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PRODUCTION TEST IP-333-D - SUPPLEMENT A -
IRRADIATION OF TWO DEFECTED UO₂ FUEL ELEMENT
ASSEMBLIES

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PRODUCTION TEST IP-333-D - SUPPLEMENT A
IRRADIATION OF TWO DEFECTED UO₂ FUEL ELEMENT ASSEMBLIES

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRODUCTION TEST IP-333-D - SUPPLEMENT A
IRRADIATION OF TWO DEFECTED UO₂ FUEL ELEMENT ASSEMBLIES

OBJECTIVE

To permit either the irradiation of two defected UO₂ four-rod-cluster fuel element assemblies during the same operating period, or the irradiation of the two assemblies singly during two different operating periods in KE Reactor front-to-rear test hole 3865. The density of UO₂ in one fuel element assembly is 90% of theoretical, in the other 87%. In both assemblies, the fuel element cladding is Zircaloy and the fuel material natural uranium. An artificial defect will be made in each assembly before irradiation by drilling a .005" diameter hole through the cladding near the mid-point of two of the rods in each assembly, thus a total of eight UO₂ rods, four defected and four non-defected, will be irradiated.

SUMMARY

A series of irradiations is planned which will provide Ceramic Fuels Development Operation with irradiated defected and non-defected fuel rods. From a study of the fuel rods, data can be obtained concerning the relative effect of different parameters on the hydriding of zircaloy. Parameters which might affect the rate of hydriding are core temperature, surface-heat flux, UO₂ fabrication methods, coolant water temperature, and defect size. Obtaining the desired data will require the irradiation of a test hole charge made up of defected UO₂ assemblies having different UO₂ densities, different uranium enrichments, different sized defects in the cladding. The first irradiation, the irradiation of one defected four-rod-cluster assembly containing UO₂ of 95% of theoretical density, has been successfully completed.⁽¹⁾

Authorization for future irradiations will require a knowledge of the fission products release from UO₂ of 87% and 90% of theoretical density. This authorization will permit the separate irradiation of the fuel elements to obtain this knowledge or the simultaneous irradiation of the two fuel elements if the knowledge becomes available from other tests before this test is performed.

BASIS AND JUSTIFICATION²

Investigations have shown that zircaloy-2 cladding on UO₂ fuel elements can hydride under various conditions. Zirconium hydride is brittle, and small concentrations in the cladding can cause failure of a fuel element. Ex-reactor tests have failed to produce the results found from in-reactor tests. Defected UO₂ fuel elements, autoclaved in steam at temperatures below 400°F, have shown little or no indication of hydriding. Hydriding, however, has been detected in irradiated defected fuel elements with coolant temperatures below 100°F. This implies that a temperature gradient from fuel core to cladding surface is required to hydride zircaloy cladding. Heat flux, core temperature, coolant water temperature, and defect size apparently control the rate at which zircaloy will hydride, but the relative importance of each of these parameters has not been determined. If several defected fuel elements are irradiated at one time under controlled conditions, the parameters can be properly evaluated.

The evaluation of the parameters is of importance to the Plutonium Recycle Program. The fuel elements for the first loading of the PRTR are clad in zircaloy-2. Westinghouse has developed a nickel-free zirconium alloy, zircaloy-4, which allegedly is less susceptible to hydriding than zircaloy-2. It is important that the relative resistance of Zr-2 and Zr-4 to hydriding be determined. If it is found that neither Zr-2 nor Zr-4 can withstand hydriding satisfactorily, then some other material, such as stainless steel, will be considered as a cladding material for future PRTR loadings.

IRRADIATION FACILITY

The fuel element assemblies will be irradiated in the single pass test hole 3865-KE. The test hole contains two concentric aluminum tubes with coolant flow, independent of the flow to the fuel elements, between the tubes. The design of the rear nozzle will permit the remote discharge of the fuel element assemblies. Normally, the water from the bore of the test hole empties into a rear header; however, fission products entering the water from the defected fuel elements would disrupt the reactor gamma monitor system and increase the radioactivity levels in the retention basins. Therefore, the header was bypassed by extending the outlet connector from the nozzle to the rear face closed drain which leads to the 107 crib.

