DOCUMENT NO.

BNWL-CC-2445

ISSUING FILE

COPY NO. 1-

DATE

February 10, 1970

PROJECT

BATTELLE-NORTHWEST

RICHLAND, WASHINGTON

TITLE

Proposal to Irradiate Intentionally Defected Mixed-Oxide Fuel Rods
In the ETR M-3 Loop Facility - The 23 kW/ft Test Series -
 (BNW-5-1 through -6)

AUTHOR

C. R. Hann, T. B. Burley, and M. D. Freshley

DISTRIBUTION

NAME

BUILDING AREA

NAME

BUILDING AREA

AL Bement	306	300		
JD Berger	Idaho Falls, Idaho, NRTS			
DW Brite	308	300		
TB Burley	309	300		
DG Carter	309	300		
JA Christensen	309	300		
FG Dawson	R0	3000		
DR de Halas	324	300		
MP Freshley (2)	309	300		
S. Goldsmith	PS-II	3000		
CR Hann	309	300		
EN Heck	327	300		
RS Hope	Idaho Falls, Idaho, NRTS			

GR Horn	309	300
JW Kutcher	R0	3000
LA Lawrence	309	300
CE Leach	308	300
DC Lehfeldt	308	300
DG Long	309	300
JL Maryott	309	300
FE Panisko	309	300
LC Schmid	R0	3000
JK Soldat	3705	300
RG Wheeler	306	300
AEC: Attn: WE Fry	Federal Bldg.	
HA House	Federal Bldg.	
Central Files	3760	300

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

FAYR

NOTICE

This document copy, since it is transmitted in advance of patent clearance, is made available in confidence solely for use in performance of work under contracts with the U. S. Atomic Energy Commission. This document is not to be published nor its contents otherwise disseminated or used for purposes other than specified above before patent approval for such release or use has been secured, upon request, from the Chief, Chicago Patent Group, U. S. Atomic Energy Commission, 6030 S. Cass Ave., Argonne, Illinois.

(CLASSIFICATION)

TO BE USED ON UNCLASSIFIED AND OFFICIAL USE ONLY DOCUMENTS

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	<u>Page No.</u>
I. Purpose	1
II. Program Descriptions	2
III. Irradiation Unit Charges	6
IV. Summary Test Data	6
A. Fabrication Data	6
B. Thermal & Hydraulic Characteristics of the Reactor and Loop Facility	9
C. Thermal Characteristics of the Test Assembly	9
V. Enrichment	11
VI. Defect Testing Assembly Description	12
A. Fuel Rod Fabrication	12
B. Fuel Element Description	14
C. Assembly Description	17
VII. Basin Manipulations	19
VIII. Irradiation Conditions	25
IX. Data Requirements	26
X. Special Considerations	26
A. Defect Testing Experience Summary	26
B. Fission Gas Release When Defecting Irradiated Fuel Rods	31
C. Fission Gas Release From Defected Fuel Rods During Irradiation	32
D. Fission Gas Release From Defected Fuel Rods During Charge-Discharge	33
E. Fuel Release From Defected Fuel Rods During Irradiation	33
F. Fuel Release From Nonirradiated $\text{UO}_2\text{-PuO}_2$ Fuel Rods During Handling	35
G. Contamination Control From Irradiated Mixed-Oxide Fuel	36

TABLE OF CONTENTS (CON'T)

	<u>Page No.</u>
XI. References	41
XII. Acknowledgements	42
XIII. Approvals	42
XIV. Appendix A: Power Generation and Heat Flux	43
A. Power Generation	43
B. Weight/inch of Major Test Components	43
C. Gamma Heating	45
D. Surface Heat Flux	45
XV. Appendix B: Fluid Flow & Coolant Temperature	52
A. Fluid Flow	52
B. Coolant Temperatures	53
C. Hot Spot and Hot Channel Factor Analysis	55
XVI. Appendix C: Fuel Temperature Calculations	58
A. Calculated Fuel Temperatures	58
B. Calculated Cladding & Fuel Temperatures with Hot Spot Factor, F_h	59
XVII. Appendix D: Defect Test Element Drawings	67

