils reacted under such conditions corresponds exactly with the observed 10 mils wall thinning seen in this cross section.

Because the rod was intact, the condition of the UO₂ was not affected, as was the case for the two failed rods above. The molten boundary was distinct, as shown in the photo insert, Figure 29b. The melt radius from this rod was also used to show the good agreement between experimental data and design prediction.

6.8 BURN-UP ANALYSES

Burn-up analyses were performed on the large diameter (0.700-inch o.d.) powder and pellet UO₂ fuel rods in which moderate amounts of central melting had occurred. Both rods were sampled at the same elevation (middle of rod or peak power region). UO₂ samples from the fuel periphery (a) and from interior positions (b) and (c) were analyzed. The results shown below are based upon Neodymium.

The results from the UO₂ on the periphery compare within 9 percent of the estimated burn-up at the peak position. The lower burn-up of the central region is attributed to dilution effects during the period that the fuel was molten.

Location	Burn-up (MWd/T)				
	r/R			Estimated	
	a	b	c*	Rod Average	Rod Peak
Rod	0.97	0.42	Core, 0		
CM-505	3810	—	3400	2920	4150
CM-224	3580	3080	2830	2760	3920

* Core sample represents molten fuel

VIEW IN POOL

VIEW IN HOT CELL

NOTE: MOST OF THE DEPOSIT HAS BEEN REMOVED DUE TO ROD
HANDLING IN SHIPMENT AND HOT CELL EXAMINATION

FIGURE 1. SURFACE DEPOSIT ON ROD CP-409, FROM ASSEMBLY D-51

FIGURE 2. OVERALL VIEW OF FAILED ROD CM-401

FIGURE 2. OVERALL VIEW OF FAILED ROD CM-401 (CONTINUED)

SEE FIGURE 27

FIGURE 3. OVERALL VIEW OF DEFECTED
ROD CM-412

SEE FIGURE 29
↑

HEAD-ON VIEW

DIMPLES

PROFILE VIEW

FIGURE 4. INCIPIENT FAILURE ROD
CM-409

FIGURE 5. NEUTRON RADIOGRAPH OF ROD CP-307, 0.570-inch O.D. UO_2 PELLET ROD

FIGURE 6. NEUTRON RADIOGRAPH OF ROD CP-511, 0.570-inch O.D. UO₂ POWDER

FIGURE 7. NEUTRON RADIOGRAPH OF ROD CP-409

FIGURE 8. NEUTRON RADIOGRAPH OF ROD CM-224, 0.700-INCH O.D. UO_2 PELLET ROD

FIGURE 9. NEUTRON RADIOGRAPH OF
ROD CM-505, 0.700-inch o.d.
POWDER ROD

FIGURE 10. NEUTRON RADIOGRAPH OF DEFECTED ROD CM-401

FIGURE 11. NEUTRON RADIOGRAPH OF DEFECTED ROD CM-412

FIGURE 12. NEUTRON RADIOGRAPH OF
INCIPIENT FAILURE ROD
CM-409

FIGURE 13. GAMMA SCAN OF ROD CP-307

FIGURE 14. GAMMA SCAN OF ROD CP-511

FIGURE 15. GAMMA SCAN OF ROD CP-409

FIGURE 16. GAMMA SCAN OF ROD CM-224

FIGURE 17. GAMMA SCAN OF ROD CM-505

FIGURE 18. GAMMA SCAN OF ROD CM-409

FIGURE 19. GAMMA SCAN OF ROD CM-401

FIGURE 20. GAMMA SCAN OF ROD CM-412

b.

50X

10X

a.

FIGURE 21. CROSS SECTION THROUGH ROD
CM-224, D-52

b.

50X

a.

10X

FIGURE 22. CROSS SECTION THROUGH ROD
CM-224, D-52

b.