FISSION PRODUCTS RELEASE TO COOLANT

The original production test authorization permitted the irradiation of one defected four-rod-cluster UO₂ assembly. The UO₂ was compacted to 95% of theoretical density, the uranium was natural. Two of the four rods each had a .005" diameter hole drilled through the cladding. The assemblies to be irradiated are the same except that the density of UO₂ in one is 90% and in the other 87% of the theoretical density. The fission products in the coolant from the first irradiation which were definitely identified were Xe¹³³, Xe¹³⁵ and I¹³¹. The following table presents the calculated concentrations upon which the original production test was based, the estimated concentration in the effluent water from radio-chemical analysis, and the estimated concentration in the effluent water for the simultaneous irradiation of two defected assemblies. All concentrations are in picacuries (pc)*/cc of effluent water.

Isotope	Calc. Conc.	Est. Conc.	Estimated Concentration
	1st Irrad., 95% _p	1st Irrad., 95% _p	Two Assemblies, 90% & 87% _p
Xe ¹³³	8000	0.03	0.6
Xe ¹³⁵	2000	0.03	0.6
I ¹³¹	2000	0.003	0.06

The calculated concentration values for the first irradiation were based on irradiations in which the fuel element surface heat fluxes were two and three times higher.³ This factor and the assumption that the UO₂ compactions in the first irradiation had better fission product retention characteristics than the WAPD compactions could account for the difference.

* pc = 10⁻¹² curies

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The estimated concentrations for the simultaneous irradiation of the two fuel elements were based on WAPD data.³⁴ Tests performed by WAPD demonstrated that a reduction in UO_2 density from 95% of theoretical to 80% would increase the rate of fission products release by about a factor of 10.

HAZARDS

The prime hazard associated with this irradiation is the occurrence of a cladding failure which would allow particulate UO_2 and large amounts of fission products to enter the coolant stream. WAPD's experience with the irradiation of defected and non-defected fuel elements demonstrated that the most likely causes of failure are embrittlement of the cladding by hydriding which generally occurs in defective elements at high surface heat fluxes and by a phenomena termed water-logging.

A total of four failures were reported by WAPD from the irradiation of 35 non-defected and 24 defected fuel elements. Three of these failures were probably due to excessive hydriding of the cladding. In two of these elements (one defected and one non-defected) center melting had occurred; center melting is not expected in the present test. The third failure happened during irradiation at a surface heat flux of 682,000 B/hr-ft². The surface heat flux in the proposed test is expected to be less than 200,000 B/hr-ft².

With a defect present in the cladding, water may enter the rod during reactor shutdown and fill the interconnected voids present in the UO_2 . This phenomena is termed water-logging. The amount of water entering the rod is a function of the density of the UO_2 , and the amount of interconnected porosity. Upon subsequent reactor startup, the trapped water could flash to steam and produce sufficiently high pressures to expand or even rupture the cladding if the steam could not escape fast enough out of the original entrance hole to relieve the pressure. Of the four failures experienced by WAPD, one rod (defected) failed by the water-logging mechanism. The rod which failed contained low density UO_2 (80% of theoretical) and therefore considerable interconnected space was available for the entrapped water.

General Electric, Hanford Laboratories Operation, recently irradiated a defected UO_2 fuel element in the MTR.⁵ During the irradiation the reactor was cycled from full power to essentially zero power twice. There was no indication of particulate UO_2 or large amounts of fission products released to the coolant stream. A rod-in-tube element containing UO_2 having a density of 86% of theoretical was irradiated in the ETR.⁶ This rod failed with fragmentation of the cladding occurring but no particulate matter was released. Normal loop flow adequately cleaned up the system; no decontamination procedures were necessary.

