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HW-65618

GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

June 14, 1960

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TITLE

COMMENT ISSUE, PRODUCTION TEST IP-333-D:
IRRADIATION OF ONE DEFECTED UO₂ FUEL
ELEMENT ASSEMBLY

4

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AUTHOR

R. K. Marshall

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COMMENT ISSUE 1-1
PRODUCTION TEST IP-333-D:
IRRADIATION OF ONE DEFECTED UO₂ FUEL ELEMENT ASSEMBLY

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June 14, 1960

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CG-PR-2, 1-27-94

By *J. E. Savely 2-25-95*

Verified By *Jerry Maley*
3-5-94

HANFORD ATOMIC PRODUCTS OPERATION
RICHLAND, WASHINGTON

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COMMENT ISSUE
PRODUCTION TEST IP-333-D
IRRADIATION OF ONE DEFECTED UO₂ FUEL ELEMENT ASSEMBLY

OBJECTIVE

To permit the irradiation of one dummy fuel element assembly for one operating period and to permit, during a subsequent operating period, the irradiation of one defected, four-rod-cluster, UO₂ fuel element assembly, in a KE front-to-rear test hole. The fuel material is natural UO₂ of 95 per cent theoretical density; the cladding is zircaloy. The defect in the assembly is artificial and will be made before irradiation by drilling a .005" diameter hole through the cladding near the mid-point of two of the rods.

BASIS AND JUSTIFICATION¹

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.

In order to obtain the desired information, it will be necessary to irradiate several defected UO₂ fuel elements at one time. This test, an irradiation of one 95 per cent of theoretical density UO₂, is the first in a series of test leading to the irradiation of several defected assemblies at one time. The second irradiation will be a test of an 88 per cent of theoretical density UO₂; the third a test of 85 per cent of theoretical density UO₂. A Production Test Authorization Approval based on data obtained from each previous irradiation will be obtained for each succeeding irradiation.

IRRADIATION FACILITY

The fuel element assembly will be irradiated in the single pass test hole 3674-KE or 3865-KE. Each test hole contains two concentric aluminum tubes with coolant

flow, independent of the flow to the fuel element, between the tubes. The design of the rear nozzle for each test hole will permit the remote discharge of the fuel element assembly. 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 element would disrupt the reactor gamma monitor system. It will, therefore, be necessary to bypass the header and extend the outlet connector from the nozzle to the rear face closed drain which leads to the Pluto crib.

FISSION PRODUCTS RELEASE TO COOLANT

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1. Mechanism of Release

Radioactive material is released from a defected fuel element by one or all of the following processes:

- (a) Solid state diffusion of fission products in the fuel lattice;
- (b) Recoil of fission products from the fuel material; and
- (c) Corrosion and/or erosion of particulate UO_2 from the exposed surface.

WAPD has demonstrated that the release of fission products by recoils and the release of particulate UO_2 by erosion-corrosion are not significant.^{2,3} General Electric, Hanford Laboratories Operation, recently irradiated a defected UO_2 fuel element in the MTR in which the release of particulate UO_2 to the coolant by the erosion-corrosion mechanism was not significant.⁴ The primary concern is the release of gaseous fission products to the coolant stream by solid state diffusion.

The results of the subsequent discussion were based on the assumption that fission products release would be by solid state diffusion and that contribution to release by the recoil and erosion-corrosion mechanisms would not be significant.

2. Elements Available for Release and Calculated Magnitude of Release of Specific Nuclides

The gaseous fission products which will be available for release to the coolant during reactor operation are krypton, xenon, iodine, bromine, cesium and rubidium. The available quantitative data on fission products release which are most applicable to this test are those of WAPD.² This document reports the results of a large number of irradiations made in high pressure-high temperature water loops at the MTR and NRX reactors. Included in the document are the results of simultaneous irradiation-corrosion tests of 35 non-defected and 24 defected Zircaloy-2 clad UO_2 fuel rods. The release of fission products through defects in the cladding was correlated by use of an escape rate coefficient defined as the fractional amount of each nuclide present within the fuel escaping per unit time. An escape rate coefficient of $2 \times 10^{-8} \text{ sec}^{-1}$ for Xe^{133} is reported for a defected UO_2 rod of 95 per cent theoretical density. Based on WAPD's escape rate coefficient, a release of 1.3 per cent of Xe^{133} is estimated. (See Appendix IA for equations relating escape rate coefficient to per cent release). The WAPD data are not directly applicable to the proposed test; therefore, the calculation of magnitude of release for specific nuclides for the proposed test is based on an arbitrarily assumed 2 per cent release of Xe^{133} at equilibrium.

Because the water flowing through the bore of the test hole will empty to a closed rear face drain and flow to the Pluto crib, the noble gases will be vented to the stack at the collection box and the water soluble fission products will flow to the crib. (See Figures 1 through 8 for the route of fission products

to the stack and crib.)