PROPOSAL TO IRRADIATE INTENTIONALLY
DEFECTED MIXED-OXIDE FUEL RODS
IN THE ETR M-3 LOOP FACILITY
- THE 23 kW/ft TEST SERIES -
(BNW-5-1 through -6)

C. R. Hann, T. B. Burley, and M. D. Freshley

I. Purpose

The purpose of the proposed tests is to evaluate the defect behavior of Zircaloy-clad mixed-oxide pellet-containing fuels and possibly establish fuel performance limits. The results are directly related to the objectives and needs of the Plutonium Utilization Program (PUP).

As commercial power reactor fuel operating conditions approach inherent performance limits, defect behavior becomes increasingly important and may ultimately prove to be the limiting performance factor. Because reactors cannot operate economically without accommodating defective fuel rods, satisfactory defect behavior must be demonstrated under high performance conditions before widespread acceptance of mixed-oxide fuels can be expected. It is therefore important to determine the behavior characteristics and operating limits for defective fuel and, in particular, the effects of specific power, burnup, and fuel form on defect fuel behavior.

While limited information on the defect behavior of mixed-oxide fuels is available, considerably more tests conducted under controlled conditions are required before behavior under any specific operating conditions can be predicted with assurance. The ETR M-3 Loop Facility is capable of accommodating irradiation tests of defected and

nondefected mixed-oxide fuels operating under high performance conditions. Defect tests similar to those proposed have been conducted in the ETR P-7 loop⁽¹⁻⁴⁾ and the Fuel Element Rupture Test Facility (FERTF) in PRTR⁽⁵⁻⁷⁾.

II. Program Description:

The defect testing program includes an investigation of three important parameters: 1) fuel design (small and large diametral gap solid pellet, dished pellet, and cored pellet), 2) power generation (23 to 27 kW/ft), and 3) burnup. Mixed-oxide pellet fuel rod designs include those which are of most current interest for the commercial application of plutonium recycle in thermal power reactors.

The irradiation performance of defected and non-defected rods incorporating several different mixed-oxide pellet fuel designs will be evaluated. The pellet designs of interest include different ways of providing the internal void space necessary for accommodating thermal expansion and fuel swelling, i.e., different pellet-clad diametral gaps, pellet dishing, and pellet coring. To be investigated are the possible effects of providing large and small diametral gaps, the possible effects of providing a pellet dish volume of three percent, and the possible effects of utilizing a cored pellet design with a core volume of ten percent.

It is planned to perform two defect test series at nominal peak linear rod power generations of 23 and 27 kW/ft. This proposal pertains to the 23 kW/ft test series only. Testing at about 23 kW/ft is expected to produce maximum fuel temperatures of approximately 2600°C in pellet-containing rods whereas, the tests planned at 27 kW/ft are expected to produce moderate fuel melting. Most of the tests will be performed at about 23 kW/ft because these conditions are more relevant to the maximum conditions projected for light water reactors now being

committed and are considered to be in a critical surface heat flux region ($500,000 \text{ Btu/hr-ft}^2$) for defect fuel performance⁽⁷⁾.

The performance data to be derived include rod dimensional changes, fuel washout, waterlogging, fission gas and fission product release, fuel rod swelling, stoichiometry changes and their effect on fuel performance.

The defect testing assembly consists of nineteen 0.565-inch diameter rods similar to the PRTR fuel element design (Figure 1). Twelve fueled Zircaloy-clad rods, three of which are removeable, are located in the outer twelve-rod ring of the element. A stainless steel-clad Al-Co flux wire assembly is located in the center support rod and the six-rod ring is composed of dummy rods to balance the flow distribution within the element. A stainless steel sleeve surrounds the element to protect the pressure tube from possible damage during irradiation.