50X

FIGURE 23. CROSS SECTION THROUGH ROD
CM-505, D-55

**FIGURE 24. CROSS SECTION THROUGH
ROD CP-307, D-50**

**FIGURE 25. CROSS SECTION THROUGH
ROD CP-511, D-51**

a. **FIGURE 26. CROSS SECTION THROUGH DEFECT, ROD CM-401, D-55**

FIGURE 27. CROSS SECTION THROUGH DEFECT,
ROD CM-412, D-54

a. ROD-CM-401

50X

SEE FIGURE 27b.

b. ROD CM-412

50X

FIGURE 28. MICROSTRUCTURE OF UO_2 IN
DEFECTED RODS CM-401 AND CM-412

FIGURE 29. CROSS SECTION THROUGH INCIPIENT FAILURE, ROD CM-409, D-54

SECTION VII

THERMAL HYDRAULIC LOOP TEST AND ANALYSES

Irradiation of high power density, centermelt, fuel in the Big Rock Point reactor resulted in the failure of two fuel rods that were adjacent to the corner angles. These rods failed under conditions remote from the expected CHF (critical heat flux) level.

The failures, as well as other areas showing signs of overheating (spalling) occurred downstream from dimples in the corner angle. (For details refer to the preceding section on post-irradiation examination.) Since calculations done prior to and during the cycle indicated substantial margin with respect to CHF, it was postulated that the failures were caused by a local overheating phenomenon. The local overheating appeared to be initiated by an axial flow disturbance propagating from the dimples. To test these hypotheses, thermal-hydraulic tests were conducted.

7.1 DESCRIPTION OF THE TEST

The test-rig geometry was an exact duplication of the corner geometry, including the corner angle with dimples of the 7 x 7 Advanced Performance fuel assembly. A picture of the test assembly, a 3 x 3, 9-rod mock-up, is shown in Figure 30. It should be noted that the earlier testing ^(1,2) conducted at the beginning of the program as part of the bundle design effort did not include the corner angle.

The heater rods, 0.700-inch o.d. made of high heat flux heaters sheathed with nickel tubing, were uniformly heated axially. The two heater rods (rods B and F, Figure 31) that were adjacent to the corner angle dimples (or in positions equivalent to those where the actual failures occurred) had four thermocouples located as shown in Figure 32a. The remaining seven colder rods each had one thermocouple, 7/8 inch from the end of the heated

length. Sheathed thermocouples, 0.02-inch o.d., were buried in grooves 0.005 inch below the outside heater surface, and the residual space filled with silver solder. From prior experience and calculations, this combination of materials and dimensions is known to minimize heat flux distortion at the thermocouples. For the last three test runs, the thermocouples were brought closer to the dimples as shown in Figure 32b.

To be able to compare the tests with the failed rods, it was necessary to make the most limiting local (sub-channel) condition of the test section equivalent to the local condition that existed at the point of failure in the centermelt fuel. The local conditions at the points of failure were determined by first using a standard one-dimensional, thermal-hydraulic, parallel bundle calculation on the whole core to determine average bundle conditions. Next, a multichannel hydraulic analysis was performed on the centermelt bundle to determine the local conditions at the failure points. (See Table 6.) The CHF was calculated by using correlations that were appropriate for the geometry. An annular correlation was used for rods adjacent to the corner angle and a multi-rod correlation was used for interior rods.

Several multi-channel hydraulic analyses were then performed on the test configuration to determine the inlet conditions that would result in the local conditions listed in Table 6.

The heater peaking factors were adjusted in order to obtain the required local conditions at the dimples of interest, and additionally to prevent rods that became corner rods in the test configuration from exhibiting CHF before the specific rods of interest. A comparison of the actual rod peaking pattern of the nine