The following table presents the calculated number of curies of I^{131} , Sr^{89} , Sr^{90} , Cs^{137} , and Ba^{140} accumulated in the crib and the release rate of Xe^{133} and Kr^{85} from the stack as a result of fission products release and decay from the defected UO_2 fuel element while being irradiated.

Nuclide	Mass No.	Rate of release		Conc. in Crib - Curies	
		from stack	$\mu\text{c/sec}$	After 30 days Irrad.	After 60 days Irrad.
Kr	85	54.			
Xe	133	51.			
I	131			18.	18.
Sr	89			0.51	0.84
Sr	90			0.0017	0.0033
Cs	137			0.0015	0.0030
Ba	140			0.27	0.32

(See Appendix IB for mathematical relationships)

It must be recognized that because of the complex nature of the release of fission products from UO_2 by the diffusion mechanism, the analysis presented in Appendix IB is necessarily a simplification of the over-all process. It is felt that the assumptions based on the data available, ^{1,2,3} are conservative and thus the concentrations and release rates from a defected normal fuel element probably will be lower than those calculated. A normal fuel element is defined as one in which there does not occur any center melting or other abnormal phenomenon.

HAZARDS

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The prime hazard involved in 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.^{2,6} 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 occurred during irradiation at a surface heat flux of 682,000 B/hr-sq. ft. The surface heat flux in the proposed test is expected to be less than 200,000 B/hr-sq.ft.

With a defect present in the cladding during reactor shut down, water may enter the rod and fill interconnected voids present in the UO_2 . This phenomena is termed water-logging. The amount of water entering the rod would be a function of the density of the UO_2 and the amount of interconnected porosity. Upon subsequent reactor start up, the trapped water would flash to steam and could 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 a water-logging type of 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 MTR⁷. 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 with a density of 86 per cent theoretical was irradiated in the ETR⁷. This rod failed with fragmentation of the cladding occurring but no particulate matter was released. Normal loop flow water cleaned up the system adequately; no decontamination procedures were necessary.

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

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TEST DETAILS

1. Installation of Temporary Rear Face Piping From Nozzle Outlet to Rear Face Drain

A one-inch OD stainless steel tube will be attached to the outlet nozzle and extended down the rear face in front of the pigtails and across the rear face below the first tube row to the Pluto drain. A 1/4" OD stainless steel tube extending from the one-inch outlet tube to the X-1 level and back to the Pluto drain will permit the continuous monitoring of the effluent water. See Figure 9 for schematic layout.

2. Dummy Fuel Element Assembly

A dummy four-rod-cluster assembly with aluminum rather than UO_2 rods will be irradiated to verify the hydraulic calculations, to check out the instrumentation in the presence of a low radiation background, and to test the remote discharge procedure.

3. Defected Fuel Element Assembly

The assembly consists of four rods maintained in a cluster configuration by means of spacers located at each end. The rods are 11 inches long and have an OD of .563 inch. Two of the rods are clad in Zircaloy-2, two in Zircaloy-4, cladding thickness is .028 inch. The fuel material is natural UO_2 of 95 per cent theoretical density. One of the Zircaloy-2 and one of the Zircaloy-4 rods have a .005 inch diameter hole drilled through the cladding near the mid-point of the rod. The other rods are not defected. See Figure 10.

The four-rod-cluster assembly is mounted in a 12-inch-long aluminum basket, 2.814" OD x 2.318" ID, in order to obtain the desired hydraulic characteristics and to protect the four-rod-cluster during charging and discharging.

4. Test Hole Loading

One basket containing one four-rod-cluster fuel element 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. The loading arrangement will be the same for both the dummy and defected fuel element assembly irradiations. (See Appendix II for charge-discharge procedures).

5. Thermal Hydraulics for Defected Fuel Element Assembly(See Figure 11 and Appendix III)

a) Total heat generation	30 KW
b) Upstream orifice	.360 Dia.
c) Downstream orifice	.397 Dia.
d) Flow through basket	55 GPM
e) Low scram trip	48 GPM
f) High scram trip	47 GPM
g) Cladding surface temperature at low trip	115°C
h) Cladding surface temperature at high trip	116°C
i) Saturation temperature at low trip	190°C
j) Saturation temperature at high trip	210°C
k) UO ₂ surface temperature at equilibrium Oper.	217°C

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The test hole will be fitted with a 10-90 Panellit gauge which will be connected into the scram circuit.

