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May 28, 1962

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PRODUCTION TEST IP-508-D,
IRRADIATION OF A POROUS STAINLESS STEEL CLAD
FUEL ELEMENT ASSEMBLY

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PRODUCTION TEST IP-508-D
IRRADIATION OF A POROUS STAINLESS STEEL
CLAD FUEL ELEMENT ASSEMBLY

J. H. Cox

May 28, 1962

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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PRODUCTION TEST IP-508-D
IRRADIATION OF A POROUS STAINLESS STEEL
CLAD FUEL ELEMENT ASSEMBLY

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OBJECTIVE

The objective of this authorization is to permit the irradiation of one natural UO_2 fuel element clad in porous sintered stainless steel.

BASIS AND JUSTIFICATION⁽¹⁾

A large number of irradiation tests have been conducted with UO_2 fuel elements to characterize the behavior of Zircaloy cladding when cooling water is allowed to penetrate the cladding to the fuel core. The test element is designed so that cooling water will be in contact with the surface of the UO_2 fuel core during irradiation. The irradiation behavior of the UO_2 will be compared to the behavior of similarly irradiated, non-defected UO_2 fuel elements. Several characteristics are of particular interest:

1. Previous investigations have shown that fission fragments formed in UO_2 fuel cores tend to migrate from their point of formation. The fission fragment distribution in this element will be compared with that observed in other irradiated UO_2 fuel cores.
2. This test will provide a direct measure of fission products released from UO_2 during irradiation.
3. Sintered UO_2 is quite resistant to erosion and is, in general, chemically stable in a water environment. This test will provide data concerning the dissolution rate of UO_2 in water during irradiation.
4. Since the coolant will be in direct contact with the fuel, the average temperature of the fuel will be comparatively low. Therefore, the in-reactor sintering characteristics of the fuel should be different from those observed in "normal" fuel elements.

IRRADIATION FACILITY

The fuel element assembly will be irradiated in the single-pass test tube 4268 in KE Reactor. The reactor front face header supplies coolant to the test tube. The test tube outlet piping bypasses the reactor outlet header and conducts the coolant to the rear face drain which leads to the 107 crib, as shown in HW-65618. The effluent water cannot flow to the reactor header because fission products from the porous clad fuel element would disrupt the reactor gamma monitor.

Turbulent water flow must be supplied to the test tube at all times to prevent the gaseous fission products from the fuel element from collecting in the outlet piping and causing high radiation levels on the rear face. At some header pressure below 10 psig, the flow changes from turbulent to laminar. Therefore, water from a source independent of the header must be supplied to the test tube when the header pressure is less than 10 psig.

The modifications required to provide an independent supply of turbulent water include locating a ball valve in the front header orifice cage, equipping the inlet nozzle with a ball valve, a check valve, a quick connective coupling, and extending piping from the inlet nozzle fittings to the source of water.

When the header pressure is in the 10-60 psig range, an operator will open the nozzle ball valve and will close the orifice cage ball valve. These operations isolate the facility from the header and provide a flow of turbulent water in the test tube.

The check valve prevents inadvertent back flow in the auxiliary supply line. The quick-connective coupling allows the supply piping to be connected to and disconnected from the inlet fittings.

The front header orifice cage has been pressure tested to 600 psi to verify the integrity of the welds. The ball valve in the orifice cage is equipped with a ring to lock the valve in the open position during reactor operation.

A mockup of the auxiliary water supply line was set up in the F-Area flow lab. Using the minimum supply pressure available at KE (60 psig), a flow of 9.0 gpm is possible (a 3 gpm flow is required for turbulent flow). A schematic of the line is shown in Figure 1. When the test is charged, a charge letter will be sent to the KE Reactor Analyst giving a detailed procedure for the independent water supply valving.

FISSION PRODUCTS RELEASE TO COOLANT

The release of fission products per kilowatt for the porous stainless steel clad fuel element is expected to be approximately the same as that experienced during the irradiation of one defected 4 per cent enriched UO_2 cluster fuel element.⁽²⁾ This should be a conservative estimate because of the difference in the UO_2 fuel structure. The value from the defected enriched cluster used in Table I, below, came from an unsintered vibrationally compacted UO_2 powder element while the fuel contained with the porous cladding was fabricated from high-density sintered UO_2 pellets. The fission product retention has been observed to be markedly greater in sintered pellets than in vibrationally compacted UO_2 powders. Further, post-irradiation analysis indicated that during the irradiation of the defected rod, the cladding had split open at one end of the rod and the escaped fission products were released without restraint. The value given in Table I was the highest of any of the periodic samples taken throughout the history of the defected 4 per cent enriched irradiation.

Table I presents a comparison of the I^{131} concentration per kw in the effluent water of the 4 per cent enriched irradiation with that expected from the porous stainless steel clad UO_2 .