Different pellet fuel rod designs will be irradiated in the twelve-rod ring of the assembly and one of the replaceable rods will be defected with a 0.015-inch diameter hole for each defect experiment (Figure 1). Pre-defect irradiation of rods incorporating the different fuel designs will be conducted in the replaceable positions of the element. A 0.015-inch diameter defect hole is judged to be most representative of an initial in-service defect and yet not conducive to waterlogging of the rod during repeated reactor startups.

The planned rod movements among the three replaceable rod positions in the element for the six tests proposed are depicted in Figure 2. Each test represents approximately one ETR cycle. A defected rod will not be irradiated during Test No. 1; however, a defected rod will be irradiated during each of the remaining five tests. A vertical arrow on a rod indicates removal of that particular

ROD TYPE:

- 1 - SOLID (0.006" DIAMETRAL GAP)
- 2 - SOLID (0.010" DIAMETRAL GAP)
- 3 - DISHED (3% VOID VOLUME WITH 0.006" DIAMETRAL GAP)
- 4 - DISHED (3% VOID VOLUME WITH 0.010" DIAMETRAL GAP)
- 5 - CORED (10% VOID VOLUME WITH 0.006" DIAMETRAL GAP)

Figure 1 - Arrangement of the Different Fuel Rod Types
in the Defect Testing Assembly

FUEL TYPE

- 1 - SOLID (0.006" DIAMETRAL GAP)
- 2 - SOLID (0.010" DIAMETRAL GAP)
- 3 - DISHED 3% VOID VOLUME (0.006" DIAMETRAL GAP)
- 4 - DISHED 3% VOID VOLUME (0.010" DIAMETRAL GAP)
- 5 - CORED 10% VOID VOLUME (0.006" DIAMETRAL GAP)

* EACH TEST REPRESENTS APPROXIMATELY ONE ETR CYCLE. A DEFECTED ROD WILL NOT BE IRRADIATED DURING TEST NO. 1. A VERTICAL ARROW INDICATES REMOVAL OF THE ROD FROM THE TEST FOR POST-IRRADIATION EXAMINATION.

Figure 2 - Disposition of Individual Replaceable Rods in the Defect Testing Assembly During the 23 kW/ft Test Series

rod for detailed postirradiation examination. With the exception of Test No. 2, it is planned that each rod to be defected will be removed from the element for one irradiation period to allow decay of fission product activity prior to being defected for irradiation. A total of nine defected and nondefected rods will be removed from the replaceable positions in the element for detailed destructive examination during the conduct of the 23 kW/ft test series. At the completion of the planned test series, the irradiation performance of the nine nonreplaceable fuel rods in the element will also be evaluated.

III. Irradiation Unit Charges

Irradiation unit charges for this test series should be charged to Budget Activity Number 04-01-21-01. The activities described in this proposal are included in the Plutonium Utilization Program 189 No. 12001, Sub-Task A-3.

IV. Summary Test Data

A. Fabrication Data

Basket:

Material	304L
Length	46.75 in.
O.D.	3.317 in.
I.D.	3.255 in.

Center Support Pin:

Material	Zr-2
Length	35.5 in.
O.D.	0.565 in.
I.D.	0.176 in.

Flux Monitor:

Tube Material	304 SS
Length	40 in.
O.D.	0.167 in.
I.D.	0.117 in.
Flux Monitor Wire	Al-0.1 wt% Co

Dummy Rods:

Material	Zr-2
Length	34.9 in.
O.D.	0.565 in.

Fuel Rod Cladding:

Material	Zr-2
Length	31.45 in.
O.D.	0.565 in.
I.D.	0.5045 in.