Based on a study of the available data, the conclusion is that the most likely cause of a failure is the hydrogen embrittlement and subsequent fragmentation of the cladding, but the probability of this occurring during the irradiation is not great. If a failure should occur, the design of the test facility will permit the fuel element assemblies to be discharged remotely after reactor shutdown.

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TEST DETAILS1. Defected Fuel Element Assemblies

Each assembly consists of four rods maintained in a cluster configuration by spacers located at each end of the rods. Each rod is 11.0 inches long and has an O.D. of 0.563 inch. Two of the rods in each assembly are clad in zircaloy-2, two in zircaloy-4, cladding thickness is 0.028 inch. The fuel rods in one assembly contain UO_2 of 90% of theoretical density; the rods in the other assembly contain UO_2 of 87% of theoretical density. The fuel material is natural UO_2 . One of the zircaloy-2 and one of the zircaloy-4 rods in each assembly have a .005 inch diameter hole drilled through the cladding near the mid-point, the other rods are not defected. See Figure 2, page 6A.

Each four-rod-cluster is mounted in a 12-inch-long aluminum basket, 2.814-inch O.D. by 2.318-inch I.D., to obtain the desired hydraulic characteristics and to protect the clustered rods during charging and discharging.

2. Test Hole Loading

Either one or two baskets will be charged into the test hole and positioned near the mid-point of the active zone. Perforated tubular spacers will be located downstream of the assembly. (See Appendix I for charge-discharge procedures).

3. Thermal Hydraulics (see Figure 1, page 6, and Appendix II)

	<u>One F.E. (90% ρ)</u>	<u>Two F.E. (87% ρ & 90% ρ)</u>
a. Heat generated	28 kw	55 kw
b. Upstream orifice	.360-inch dia.	.360-inch dia.
c. Downstream orifice	.397-inch dia.	.397-inch dia.
d. Equilib. Panellit Pressure	225 psi	225 psi
e. Panellit pressure, low scram trip	180 psi	180 psi
f. Panellit pressure, high scram trip	270 psi	270 psi
g. Clad surface temp. at low trip	111°C	111°C
h. Clad surface temp. at high trip	112°C	112°C
i. Sat. temp. at low trip	193°C	193°C
j. Sat. temp. at high trip	212°C	212°C
k. UO_2 surface temp. at equil. op.	208°C	208°C
l. Coolant temp. rise	2°C	4°C

The test hole will be fitted with a 10-90 Panellit gauge which will be connected into the scram circuit. The outlet thermohm will be removed and the hole plugged to reduce the chance of leakage on the rear face.

4. Monitoring

The gamma activity from the fission products in the coolant stream will be continuously monitored by a geiger tube. Periodically, water samples will be taken and analyzed on a multi-channel analyzer to determine the presence and concentrations of specific isotopes. The gamma monitor and water sample manifold are located on the north wall of the X-1 level. A tubing

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Test hole loading: One or two baskets each containing one
 11-inch-long, four-rad-cluster assembly and perforated spacers downstream.

Figure 1

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Figure 2
Defected Fuel Element Assembly

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line from the outlet nozzle connector pipes water at a low flow rate through the manifold and to the rear face drain. The signal from the geiger tube will be continuously recorded on a strip chart recorder located in the reactor control room. An alarm in the control room will be energized if the activity from the effluent water exceeds a pre-set limit. Based on experience from the first irradiation, radiation levels in the vicinity of the manifold are expected to be less than 25 mr/hr.

The stack air also will be periodically sampled and analyzed.

5. Shutdown Time

	<u>Time in Hours</u>	
	<u>Front Elevator</u>	<u>Rear Elevator</u>
Charging test	0	1
Discharging Test	1	1

If the fuel element assemblies are irradiated separately, the above times will be required during two different reactor shutdowns. In addition, there is a risk of an unscheduled reactor outage should it become necessary to prematurely discharge the test.