TABLE I

FISSION PRODUCT RELEASE

Test	$\mu c/cc-kw$ $I-131$		
	Calculated	Results of Lab. Analysis	Predicted (Corrected for Lab. Analysis)
Defected 4% enriched cluster	1.8×10^{-4}	6.4×10^{-4}	-----
Porous stainless steel clad UO_2	1.9×10^{-4}	-----	6.7×10^{-4}

The total concentration of nuclides per cc in the effluent water from the proposed test will be approximately three times greater than that previously experienced because the power generation is higher.

Independent Water Supply - 1268 IC

1. If the porous stainless steel jacket should plug completely, a pressure buildup in the jacket can be expected. The worst case would be a failure of the jacket. Tests run by Ceramic Fuels Development Operation indicate that if a failure did occur, the jacket would fail by cracking; the stainless steel jacket appeared quite ductile and gave no indication of brittle shattering.
2. During a reactor shutdown period if the flow in the tube were allowed to reach laminar, a fission gas buildup may be encountered in the rear face piping. To prevent this, a water source completely independent of the header will be used during reactor shutdown to ensure turbulent flow in the tube at all times. Therefore, an appreciable increase in the radioactivity level is not expected on the rear face during shutdown.

TEST DETAILS

1. Fuel Element Assembly

Sintered pellets of UO_2 are clad in a porous stainless steel tube. The length of the tube cladding is $5\frac{7}{8}$ ", the inside diameter is 1.375", and the outside diameter is 1.500". End caps are welded to the tube. This assembly is concentrically mounted in an aluminum tube which is called a basket. The basket has a 20-mesh wire screen disc affixed to each end. The length of the basket is $6\frac{3}{16}$ ", the inside diameter is 1.680", and the outside diameter is 2.020". (SEE FIGURE 2).

2. Test Tube Loading

The assembly will be positioned near the mid-point of the active zone. Twenty-three feet of perforated tubular spacers will be located downstream of the fuel element.

3. Thermal Hydraulics

Diameter of UO_2	1.375 $\pm$.002"
Length of UO_2	5.750"
Diameter of clad sample	1.500"
Length of clad sample	5.875"
Density of UO_2 (.95)(10.9)	10.35 g/cc
Basket ID	1.680"
Basket OD	2.02"
Test Tube ID	2.100"

Equilibrium

Flow rate	59 gpm
Tube pressure	236 psig
Panellit pressure	250 psig

Normal High Trip Scram

Flow rate	51 gpm
Tube pressure	275 psig
Panellit pressure	290 psig

Maximum High Trip Scram

Flow rate	42 gpm
Tube pressure	315 psig
Panellit pressure	325 psig

Normal Low Trip Scram

Flow rate	53 gpm
Tube pressure	195 psig
Panellit pressure	210 psig

Minimum Low Trip Scram

Flow rate	48 gpm
Tube pressure	162 psig
Panellit pressure	175 psig

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FIGURE 2

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FIGURE 3

TEST HOLE HYDRAULIC CHARACTERISTICS

FIGURE 4

PANELLIT PRESSURE VERSUS INLET TUBE PRESSURE

Upstream Orifice .405"

Downstream Orifice .406"

400

300

200

100

50

Panellit Pressure

Maximum High Trip Scram

Normal High Trip Scram

Equilibrium Point

Normal Low Trip Scram

Minimum Low Trip Scram

Inlet Tube Pressure psig

50

100

200

300

400

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Neutron flux in surrounding tubes at fuel element location 4.1×10^{13} /cm²/sec.

Ratio of volume average flux in UO₂ to flux in surrounding tubes-----1.57

Heat generated by one gram of U²³⁵ at 1×10^{13} /cm²/sec.-----484 kw

a. Total Power Generated - 26.9 kw.

b. Surface Heat Flux - 5.34×10^5 BTU/hr.ft.²

c. Coolant Temperature Rise at Equilibrium - 3.1 C.

d. Flow Distribution - Flow data were obtained from a flow mockup of an exact model of the sample to be irradiated.

Basket annulus-----40 %

Fuel element annulus-----60 %

e. Burnout Heat Flux at Minimum Low Trip

Total flow = 48 gpm

P = 162 psig

Using the Bernath⁽³⁾ equation:

$$(h)_{bo} = 5710 \left(\frac{De}{Di} \right)^{.6} + 48 \frac{V}{De^{.6}}$$

$$= 5710 \left(\frac{.015}{.125} \right)^{.6} + \frac{48}{(.015)^{.6}} \frac{(48)(.6)(2.228)10^{-3}}{(3.11)10^{-3}}$$
$$= 13,870 \text{ Pcu/hr ft.}^2 \text{ C}$$