6. Monitoring

The gamma activity from the fission products in the coolant stream will be continuously monitored by two Anton Electronics Company geiger tubes. 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 will be located on the north wall of the X-1 level; the signal from the geiger tubes 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. A tubing line from the outlet nozzle connector will pipe water at a low flow rate through the manifold and back to the Pluto drain. Although radiation levels are not expected to be high, the manifold will be shielded to minimize exposure to personnel. Activity measurements on effluent piping during an irradiation of a defected fuel element in the MTR indicated activity levels of about 250 mr/hr at four inches in air after about a one minute decay.⁵ The volume of water piped to the manifold will be less, and the decay time will be greater than in the MTR test; therefore, the readings at the manifold location for this test should be less than 250 mr/hr at four inches in air.

The manifold consists of two Anton geiger tubes, a flow meter, a metering valve, several shut-off valves, 1/8" OD stainless steel tubing; miscellaneous fittings, and two Hansen, two-way shut-off, quick disconnects. The water samples are taken by plugging a special sample container into the Hansen disconnects and allowing the container to fill with water. The sample container is removed and taken to the laboratory for analysis of the water. The manifold is enclosed in a sheet metal box. A flexible hose connects the bottom of the box to the hot drain pipe which extends from the X-1 to the X-2 level so that in case the manifold springs a leak, the water will flow to the hot drain.

7. Shutdown Time

	Time in Hours	
	Front Elevator	Rear Elevator
Charging dummy and installing piping	1	2
Discharging dummy and charging defected assembly	1	1
Discharging defected assembly	1	1

8. Reactivity Loss

The test holes are presently charged with B metal. A reactivity loss will occur when the assemblies are irradiated. This Production Test authorizes the charging of up to 8 columns of enrichment metal to compensate for the loss caused by the irradiation of the dummy and the defected assembly.

9. Exposure

The dummy assembly will be irradiated for one operating period. The defected assembly will be irradiated for two operating periods. An operating period for the defected assembly test is defined as at least eight continuous days of equilibrium reactor operation.

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10. Data Required

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

11. 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-Theoretically the escape of fission products from the fuel element during shutdown will be negligible, and the radiation level from the rear face connector should drop to a low level. That this probably will happen was demonstrated by a test of a defected UO_2 rod in the MTR by General Electric, Hanford Laboratories Operation. During operation the readings through four inches of air were about 250 ~~mr~~mr/hr immediately after shutdown the level dropped to four or five mr/hr. It is not expected that this test will substantially affect the normal readings on the rear face during shutdown.

12. 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~~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 shutdown and the test discharged.

SUMMARY OF SPECIFIC ACTIONS PERMITTED BY THIS PRODUCTION TEST

1. Installing temporary piping from the outlet nozzle to the Pluto drain as shown in Figure 9.
2. Irradiating one dummy fuel element assembly for one operating period.
3. Irradiating one defected fuel element assembly for two operating periods.
4. Loading up to eight columns of E metal to compensate for the reactivity loss.
5. Installing the gamma monitor and water sample manifold on the X-1 level.
6. Locating temporary instrumentation in the control room for recording the signals from the gamma monitor manifold.

7. Extending temporary electrical wiring from the monitor location to the recorder, if necessary.
8. Using two spare alarms in the control room to announce the occurrence of high gamma activity.

RESPONSIBILITIES

Irradiation Processing Department

A. KE Processing Operation:

1. Charge and discharge the supporting enrichment columns authorized by this Production Test.
2. Maintain safety and production continuity of the reactor.

B. Irradiation Testing:

1. Charge and discharge the dummy and defected fuel element assemblies.
2. Collect the data.
3. Take water samples.
4. Provide instrumentation.
5. Operate the test facility.

C. Pile Physics:

1. Specify the number and location of the supporting enrichment columns authorized by this Production Test.

Hanford Laboratories Operation

A. Ceramic Fuels Development:

1. Provide the defected and dummy fuel element assemblies.
2. Write the final report.

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R K Marshall

Irradiation Testing

R. K. Marshall:bmr

[illegible]

ITEM NO.	SYMBOL	DESCRIPTION	DRAWING, BFP OR CATALOG NO.	QUANTITY	REMARKS
1		CND GROUND PLAN 15'-0" EL.	N-1-22513	1	
2		END CONNECTOR	BRWY CND 000000	1	
3		8"x8" CAST IRON BOX	02 "V 3221	1	
4		CABLE SUPPORT WITH PLATES FOR 2-7/8 CABLES	02 "S 1000	1	
5		CABLE CONNECTOR (FEMALE END)	POWER MODEL VOL CND 0000	1	

Process Area

D. Equipment

Sampling Room

AS-BUILT DRAWING
GENERAL COMPANY
ENGINEERING DEPARTMENT
PROJECT: 100K REACTOR FACILITIES
DESIGN DATE: 10/1/54
CERTIFIED-REV NO. 1

AS BUILT

FOR CONSTRUCTION
APPROVED
DATE: 8-10-54
FOR C.C.