FIGURE 30. A 3 x 3, 9-ROD MOCK-UP OF A 7 x 7 BUNDLE CORNER

FIGURE 31. ACTUAL AND TEST PEAKING FACTORS

TABLE 6

LOCAL CONDITIONS AT FAILURE POINTS

<u>Distance from Fuel Bottom (inches)</u>	<u>Quality</u>	<u>Enthalpy Btu/lb</u>	<u>Mass Flow lb/ft²-h</u>	<u>Heat Flux Btu/h-ft²</u>	<u>CHFR</u>
35	0.132	669.5	4.72×10^5	4.75×10^6	1.44
44	0.148	678.7	4.59×10^5	4.53×10^6	1.44

a. LOCATION OF THERMOCOUPLES FOR FIRST TWELVE TEST RUNS

0.1 in. CENTERS TANGENTIALLY.
1/2 in. OR 1/4 in. CENTERS
LONGITUDINALLY WITH AVERAGE
3/4 in. DOWNSTREAM FROM DIMPLE.

b. RODS AFTER ADJUSTMENT TO BRING THERMOCOUPLES CLOSER TO DIMPLES

FIGURE 32. LOCATION OF THERMOCOUPLES IN THE TWO SPECIFIC RODS OF INTEREST ADJACENT TO THE CORNER ANGLE

rods in the corner of the 49-rod assembly in the reactor, normalized on the nine rods, and the peaking pattern used in the loop tests is shown in Figure 31.

All tests were run at 1350 psia. The "base" conditions (Table 6) were first tested. The strip chart outputs of the thermocouples were monitored, but no evidence of overheating was detected. CHF was then induced by various combinations of reducing flow, reducing subcooling, and increasing heater power. A total of 11 valid CHF points were obtained (a total of 15 test runs was made).

The test rig was removed from the loop section and inspected after the 12th run. It was at this time that the small discolored spots near the corner angle dimples were observed. The corner angle-spacer assembly was moved and rod B rotated to bring one of the thermocouples within the discolored spot at the top dimple (see Figure 32b). In addition, a "ski-jump" crud formation was simulated on rod F immediately downstream from the top dimple. This was purely a geometric simulation using a nickel-weld bead shaped with a small grinding wheel.

7.2 TEST RESULTS AND ANALYSES

The test results indicated that the local overheating in the reactor must be quite different from the condition normally referred to as CHF.* No evidence was found of overheating of heaters in the precise regions where fuel rods had failed, either at reactor operating conditions or at any condition below normal CHF. Normal CHF was induced at heat fluxes and qualities substantially higher than reactor conditions. These test results, interpreted by our multi-channel hydraulic

analysis and extrapolated to reactor condition, indicate that the MCHFR, for reactor conditions, was about 1.9 (see Figure 33).

The tests did indicate overheating (discoloration) of small areas ($1/16$ inch²) immediately downstream from the dimples. While the in-pile failure regions occurred further downstream, it nevertheless appears that these spots are related to reactor fuel failures. However, their small size, and their distribution along the entire heated length, including the subcooled region, are not at all characteristic of normal CHF.

A thermocouple located at one of these spots showed a temperature fluctuation beginning at about 170,000 Btu/h-ft² and 1 percent average exit quality conditions well below the most severe reactor conditions. Fluctuation amplitude increased with heat flux, up to about 450,000 Btu/h-ft², and subsided somewhat beyond that. Preliminary evaluation of the thermocouple transient response indicates a temperature excursion at the heater surface, directly above the thermocouple, of about 75°F.

Extrapolation is necessary to connect the loop test results with the actual reactor experience involving the two fuel rod failures, especially when a corrosion phenomenon is involved. The difference in the positions of the overheated spots in the heat-transfer test and of the cladding deterioration areas observed downstream from the corner angle dimples can be reconciled by important differences between the loop test and the reactor operations. In addition to the clean environment (crud free) of the loop test, the test runs were very short, and nickel sheathing was used for the heater rods because it is more resistant to corrosion at the higher temperatures. These conditions, therefore, precluded any possibility of the formation of the unique "ski-jump" crud deposit in the wake of the dimples observed on the failed powder rods. These "ski-jumps" were located exactly at

* Normal CHF is defined as the heat flux at which the heat transfer mechanism breaks down with temperature oscillations beyond normal operating levels. This specifically excludes time dependent processes.