6. Reactivity Loss

The test hole is empty except for water flow. The charging of enrichment metal to compensate for reactivity loss is authorized as is the disposal or retention of the columns upon termination of the test.

7. Exposure

The irradiation period will be for less than six months.

8. Data Required

The data required are the MWD exposure for the four surrounding tubes.

9. Expected Fuel Element Behavior

- a. During Operation - Sudden increases in radioactivity levels can be expected during increases and decreases in reactor power. These increases are usually several times the equilibrium values and are of short duration. Soon after a new power level is attained and held for a period of time, the gamma activity can be expected to decrease to an equilibrium value.
- b. During Shutdown - The escape of fission products from the fuel elements during shutdown should be negligible; therefore, the radiation level from the rear face connector should not contribute significantly to the rear face activity.

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10. Operating Limits Due to High Gamma Activity

The gamma monitor will not be in the scram circuit. The equilibrium readings should be below 200 mr/hr. If the equilibrium value increases by a factor of 10 and remains at the higher level for 30 minutes, the reactor should be shut down and the test discharged.

RESPONSIBILITIES

Irradiation Processing Department

1. KE Processing Operation

- a. Charge and discharge the supporting enrichment columns authorized by this Production Test.
- b. Maintain safety and production continuity of the reactor, except for conditions associated with this test.

2. Irradiation Testing

- a. Charge and discharge the defected fuel element assemblies.
- b. Collect the data.
- c. Take water samples.
- d. Provide instrumentation.
- e. Operate the test facility.
- f. Maintain reactor safety and production continuity as determined by the conditions of this test.

3. Pile Physics

- a. Specify the number and location of the supporting enrichment columns authorized by this Production Test.
- b. Specify disposition of the enrichment columns upon termination of this test.

Hanford Laboratories Operation

1. Ceramic Fuels Development

- a. Provide the defected fuel element assemblies.
- b. Write the final report.

R K Marshall
Irradiation Testing Operation

RK Marshall:jgl

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APPENDIX I

Procedures

1. Charging Fuel Element Assemblies

- a. Reduce water pressure.
- b. Remove test hole rear nozzle cap.
- c. Charge baskets containing fuel elements from the rear.
- d. Charge 16, 18"-long perforated tubular spacers from the rear.
- e. Install rear nozzle cap.

2. Discharging Fuel Element Assemblies

- a. Reduce water pressure
- b. Remove rear nozzle cap and install tip-off.
- c. Raise elevator above test hole.
- d. Remove front nozzle cap.
- e. Push assembly and perforated spacers into discharge basin.
- f. Install front nozzle cap.
- g. Remove tip-off and install rear nozzle cap.

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APPENDIX II

1. Heat Generated and Coolant Temperature Rise

Data:

Diameter of UO ₂ -	0.507"
Length of UO ₂ -	10.5"
Length of clad rod -	11.0"
Diameter of clad rod -	0.563"
Density of 100% dense UO ₂ -	10.9 gms/cm ³
Density of UO ₂ in test -	90% and 87% of theoretical
Flow through basket -	55 gpm
Neutron flux in surrounding tubes at fuel element location -	3.80×10^{13} m/cm ² -sec.
Ratio of volume average flux in UO ₂ to flux in surrounding tubes -	1.8
Heat generated by one gram of U ²³⁵ at 1×10^{13} m/cm ² -sec. -	0.484 KW

Calculations:

Power/ft-rod

a. 90% of theoretical ρ

$$\frac{\pi}{4} (.507)^2 (12) (16.4) (10.9) (.90) \left(\frac{238}{270}\right) (.0071) (3.8) (1.8) (.484) = 8.1 \text{ KW/ft-rod}$$

b. 87% of theoretical ρ

$$(8.1) \left(\frac{.87}{.90}\right) = 7.8 \text{ KW/ft-rod}$$

Total Tube Power

a. 90% of theoretical ρ

$$\frac{(8.1)(4)(10.5)}{12} = 28 \text{ KW}$$

b. 87% of theoretical ρ

$$\frac{(7.8)(4)(10.5)}{12} = 27 \text{ KW}$$

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- c. For simultaneous irradiation of one 90% and one 87% of theoretical dense UO₂.