- KEY PLAN
1. N-1-22512 SEC. 1 & 2 DETAILS
 2. N-1-22513 CND GND PLAN 15'-0" EL.
 3. N-1-22514 CND GND PLAN 15'-0" EL.
 4. N-1-22515 CND GND PLAN 15'-0" EL.
 5. N-1-22516 CND GND PLAN 15'-0" EL.
 6. N-1-22517 CND GND PLAN 15'-0" EL.
 7. N-1-22518 CND GND PLAN 15'-0" EL.
 8. N-1-22519 CND GND PLAN 15'-0" EL.
 9. N-1-22520 CND GND PLAN 15'-0" EL.
 10. N-1-22521 CND GND PLAN 15'-0" EL.
 11. N-1-22522 CND GND PLAN 15'-0" EL.
 12. N-1-22523 CND GND PLAN 15'-0" EL.
 13. N-1-22524 CND GND PLAN 15'-0" EL.
 14. N-1-22525 CND GND PLAN 15'-0" EL.
 15. N-1-22526 CND GND PLAN 15'-0" EL.
 16. N-1-22527 CND GND PLAN 15'-0" EL.
 17. N-1-22528 CND GND PLAN 15'-0" EL.
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 19. N-1-22530 CND GND PLAN 15'-0" EL.
 20. N-1-22531 CND GND PLAN 15'-0" EL.
 21. N-1-22532 CND GND PLAN 15'-0" EL.
 22. N-1-22533 CND GND PLAN 15'-0" EL.
 23. N-1-22534 CND GND PLAN 15'-0" EL.
 24. N-1-22535 CND GND PLAN 15'-0" EL.
 25. N-1-22536 CND GND PLAN 15'-0" EL.
 26. N-1-22537 CND GND PLAN 15'-0" EL.
 27. N-1-22538 CND GND PLAN 15'-0" EL.
 28. N-1-22539 CND GND PLAN 15'-0" EL.
  

SECTION A-A

SECTION B-B

SECTION C-C

SECTION D-D

SECTION E-E (THIS DRAWING)

SECTION F-F (THIS DRAWING)

FOR CONTINUATION SEE H/22159

SECTION 'F-F' (THIS DEPT.)

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NOTES

1. FOR GENERAL NOTES SEE H-122340
2. FOR ADDITIONAL "WATER STUDIES SEMI-WORKS"
PIPING ON OUTLET FACE OF 105-RE ONE.
SEE DWG. # H-1-2495-S-RE.

Construction Codes
• Part 07 - Wood
91-a-390-632
91-a-390-634

AS-BUILT DRAWING
ENGINEERING DEPARTMENT
DATE
DATE
CERTIFIED-REV NO 4

AS BUILT

AS-BUILT DRAWING
ENGINEERING CO. 100 W. 10TH ST. NEW YORK, NY 10011
DATE 4/2/82
DATE
DESIGN
DATE
CERTIFIED-REV NO. 3 KW

SUPERSEDES DRAWING OF SAME
NUMBER DATED 4-4-52

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 1/1/1964
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REVISIONS						
SCALE: 1" = 10'		APPROVALS				
DESIGNED BY	DATE 3-10-54	DR	DATE	DATE	DATE	DATE
CHECKED BY	DATE 3-10-54	DR	DATE	DATE	DATE	DATE
REVIEWED BY	DATE 3-10-54	DR	DATE	DATE	DATE	DATE
PUBL. NAME HERKE						
REV. NO.	105K-3-490R					
PROJECT NO.	CR-518					
U. S. ATOMIC ENERGY COMMISSION						
RESEARCH WORK						
GENERAL ELECTRIC						

Figure 3
PROCESS PIPING
OUTLET FACE - PLANS

PROJECT TITLE			
100K REACTOR FACILITIES			
SLUG	DWG.	REVISION	DATE
105K	H-1-22341		

 

Key rooms and areas include:

- CONTROL ROOM**: Located in the upper right section.
- LABORATORY**: Multiple locations, including a large one in the lower right and smaller ones in the lower left and center.
 