FIGURE 33. CHF POINTS ON CENTERMELT TESTS

the position where the spot overheating was observed in the loop test and could have aggravated the already perturbed coolant flow and/or heat flux to displace the overheating phenomenon downstream from the corner angle dimple. The reason such an aggravation was not observed in the loop test with the simulated nickel "ski-jump" is attributed to the fact that it was not a heat insulator.

It can be concluded that the center-melt rod failures occurred only in high power rods operating for long times adjacent to the corner angle. The actual failures are due to highly localized overheating near the dimples causing accelerated corrosion of the cladding. Therefore, the proposed "fix" is to significantly reduce the power in the fuel rods adjacent to the corner angle (from a relative power of 1.81 to 0.22) by rearranging fuel rods in the bundle (see Section IX).

Analyses of the centermelt bundle modified in this manner have been performed

for the limiting (by license) conditions. The new maximum local peaking factor, with appropriate correction for flux tilt is 2.07, compared to 1.89 during the previous irradiation. The peak rod is in an interior position away from the corner angle. We are confident in the ability of the multi-channel mixing code to conservatively predict the local (sub-channel) flow and quality for the new configuration. This ability was confirmed by the analyses made on the critical heat flux tests for the original configuration. Bundle power and heat flux conditions were selected to yield a MCHFR of 1.50 at 122 percent overpower from 235 MWt using the standard whole core analysis. This condition was then analyzed using the multi-channel mixing code. The resulting MCHFR is 1.71, and confirms that the whole core analysis MCHFR is conservative relative to the multi-channel analysis and that the center-melt bundles may be operated within reactor license limits. Table 7 summarizes these analyses.

TABLE 7

CRITICAL HEAT FLUX ANALYSIS OF
MODIFIED CENTERMELT BUNDLES

<u>Whole Core Calculations</u>		<u>Local Conditions</u>	
Peak heat flux	= 587,000 Btu/h-ft ²	Peak heat flux	= 578,000 Btu/h-ft ²
Mass flow rate	= 8.84×10^5 lb/h-ft ²	(at MCHFR)	
MCHFR	= 1.50	Mass flow rate	= 8.945×10^5 lb/h-ft ²
Average quality	= 0.016	MCHFR	= 1.708
(at MCHFR)		Quality	= 0.056
Distance from	= 29 inches	(at MCHFR)	
fuel bottom		Distance from	= 35 inches
		fuel bottom	

SECTION VIII

DISCUSSION

8.1 DESIGN CONFIRMATION OF THE DEGREE OF MELTING

The degrees of melting that were determined by the examinations of the fuel-rod cross section corroborate the design predictions very satisfactorily, as was shown in Table 5. Reference should be made to the design curves showing the predicted melt fraction versus heat flux, in order to appreciate the fact that due to the steep slope of the curves, small changes in heat flux cause significant differences in the melt fractions. (See Appendix for design curves.) Consequently, a comparison of the observed melt radius with that predicted for a given heat flux may sometimes show differences that may appear to be relatively large. Another way to show the good correlation between the experimental data and the calculated predictions is to compare heat fluxes, as illustrated in Figure 34. Heat fluxes from three sources are compared:

- a. Heat fluxes calculated from reactor operating data for the last day's operation, and using design rod local power factors.
- b. Heat fluxes derived from the design curves and using the melt radius determined by hot cell examinations of the rods.
- c. Relative heat fluxes calculated using the rod relative power factors from the gamma scan data and an arbitrary base of 470,000 Btu/h-ft² (peak heat flux of rod CM-505 derived from above).

The data from the two failed rods and from the one incipient failure rod are included

in the comparison. As discussed previously, the structure of the fuel in the cross section from rod CM-412 had been altered as a result of the defect. This is the reason for the uncertainty band shown on Figure 34.

It can be seen that the values generally agree within 5 percent.

8.2 ANALYSIS OF THE UO₂ BEHAVIOR

Thermal calculations were performed to reconstruct the behavior of the UO₂ during the actual rod operation that resulted in the void distributions in the UO₂ fuel column, observed in the neutron radiographs. The analysis was based upon rods CM-224 and CM-505, which were the highest power pellet and powder UO₂ rods, respectively, in the Advanced Performance (7×7) Fuel assemblies.