$$\text{Total KW} = 28 + 27 = 55 \text{ KW}$$

Surface Heat Flux

- a. 90% of theoretical ρ

$$\frac{(8.1)(3413)(144)}{\pi (.563)(12)} = 188,000 \frac{\text{B}}{\text{hr ft}^2}$$

- b. 87% of theoretical ρ

$$(188,000) \frac{(7.8)}{(8.1)} = 181,000 \frac{\text{B}}{\text{hr ft}^2}$$

Coolant Temperature Rise

- a. 90% or 87% of theoretical ρ

$$\frac{(28)(3413)}{(55)(498)(1)(1.8)} = 2 \text{ C}$$

- b. 90% and 87% of theoretical ρ

$$\frac{(55)(3413)}{(55)(498)(1)(1.8)} = 4 \text{ C}$$

2. Cladding Surface Temperature for Worst Case - Low Scram Trip with 90% of Theoretical ρ

Data:

Basket I.D.	2.318"
Fuel element O.D. -	0.563"
Bulk water temp. -	15°C
Flow -	49 gpm (from Figure 1)
Water pressure -	180 psi (from Figure 1)
Saturation temp. -	193°C
Surface heat flux	188,000 B/hr-ft ²

Calculations:

Correlation used $h = K_t \frac{V^{.8}}{De^{.2}}$

- from Fuel Element Tech. Manual
Figure III-2, HW-40000.

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h in B/hr-ft² F

V in ft/sec

D_e in ft

K_t at 15°C = 165

$$V = \frac{(4)(144)(48)(2.228 \times 10^{-3})}{\pi [(2.318)^2 - 4(.563)^2]} = 4.77 \text{ ft/sec}$$

$$D_e = \frac{[(2.318)^2 - 4(.563)^2]}{[2.318 + 4(.563)](12)} = .0747 \text{ ft}$$

$$h = \frac{165(4.77)^{.8}}{(.0747)^{.2}} = 1090 \text{ B/hr-ft}^2 \text{ F}$$

$$\Delta T = \frac{188,000}{(1090)(1.8)} = 96^\circ\text{C}$$

Surface temperature of cladding: $96 + 15 = 111^\circ\text{C}$

3. UO₂ Surface Temperature

Data:

Bulk water temperature -	15°C
Water film ΔT -	96°C
Zircaloy thermal cond. -	11.8 B/hr-ft F
Cladding - UO ₂ interface heat transfer coeff. -	2000 B/hr-ft ² F

a. 90% of theoretical ρ - power/ft.rod = 8.1 kw

$$\text{Cladding } \Delta T = \frac{(8.1)(3413) \ln \frac{.563}{.507}}{2\pi (11.8)} = 39^\circ\text{C}$$

$$\text{Cladding - UO}_2 \text{ interface } \Delta T = \frac{(8.1)(3413)(144)}{\pi (.507)(12)(2000)(1.8)} = 58^\circ\text{C}$$

UO₂ surface temperature

$$15 + 96 + 39 + 58 = 208^\circ\text{C}$$

b. 87% of theoretical ρ - power/ft.rod = 7.8 kw

$$\text{Cladding } \Delta T = \frac{(7.8)(3413) \ln \frac{.563}{.507}}{2\pi (11.8)} = 37^\circ\text{C}$$

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$$\text{Cladding - UO}_2 \text{ interface } \Delta T = \frac{(7.8) (3413) (144)}{\pi (.507) (12) (2000) (1.8)} = 56 \text{ } ^\circ\text{C}$$

UO₂ surface temperature

$$15 + 96 + 37 + 56 = 204 \text{ } ^\circ\text{C}$$

4. Flow Split

Data:

Test Hole Tube I.D.	2.875 inch
Basket O.D.	2.814 inch
Basket I.D.	2.318 inch
Basket length	12.0 inch
Fuel element O.D.	0.563 inch
Fuel element length	11.5 inch

Calculations:

Equations (from McAdams, Third edition, pages 155 and 158)

$$\Delta p = \frac{f G^2 L}{2 \rho g r}, \quad f = \frac{.046}{N^{.2}} \quad \text{for } N = 5000 - 200,000, \quad N = \frac{4r G}{\mu}$$

f = friction factor

G = mass velocity = $\frac{W}{A}$

L = length

p = density

g = gravitational acceleration

r = hydraulic radius

N = Reynolds No.

μ = viscosity

W = mass flow rate

A = cross sectional flow area

$$P_1 = P_2$$

$$\mu_1 = \mu_2$$

$$\rho_1 = \rho_2$$

Assume $L_1 = L_2$ (conservative assumption)

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Constants cancel

$$\left(\frac{G_1}{G_2}\right)^{1.8} = \left(\frac{r_1}{r_2}\right)^{1.2} \quad \begin{array}{l} \text{Substituting } G = \frac{W}{A} \text{ and} \\ \text{Simplifying} \end{array}$$

$$\frac{W_1}{W_2} = \frac{A_1}{A_2} \left(\frac{r_1}{r_2}\right)^{.667}$$

$$\frac{A_1}{A_2} = \frac{(2.318)^2 - 4(.563)^2}{(2.875)^2 - (2.814)^2} = 11.1$$

$$\left(\frac{r_1}{r_2}\right)^{.667} = \left\{ \frac{[(2.318)^2 - 4(.563)^2] [(2.875 + 2.814)]}{[2.318 + 4(.563)] [(2.875)^2 - (2.814)^2]} \right\}^{.667} = 5.75$$

$$\frac{W_1}{W_2} = (11.1) (5.75) = 63.8$$

For this case $\frac{W_1}{W_2} = \frac{F_1}{F_2}$ where F = flow in GPM

$$\frac{F_1}{F_2} = 63.8$$

at equilibrium

$$F_1 + F_2 = 56$$

$$F_2 = 56 - 63.8 F_2$$

$$64.8 F_2 = 56$$

$$F_2 = .86$$

Flow through the basket assumed to be 55 GPM

5. Pressure drop through one basket containing one four-rod-cluster. Data and nomenclature same as D.

Assumed flow through basket: 60 gpm.

Physical properties of water evaluated at 60 °F.

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$$\text{Reynolds Number (N)} = \frac{4 r G}{b \mu} = \frac{4 W}{b \mu} \quad b = \text{wetted perimeter}$$

$$W = \frac{(60) (8.34)}{60} = 8.34 \frac{\text{lb}}{\text{sec.}}$$

$$b = \frac{\pi}{12} [2.318 + 4 (.563)] = 1.20 \text{ ft.}$$

$$\mu = 7.4 \times 10^{-4} \frac{\text{lb}}{\text{sec.ft.}}$$

$$N = \frac{(4) (8.34)}{(1.20) (7.4 \times 10^{-4})} = 3.75 \times 10^4$$

$$f = .0055$$

$$\Delta p = \frac{f G^2 L}{2g \rho r} = \frac{f W^2 L b}{2g \rho A^3} \quad b = \text{wetted perimeter}$$

$$W = 8.34 \frac{\text{lb}}{\text{sec.}}$$

$$L = 1 \text{ ft.}$$

$$b = \frac{\pi (2.318) + (4) (.563)}{12} = 1.20 \text{ ft.}$$

$$A = \frac{\pi (2.318)^2 - 4(.563)^2}{(4) (144)} = .0223 \text{ ft.}^2$$

$$\Delta p = \frac{(.0055) (8.34)^2 (1.20) (1)}{(2) (32.2) (62.4) (.0223)^3 (144)}$$

$$\Delta p = .072 \text{ psi}$$

The pressure drop across the basket can be neglected.

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END

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