[illegible]

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LEGEND

- FLOOR DRAIN (SCREW PLUG) SEE NOTE #5a —●—
 CESSPOOL DRAIN SEE NOTE #5a —□—
 FLOOR DRAIN (OPEN GRATE) —○—
 FLOOR DRAIN (COVERED GRATE) —○—
 CONTAMINATED DRAIN —○—
 POTENTIALLY CONTAMINATED DRAIN —○—
 CLEAN DRAIN —○—
 DRAIN TO PLUTO CRIB —○—
 BASIN DRAIN LINE —○—
 INK SYSTEM DRAIN —○—
 ROD COOLING WATER DRAIN —○—

NONE

AS BUILT

U.S. ATOMIC ENERGY COMMISSION MANFORD WORKS GENERAL ELECTRIC	
Figure 4 LIQUOR PIPING 107th REACTOR FACILITIES	
PROJECT NO. 107th REACTOR FACILITIES	SHEET NO. H-1-20904
REVISIONS NO. DESCRIPTION 1. 7-30-54 2. 7-30-54 3. 7-30-54	
APPROVALS DESIGNED BY CHECKED BY DRAWN BY DATE	

PLAN
SCALE 1/8" = 1'-0"

GENERAL NOTES

- UNLESS OTHERWISE NOTED, TOP OF ALL FLOOR DRAINS TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
- ALL DRAIN LINES TO BE SLOPED 1/8" IN DIRECTION INDICATED.
- UNLESS OTHERWISE NOTED, DRAIN PIPING SHALL BE 1/2" DIA. TO BE INSTALLED IN ACCORDANCE WITH MFG. SPEC. # 4762.
- FOR LOCATION OF PIPE SLEEVES IN THIS AREA, SEE M-1-21036.
- FLOOR DRAINS (AS SPECIFIED) SHALL BE LOCATED AS FOLLOWS:
 a. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 b. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 c. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 d. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
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 l. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 m. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 n. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 o. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 p. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 q. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 r. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 s. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 t. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
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 v. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 w. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 x. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 y. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.
 z. FLOOR DRAIN TO BE LOCATED 1" BELOW HIGH POINT OF FLOOR.

DECLASSIFIED

LEGEND

CONTAMINATED DRAIN	— (1) — (1) —
POTENTIALLY CONTAMINATED DRAIN	— (2) — (2) —
CLEAN DRAIN	— (3) — (3) —
DRAIN TO PLUTO CRIB	— (4) — (4) —
BASIN DRAIN LINE	— (5) — (5) —
INK SYSTEM DRAIN	— (6) — (6) —
ROD COOLING WATER DRAIN	— (7) — (7) —
FLOOR DRAIN (OPEN GRATE)	— (8) — (8) —
FLOOR DRAIN (CLOSED GRATE)	— (9) — (9) —
FLOOR DRAIN (OPEN GRATE) WITH REMOVABLE SAND TRAP	— (10) — (10) —
DECON. DRAIN TO FILTER	— (11) — (11) —
BASIN OVERFLOW DRAIN	— (12) — (12) —

ISOMETRIC SHOWING DRAINAGE PIPING IN AREA OF COMPARTMENT SINKS & ACID RINSE TANKS.

PLAN
 SCALE: 1/8" = 1'-0"
 FOR GENERAL NOTES SEE HW-20904

DETAIL OF DRAIN LINE ENCASED IN
 CORR. GALV. DR. SLEEVE

18" CORR. GALV. IRON
 DRAINAGE PIPE WITH RIVETS
 PROVED INVERT

10" DRAIN LINE

NONE

NO.	DESCRIPTION	DATE	BY
1	DRAINAGE PIPING - PLAN AT EL. 0'-0"	10/1/68	J. J. [Signature]
2	DRAINAGE PIPING - PROPOSED AREA AT EL. 0'-0"	10/1/68	J. J. [Signature]
3	DRAINAGE PIPING - PROPOSED AREA AT EL. 0'-0"	10/1/68	J. J. [Signature]
4	DRAINAGE PIPING - KEY PLAN ON 7' SCALE	10/1/68	J. J. [Signature]

PROJECT COMPLETE (10/1/68) DRAWING FOR AS-BUILT CHECKED BY: J. J. [Signature] DATE: 10/1/68 DESIGNED BY: J. J. [Signature] DATE: 10/1/68 PROJECT NO. 105-X-3037 PROJECT NAME: CH-512-R		APPROVALS BY: J. J. [Signature] DATE: 10/1/68 BY: J. J. [Signature] DATE: 10/1/68	
U. S. ATOMIC ENERGY COMMISSION HANFORD WORKS 8509 ELECTRIC PLANT C DRAINAGE PIPING PLAN AT ELEVATION 0'-0" STORAGE & TRANSFER AREA 100K REACTOR FACILITY			

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SECTION A-A.

DECLASSIFIED

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DETAIL A.
KE COLLECTING BOX.
FOR STRUCTURAL DETAIL
SEE DWG M-1-2321.

DETAIL B.
KW COLLECTING BOX.
FOR STRUCTURAL DETAIL
SEE DWG M-1-25281

DETAIL G.
DRY WELL.
PLAN AT EL 447.0 OF US BLDG
WITH CONDENSATE DISPOSAL
SYSTEM.