Calculations were performed for the two reactor power levels of interest, 200 MWe and 185 MWe, which correspond to the maximum power achieved and the last day's run, respectively. The equivalent peak heat fluxes for the rods were 525,000 and 472,000 Btu/h-ft².

In this analysis, two main assumptions were made:

- A temperature of 1800°C (3275°F) was assumed for the onset of fuel densification. Further, the UO₂ density was assumed to reach 98 percent of theoretical, regardless of the original manufactured density; i.e., nominally 85 percent T.D. for the powder fuel and 94 percent T.D. for the sintered pellet fuel.
- All of the UO₂ molten at full power was assumed to relocate upon cooling.

FIGURE 34. PREDICTED HEAT FLUX COMPARED WITH EXPERIMENTALLY DETERMINED ABSOLUTE AND RELATIVE HEAT FLUX FOR FIVE CENTERMELT RODS

to the lower part of the once molten central core region. The solidified UO_2 was assumed to be 100% dense. Thus, for this discussion, central void is defined as space formed by the UO_2 at temperatures between 3275°F and 5080°F densifying and moving away from the center (or originally distributed porosity coalescing and moving into the central region) and by molten UO_2 relocating to a lower region.

The radial temperature profiles at the two peak heat fluxes are shown in Figures 35 and 36 for the pellet and powder rods. Moderate amounts of melting were calculated to occur.

The radii at which densification and melting begin as a function of heat flux for pellet and powder fuels are shown in Figures 37 and 38. Using these curves, the volume

of fuel affected by densification and by melting for any given heat flux is easily determined.

The heat flux to cause central melting as a function of the densification void is shown in Figure 39. The assumption here is that the void is fully developed before the melting, if any, occurs. The use of this curve is to determine whether melting occurs. From previous curves, the void volume can be determined for a given heat flux, knowing the volume of fuel that is above the densification temperature. If the specific heat flux is equal to or above the heat flux determined from Figure 39, melting occurs. If this is the case, then the melt radius is obtained by referring back to Figures 37 and 38, and using the $\alpha 5080$ curves.

Using the wire scan data for axial power distribution, the assumptions stated at the beginning, and the method of analysis

FIGURE 35. TEMPERATURE DISTRIBUTION FOR A 7 x 7 PELLET ROD FOR SEVERAL HEAT FLUXES (EOC-1)

FIGURE 36. TEMPERATURE DISTRIBUTION FOR A 7 x 7 POWDER ROD FOR SEVERAL HEAT FLUXES (EOC-1)

FIGURE 37. MOLTEN RADIUS AND ALPHA (3275) VERSUS HEAT FLUX FOR 7 x 7 PELLET ROD (CM-224)

FIGURE 38. MOLTEN RADIUS AND ALPHA (3275) VERSUS HEAT FLUX FOR 7 x 7 POWDER ROD (CM-505)

FIGURE 39. HEAT FLUX TO CAUSE INCIPIENT CENTER MELTING VERSUS VOID RADIUS

described above, predictions can be made as to the size and position of the void and relocated fuel. The predictions made for the two fuel rods considered are graphically represented in Figures 40 to 43.

In actual operation, all of the molten fuel does not relocate to the lower part of the once-molten region. Some of it will solidify on the solid-molten boundary, so that the actual void seen in the neutron radiograph is smaller than the prediction. Also, differences in heat transfer locally can cause premature solidification of molten UO_2 to form plugs that trap voids below in regions where the predictions indicate that completely solid fuel should exist on cooling.

Figures 42 and 43, which reflect the prediction of the UO_2 condition on cooling from the last day's run, compare quite well with that observed in the neutron radiographs. The relocation of the fuel as predicted also compares well with the gamma scans. For ease in making these comparisons, the scales near the upper margin correspond to those on the gamma scan and the neutron radiograph.