$$(T_w)_{bo} = 57 \ln P - 54 \frac{P}{P + 15} - \frac{V}{4 \frac{De}{Di}^{.6}}$$

$$= 57 \ln 162 - 54 \left(\frac{162}{177} \right) - \frac{1}{(4)(.278)} \frac{(48)(.6)(2.228)10^{-3}}{(3.11)10^{-3}}$$
$$= 222 \text{ C}$$

$$(T_b)_{bo} = 48 \text{ C (from Bernath trial and error method)}$$

$$\frac{Q}{A}_{bo} = (h)_{bo} [(T_w)_{bo} - (T_b)_{bo}]$$

$$= 13870 (222 - 48)$$

$$= 2.41 \times 10^6 \text{ Pcu/hr.ft.}^2$$

$$= 4.34 \times 10^6 \text{ BTU/hr.ft.}^2$$

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f. Burnout Heat Flux at Maximum High Trip

$$\text{Total flow} = 42 \text{ gpm}$$

$$P = 315 \text{ psig}$$

$$(h)_{bo} = 5710 \frac{(.015)^{.6} + 48}{(.125)} \cdot \frac{(42)(.6)(2.228)10^{-3}}{(3.11)10^{-3}} \cdot \frac{1}{(.015)^{.6}}$$

$$= 12,320 \text{ Pcu/hr.ft.}^2 \text{ C}$$

$$(T_w)_{bo} = 57 \ln 315 - 54 \frac{315}{330} - \frac{1}{(4) \cdot \frac{(.015)^{.6}}{(.125)}} \cdot \frac{(42)(.6)(2.228)10^{-3}}{(3.11)10^{-3}}$$

$$= 260 \text{ C}$$

$$(T_b)_{bo} = 53 \text{ C}$$

∴

$$\left(\frac{Q}{A} \right)_{bo} = 12,320(260 - 53)$$

$$= 2.49 \times 10^6 \text{ Pcu/hr.ft.}^2$$

$$= 4.47 \times 10^6 \text{ BTU/hr.ft.}^2$$

g. Surface Temperature at Maximum High Trip

$$T = 137 \text{ C}$$

h. Surface Temperature at Minimum Low Trip

$$T = 133 \text{ C}$$

4. Monitoring

- a. Panellit Protection - A 10-90 Panellit gauge monitors the flow. The gauge is in the scram circuit.
- b. Test Tube Outlet Temperature - The outlet thermohm will not be installed. The hole is plugged to reduce the chance of leakage of fission products on the rear face.
- c. Fission Product Activity - A 1/4-inch OD tube conducts a small part of the effluent water through a monitor and sample station located on the X-1 level to the rear face drain, as shown in HW-65618. This water is continuously monitored for gamma activity and for delayed neutrons. The signals from the gamma and delayed neutron sample box on the X-1 level will be read and recorded on the X-0 level. Higher than normal gamma or delayed neutron activity will energize an alarm in the reactor control room. Neither the gamma monitor nor the delayed neutron monitor are in the scram circuit.

Periodically, water samples will be taken and radiochemical analyses performed to determine the presence and concentrations of specific nuclides.

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5. Shutdown Time

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

There is a risk of an unscheduled reactor outage should it become necessary to discharge the test early.

6. Special Discharge Procedure

The sample will be discharged into the basin by use of a pushpole. Immediately after discharge, the sample will be placed in a special container provided by Irradiation Testing and designed to capture any fission gases; the container with the sample will be stored in the basin.

7. Reactivity Change

None.

8. Exposure

The irradiation period for the assembly will be less than three months.

9. Data Required

The data required are the MWD/T exposures for the four surrounding tubes.

10. Expected Fuel Element Behavior

a. During Operation

Sudden increases in radioactivity levels can be expected during changes in reactor power. These increases are usually several times the equilibrium values and are of short duration. After a new power level is attained and held for a short time, the gamma activity can be expected to decrease to an equilibrium value.

b. During Shutdown

An independent water source will permit turbulent flow in the tube during shutdown. Any fission products from the sample during shutdown will be carried out by the turbulent flow. As a result, the radiation level from the outlet connector should not contribute significantly to the rear face activity.

11. Conditions For Reactor Shutdown

a. Significant reduction or increase in tube coolant flow will cause the Panellit gauge to trip the reactor safety circuit.

b. The gamma monitor and the delayed neutron monitor will not be in the safety circuit. If, after the reactor has reached equilibrium, the readings on both the gross gamma monitor and the delayed neutron monitor are increased by a factor of 100 for a period of 30 minutes, the reactor should be shut down.

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RESPONSIBILITIES

IRRADIATION PROCESSING DEPARTMENT - KE Processing Operation

- a. Open and close ball valves as required in this Production Test.
- b. Maintain safety and production continuity of the reactor except for conditions associated with this test.

IRRADIATION PROCESSING DEPARTMENT - Irradiation Testing

- a. Charge and discharge the defected fuel element assembly.
- 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.
- g. Install independent water supply.
- h. Return test facility to normal condition after the test.

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