PLOT PLAN
SCALE: 1"=30'

PLAN AT EL 436.0 OF 105 BLDG
SUB-BASIN DRAINAGE DISPOSAL
SYSTEM

SECTION E-E
SCALE: 1" = 1'

LEGEND

SW - SANITARY WATER
FW - FILTERED WATER
HW - HOT WATER
PS - PROCESS SEWER
ST - STEAM

CONSTRUCTION CODES
SEE REFERENCED DRAWINGS
CONSTRUCTION CODES
51-07-Subst-211-772
51-09-Subst-211-772
51-08-Subst-211-772
51-06-Subst-211-772
See referenced draw-
ings.

AS BUILT

AS-BUILT DRAWING
ENGINEERING DEPARTMENT
DESIGN DATE
CERTIFIED-REV NO

NOVA

4	PROJECT COMPLETE R-BOMB CHECKED FOR AS BUILT MOVED POST BARRICADE	MADE 6-6-52	BY	DATE
5	CANALITY REAGNAN (N 2782)	MADE 6-4-52	BY	DATE
6	REVISIONS	MADE 6-12-52	BY	DATE
APPROVALS		MADE	BY	DATE
SCALE: 1/8" = 1'		MADE	BY	DATE
DATE: 10-14-52		MADE	BY	DATE
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105-K AREAS
OUTSIDE LINES
DETAILS AND SECTIONS
Figure 7

100K PRODUCTION FACILITIES

1900 CWS. H-1-23207

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- LEGEND**
- ① CONTAMINATED DRAIN.
 - ② POTENTIALLY CONTAMINATED DRAIN
 - ③ CLEAN DRAIN
 - ④ PLUTO DRAIN.
 - ⑤ BASIN DRAIN
 - ⑥ DECONTAMINATION DRAIN
 - ⑦ BASIN OVERFLOW.
 - ⑧ VENT BACK TO STACK.
 - ⑨ ADD COOLING WATER
- C.T.M. — C.T.M. CO.
G.E. — GENERAL ELECTRIC CO.

DETAIL A

SEE DETAIL A

DESIGN FLOW - 1000 GPM WITH AREA SEWER FLOWING AT MAX. DESIGN FLOW. MAX. TEMP. 130°F. SINCE EACH AREA BACK-UP IN COLLECTION BOX NOT TO EXCEED EL. 444.0. TEMP. CHANGE 150° TO 160° IN 1/2 HR.

AVERAGE OVER ALL GRADES = 0.6%

TERRAIN ELEVATION WITH AREA SEAM

SEE DETAIL C

DECLASSIFIED

APPROVED

NOT FOR CONSTRUCTION

REVISIONS		REV.	DATE
1		10/1/54	
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100		10/1/54	

U. S. ATOMIC ENERGY COMMISSION
HANFORD WORKS
GENERAL ELECTRIC

105 K BLDG.
PROCESS
DRAINAGE SYSTEM
Figure 8

1904K
H-1-2322

THIS DRAWING WAS FORMERLY SK-1-20481.

Figure 9
Schematic Arrangement of Piping on the Rear Face

Figure 10
Defected Fuel Element Assembly

DATE 10/1/58

Test hole loading: One basket containing one, 11 inch long, four rod cluster fuel element located at mid-point of active zone. Perforated tubular spacers downstream.

Figure 11

REFERENCES

1. G. R. Horn, Test Proposal - Irradiation of Defected Fuel Elements in Hanford Reactors, 3-25-60, HW-64471 RD
2. J. D. Eichenberg et.al., Effects of Irradiation on Bulk UO₂, WAPD-183, 1957
3. J. M. Lojek and W. T. Lindsay, Jr., The Effect of Oxygenated Water on Clad-and-Defected UO₂ Fuel Specimens, WAPD-PWR-CP-3166, 1957
4. J. L. Bates, Irradiation of Defected UO₂ Swaged Fuel Rod, Preliminary Report, HW-62762
5. Personal Communication, G. R. Horn to R. K. Marshall, 4-27-60
6. J. D. Eichenberg, R. M. Lieberman, F. P. Mrazik, Irradiation of UO₂ Fuel Rods-The X-1-L Experiment, WAPD-208, 1960
7. J. L. Bates, M. K. Millhollen, and W. C. Roake, Irradiation Proposal for a UO₂ Cylinder and Rod in ETR-7-P-33 Facility, HW-58508 RD, 12-15-58

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APPENDIX I

A. Relationship of Fractional Release of Nuclide to Escape Rate Coefficient

Fractional release is defined as the ratio of the number of nuclides of species A anywhere outside the fuel body to the total number of nuclides of species A existing at equilibrium.

Escape rate coefficient is defined as the fractional amount of each nuclide present within the fuel escaping per unit time.