The conclusion drawn from this analysis is that the void distribution and fuel redistribution can be predicted fairly well using a relatively simple model and the calculational methods used for the design.

FIGURE 40. EXPECTED VOID FOR ROD CM-224 AT 200 MWt

FIGURE 41. EXPECTED VOID FOR ROD CM-505 AT 200 MW;

FIGURE 42. EXPECTED VOID FOR ROD CM-224 AT 184 MWt

FIGURE 43. EXPECTED VOID FOR ROD CM-505 AT 184 MWt

SECTION IX

CORRECTIVE ACTION

The failure analysis has shown that the basic cause of the two failures in the center-melt assemblies was the very local thermal-hydraulic perturbation caused by the corner angle dimples in conjunction with and high rod surface heat fluxes. In order to eliminate this problem so that the assemblies can be returned to the core for continued irradiation, it is planned to replace the power producing rods now adjacent to the corner angles with very low power producing rods containing depleted UO_2 . To accomplish this change, rods already in the bundle will be rearranged, and

eight new rods of low enrichment will be inserted into each assembly. Figure 44 illustrates the basic moves that will be made. A quarter of a 7×7 bundle is shown. A similar rearrangement of rods is planned for the 8×8 bundles.

The thermal-hydraulic characteristics of the revised fuel assemblies have been thoroughly analyzed to show that they comply with license thermal-limit requirements, as discussed in a previous section.

FIGURE 44. REARRANGEMENT OF RODS IN BUNDLE

APPENDIX A

SUMMARY OF THE CENTERMELT
FUEL DESIGN

A.1 STATEMENT OF WORK

The Power Reactor High Performance UO_2 Program encompasses activities in the design, fabrication, irradiation, and evaluation of the performance of fuel rods operated in a power-producing BWR at high specific power levels to achieve various degrees of melting in the central regions of the UO_2 fuel. The objectives of the program are to demonstrate the feasibility of applying this mode of fuel operation sustained over a large percentage of the fuel lifetime to several full-size BWR fuel assemblies, to determine the condition of the fuel, and to evaluate its performance by obtaining specific data from post-irradiation examinations, both nondestructive and destructive, of selected rods at interim exposures and at end-of-life.

A.2 DESCRIPTION OF THE FUEL

Six fuel assemblies have been designed and fabricated for irradiation in the Consumers Big Rock Point reactor. A summary of the main characteristics of these assemblies is shown in Table A-1. Detailed design of the fuel was described in the previous four progress reports listed as part of the introduction to this report. The two levels of central melting in the UO_2 for which the fuel rods were designed separate the assemblies into two groups as follows (see Figure A-1).

A.2.1 Intermediate Performance Fuel

The two Intermediate Performance Fuel assemblies are designed for operation with incipient central melting of the UO_2 at rated power. Each assembly has 64 rods in an 8×8 array; of these, 36 rods are power producing; the other 28 rods, containing

vibratorily compacted depleted UO_2 , are essentially non-power-producing and are used for achieving specific thermal-hydraulic conditions in the assembly design. The power-producing rods in one assembly contain cold-pressed and sintered pellets, and those in the other assembly, vibratorily compacted powder. The power rods in each assembly are designed to run at nearly the same relative power levels. Three U-235 enrichments (4.3, 5.0, and 5.6%) are used to achieve their power flattening.

The rods in these two assemblies are all clad with 0.570-inch-o.d. $\times$ 0.035-inch-wall Zircaloy-2 tubing. A depleted UO_2 sintered insulator pellet is placed at each end of the active fuel column to minimize temperature effects at the lower end plug region and on the spring used in the gas plenum at the upper end of the rod.

The accommodation of the phase change volume expansion of the UO_2 upon melting is by a dish in one end of each pellet, providing a void volume of 5%. The powder fuel, compacted to a bulk density of 85% of theoretical, will have a 15% distributed void volume, which is adequate to accommodate the volume expansion that will occur upon melting in these rods.