A_0 = number of nuclides of species A anywhere outside the fuel body

A_1 = number of nuclides of species A inside the fuel body

λ = decay constant for A, sec^{-1}

ν = escape rate coefficient, sec^{-1}

F = fractional release of A at equilibrium

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$$\frac{dA_0}{dt} = \nu A_1 - \lambda A_0 \quad \text{at equilibrium; } \frac{dA_0}{dt} = 0$$

$$A_1 = \frac{\lambda}{\nu} A_0$$

$$F = \frac{A_0}{A_1 + A_0}$$

$$F = \frac{A_0}{\frac{\lambda}{\nu} A_0 + A_0} = \frac{\nu}{\lambda + \nu}$$

WAPD, in their development work for the PWR, obtained an escape rate coefficient of $2 \times 10^{-8} \text{ sec}^{-1}$ for Xe^{133} . λ for Xe^{133} is 1.52×10^{-6} .

$$F = \frac{2 \times 10^{-8}}{1.52 \times 10^{-6} + 2 \times 10^{-8}} = \frac{2}{154} = .013$$

For the proposed test, the fractional release of Xe^{133} at equilibrium was arbitrarily assumed to be 2 per cent.

B. Relationship of Rate of Release of Other Nuclides to That of Xe^{133} at Equilibrium

It can be shown that the escape rate of nuclides at equilibrium by diffusion from a cylindrical fuel body is given approximately by the equation:*

$$E \cong 3.12 \times 10^{13} Y \frac{2}{r} \sqrt{\frac{D}{\lambda}} \quad \text{Eq. (1)}$$

E = nuclides escaping/sec. - KW

Y = fission yield

D = diffusion coefficient - cm^2/sec

λ = decay constant - sec^{-1}

r = radius of fuel body - cm

*Mathematics derived by R.E. Baars

For specific nuclides, subscript x and a refer to Xe^{133} and any other nuclide respectively:

$$E_x \cong 3.12 \times 10^{13} Y_x \frac{2}{r} \sqrt{\frac{D}{\lambda_x}} \quad \text{Eq. (2)}$$

$$E_a \cong 3.12 \times 10^{13} Y_a \frac{2}{r} \sqrt{\frac{D}{\lambda_a}} \quad \text{Eq. (3)}$$

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Because the diffusion coefficient is not known for the diffusing nuclides, it is assumed that all diffusing nuclides have the same diffusion coefficient as Xe^{133} .

Dividing eq. 3 by eq. 2 gives:

$$E_a \cong E_x \frac{Y_a}{Y_x} \sqrt{\frac{\lambda_x}{\lambda_a}}$$

At equilibrium $E_x = 3.12 \times 10^{13} F_x Y_x$ where F_x is the fractional release of Xe^{133} , for this test assumed to be .02

$$E_a \cong 3.12 \times 10^{13} F_x Y_a \sqrt{\frac{\lambda_x}{\lambda_a}} \quad \text{or}$$

$$E_a \cong 3.12 \times 10^{13} F_x Y_a \sqrt{\frac{T_a}{T_x}} \quad \text{where } T = \text{half life.}$$

This equation was used to calculate the release rates of nuclides other than Xe^{133} from the defected UO_2 fuel element.

APPENDIX II

Procedures

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A. Charging Dummy Assembly

1. Reduce water pressure.
2. Install temporary rear face piping.
3. Remove test hole rear nozzle cap.
4. Install upstream and downstream orifices.
5. Charge basket containing dummy assembly from rear.
6. Charge 16, 18" long perforated tubular spaces from rear.
7. Install rear nozzle cap.

B. Remote Discharge of Dummy Assembly and Charging of Defected Assembly

1. Reduce water pressure.
2. Raise elevator above test hole.
3. Remove rear nozzle cap by operating hydraulic cylinder in old beta sample room. Observe cap removal from viewer.
4. Remove front nozzle cap.
5. Push assembly and perforated spacers into discharge basin.
6. Replace front nozzle cap.
7. Charge defected assembly from rear.
8. Charge 16, 18" long perforated tubular spaces from rear.
9. Install rear nozzle cap.

C. Discharging Defected Fuel Element Assembly

1. Reduce water pressure.
2. Remove rear nozzle cap and install tip-off.
3. Raise elevator above test hole.
4. Remove front nozzle cap.
5. Push assembly and perforated spacers into discharge basin.
6. Install front nozzle cap.
7. Remove tip-off and install rear nozzle cap.

APPENDIX III

Thermal Hydraulics

A. Heat Generated and Coolant Temperature Rise

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Data:

Diameter of UO₂: .507"
 Length of UO₂: 10.5"
 Length of clad rod: 11.0"
 Diameter of clad rod: .563"
 Density of 100 % dense UO₂: 10.9 gms/cm³
 Density of UO₂ in test: 95 % of theoretical

Neutron flux in surrounding tubes at fuel element location: $3.80 \times 10^{13} \text{ n/cm}^2/\text{sec}$
 Ratio of volume average flux in UO₂ to flux in surrounding tubes: 1.8
 Heat generated by one gram of U²³⁵ at $1 \times 10^{13} \text{ n/cm}^2/\text{sec}$: .484 KW

Calculations:

Power/ft-rod

$$\frac{\pi}{4} (.507)^2 (12)(16.4)(10.9)(.95) \left(\frac{238}{270} \right) (.0071)(3.8)(1.8)(.484) = 8.5 \text{ KW/ft-rod}$$

Total tube power (one four-rod-cluster)

$$(8.5)(4) \left(\frac{10.5}{12} \right) = 30 \text{ KW}$$

Surface heat flux

$$\frac{(8.5)(3413)(144)}{\pi (.563)(12)} = 197,000 \frac{\text{B}}{\text{hrft}^2}$$

Coolant temperature rise

$$\frac{(30)(3413)}{(55)(498)(1)(1.8)} = 2^\circ\text{C}$$

B. Cladding Surface Temperature For Worst Case-Low Scram Trip

Data:

Basket ID: 2.318"
 Fuel element OD: .563"
 Water temperature: 150° C
 Flow: 48 GPM (From Figure 11)
 Water pressure: 175 psi (From Figure 11)
 Saturation temperature: 190° C
 Surface heat flux: 197,000 B/hr ft²

Calculations:

Correlation used: $h = K_t \frac{V}{De^{.2}}$ from: Fuel Element Tech. Manual, Fig. III-2
 HW 40,000. Based on Dittus-Boelter
 Equation: $\frac{hDe}{K} = .023(DeG)^{.8} \left(\frac{\mu_c}{\mu} \right)^{.4}$

h in B/hrft² F

V in ft/sec

D_e in ft

K_t at 15°C = 165

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

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

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

$$\Delta T = \frac{197000}{(1090)(1.8)} = 100^\circ \text{C}$$

Surface temperature of cladding: $100 + 15 = 115^\circ \text{C}$

C. UO₂ Surface Temperature

Data:

Bulk water temperature: 15°C

Water film ΔT : 100°C

Zircaloy thermal conductivity: 11.8 B/hrft °F

Cladding-UO₂ interface heat transfer coefficient: 2000 B/hrft²°F

Power/ft-rod: 8.5 KW

Calculations:

$$\text{Cladding } \Delta T: \Delta T = \frac{(8.5)(3413) \ln \frac{.563}{.507}}{2\pi(11.8)} = 41^\circ \text{C}$$

$$\text{Cladding-UO}_2 \text{ Interface } \Delta T: \Delta T = \frac{(8.5)(3413)(144)}{\pi(.507)(12)(2000)(1.8)} = 61^\circ \text{C}$$

UO₂ Surface Temperature

$$15 + 100 + 41 + 61 = 217^\circ \text{C}$$

D. Flow Split

Data:

Test hole tube ID: 2.875"

Basket OD: 2.814"

Basket ID: 2.318"

Basket length: 12.0"

Fuel element OD: 0.563"

Fuel element length: 11.5"

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Calculations

Equations (from McAdams 3rd edition pages 155 and 158)

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

f = friction factor

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

L = Length

ρ = Density

g = Gravitational acceleration. $\Delta P, G, \mu, \rho, L, A, r, \mu$

r = Hydraulic radius

N = Reynolds No.

μ = viscosity

W = Mass flow rate

A = Area

$$\Delta P_1 = \Delta P_2$$

$$\mu_1 = \mu_2$$

$$\rho_1 = \rho_2$$

constants cancel

$$\frac{(G_1)^{1.8}}{(G_2)^{1.8}} = \frac{L_2}{L_1} \left(\frac{r_1}{r_2} \right)^{1.2} \quad \text{Substituting } G = \frac{W}{A} \text{ and simplifying}$$

$$\frac{W_1}{W_2} = \frac{A_1}{A_2} \left(\frac{L_2}{L_1} \right)^{.555} \left(\frac{r_1}{r_2} \right)^{.667}$$

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

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$$\left(\frac{L_2}{L_1}\right)^{.555} = \left(\frac{12}{11.5}\right)^{.555} = 1.02$$

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

$$\frac{W_1}{W_2} = (11.1)(1.02)(5.75) = 65.2$$

DECLASSIFIED

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

$$\frac{F_1}{F_2} = 65.2$$

at equilibrium $F_1 + F_2 = 56$

$$F_2 = 56 - 65.2 F_2$$

$$66.2 F_2 = 56$$

$$F_2 = .85 \text{ GPM}$$

The assumption is made that the flow through the basket is 55 GPM, the flow around 1 GPM.

DATE

FILMED

8/3 /94

END