A.2.2 Advanced Performance Fuel

The Advanced Performance Fuel is designed for operation with definite, but moderate, central melting of the UO_2 at rated power. There are four assemblies in this group. Each assembly has 49 rods in a 7×7 array. Of these, 29 rods are power producing; the other 20 rods, containing vibratorily compacted depleted UO_2 , are essentially non-power-producing, and are

TABLE A-1
SUMMARY OF CENTERMELT FUEL DESIGN DETAILS
AND OPERATING CONDITIONS

Fuel Assembly	UO ₂ Fuel		Number of Rods	Cladding (Zircaloy-2 Tube Reduction Process)			Condition
	Type	% TD		% U-235	Outside Diameter (in.)	Wall Thickness (in.)	
	Intermediate Performance						
D-50	Pellet	95	4.3	12	0.570	0.035	Cold Worked and Stress Relieved
(8 × 8 Lattice)	Pellet	95	5.0	16			
	Pellet	95	5.6	8			
	Powder	85	0.22	28			
D-51	Powder	85	4.3	12	0.570	0.035	Cold Worked and Stress Relieved
(8 × 8 Lattice)	Powder	85	5.0	16			
	Powder	85	5.6	8			
	Powder	85	0.22	28			
Advanced Performance							
D-52 and	Pellet	95	4.3	12	0.700	0.040	Cold Worked and Stress Relieved
D-53	Pellet	95	5.0	12			
(7 × 7 Lattice)	Pellet	95	5.6	5			
	Powder	85	0.22	20			
D-54 and	Powder	85	4.3	12	0.700	0.040	Cold Worked and Stress Relieved
D-55	Powder	85	5.0	12			
(7 × 7 Lattice)	Powder	85	5.6	5			
	Powder	85	0.22	20			

UO ₂ Fuel		Design Burnup (MWd/TU)	Design Heat Flux (Btu/h-ft ²)		Fuel Rod Power (kW/ft)	
			Normal	Overpower	Normal	Overpower
<u>Intermediate Performance</u>						
Cold Press and Sinter	Standard	20,000	450,000	550,000	19.65	24.0
	Dished		±50,000		±2.18	
Dynapak	Vibratory	20,000	450,000	550,000	19.65	24.0
	Compaction		±50,000		±2.18	
<u>Advanced Performance</u>						
Cold Press and Sinter	Standard	20,000	450,000	550,000	24.15	29.6
	0.100-in. Central Hole		±50,000		±2.68	
Dynapak	Vibratory	20,000	450,000	550,000	24.15	29.6
	Compaction		±50,000		±2.68

FIGURE A-1. UO₂ MELT FRACTION VERSUS HEAT FLUX

also used for achieving specific thermal-hydraulic conditions in the assembly design.

Sintered pellet and compacted powder UO_2 fuels are used. The power-producing rods in two assemblies will have pellet fuel, and those in the other two assemblies will have compacted powder fuel. The U-235 enrichments used in these fuels are the same as those used in the Intermediate Performance Fuel.

The rods in the four assemblies are all clad with 0.700-inch-o. d. $\times$ 0.040-inch-wall Zircaloy-2 tubing. A depleted UO_2 sintered

insulator pellet is placed at each end of the active fuel column. The enriched pellets in the rods have a 0.100-inch-diameter hole at the center to provide a void volume for the phase change volume expansion of the UO_2 on melting. The 15 percent distributed void volume in the compacted powder fuel is adequate to accommodate the volume expansion on melting in this fuel.

General dimension of the fuel rods, location of spacers and corner angle dimples are shown in Figure A-2.

FIGURE A-2. AXIAL LOCATIONS OF SPACERS AND DIMPLE IN CORNER ANGLE

REFERENCES

1. Lees, E. A., "Power Reactor High Performance UO₂ Program Progress Report Number 4, April 1, 1967-June 30, 1967," GEAP-5521, pp. 7-18, July 1967.
2. Ogawa, S. Y., Lees, E. A., and Lyons, M. F., "Fuel Design Summary and Program Status," GEAP-5591, January 1968.

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