Fuel:

Depleted UO_2 Insulator Pellets:

Length	0.400 in.
O.D.	0.495 in.
Weight, each	12.5 gm.
Weight, per rod	25.0 gm.
Weight, per element	260 gm.
Enrichment	0.22% ^{235}U

$\text{UO}_2\text{-PuO}_2$ Pellet Fuel:

Rod Type	1	2	3	4	5
Pellet Type	Solid	Solid	Dished	Dished	Cored
O.D., in.	0.498	0.494	0.498	0.494	0.498
Column length, in.	26.8	26.8	26.8	26.8	26.8
Diametral Gap	0.006	0.010	0.006	0.010	0.006
Density, gm/cm^3	10.28	10.28	10.28	10.28	10.28
Density, % TD	93.5	93.5	93.5	93.5	93.5
% UO_2	96	96	96	96	95.6
% ^{235}U	0.71	0.71	0.71	0.71	0.71
% PuO_2	4.0	4.0	4.0	4.0	4.4
% $^{239+241}\text{Pu}$	89.40	89.40	89.40	89.40	89.40
UO_2 per rod, gms*	836.7	822.9	811.7	798.2	750.1
^{235}U per rod, gms*	5.24	5.15	5.08	5.00	4.69
PuO_2 per rod, gms*	34.87	34.29	33.82	33.26	34.52
$^{239+241}\text{Pu}$ per rod, gms*	27.49	27.03	26.67	26.22	27.21
Pu per rod, gms*	30.75	30.24	29.83	29.33	30.44

* Calculated weights.

B. Thermal and Hydraulic Characteristics of the Reactor and Loop Facility

Reactor Power Level	175 MW
Irradiation Facility	M-3 Loop
Loop Flow Rate (Rated Max)	166 gpm
Loop Flow Rate (Desired)	150 gpm
Loop Pressure	2000 psi
Coolant pH	9.8 to 10.2
Coolant Oxygen Concentration	<0.05 ppm
Saturation Temperature of Loop	635°F
Coolant Velocity Over the Fuel	11.85 ft/sec.
Heat Transfer Film Coefficient	5169 Btu/hr-ft ² -°F
Inlet Water Temperature	480°F
Outlet Water Temperature	502°F
with hot channel factor	511°F
Maximum Cladding Surface Temperature	603°F
with hot spot factor	640°F
Pressure Drop Across Assembly (Measured)	4.4-5.5 psi

C. Thermal Characteristics of the Test Assembly

Peak Linear Rod Power Generation	~23 kW/ft
Average Linear Rod Power Generation	~14 kW/ft
Peak Surface Heat Flux	546,000 Btu/hr-ft ²
Average Surface Heat Flux	329,000 Btu/hr-ft ²
Total Power Generation	410 kW
Heat Transfer Film Coefficient	5169 Btu/hr-ft ² -°F
Maximum Cladding Surface Temperature	603°F
with hot spot factor	640°F
Maximum Cladding Temperature	770°F
with hot spot factor	816°F
Maximum Fuel Surface Temperature	1165°F
with hot spot factor and $h_{\text{gap}} = 1000$	1260°F

Maximum Fuel Temperature:

Rod types 1 and 3 (Fig. 1)

 $h_g = 2000 @ 23 \text{ kW/ft}$ 4555°F

with hot spot factor and

 $h_{gap} = 1500 @ 22.6 \text{ kW/ft}$ 4665°F

Rod types 2 and 4 (Fig. 1)

 $h_g = 1500 @ 23 \text{ kW/ft}$ 4700°F

with hot spot factor and

 $h_{gap} = 1000 @ 22.6 \text{ kW/ft}$ 4900°F

Rod type 5 (Fig. 1)

 $h_g = 2000 @ 23 \text{ kW/ft}$ 3800°F

with hot spot factor and

 $h_{gap} = 1500 @ 23.3 \text{ kW/ft}$ 3860°F

V. Enrichment

The pellet fuel material used in the rods for the 23 kW/ft defect test series is a homogeneous mixture of natural UO_2 and PuO_2 . The PuO_2 concentration was adjusted for the different fuel designs to produce the desired power generation in the flux available in the ETR M-3 Loop facility. The same plutonium concentration (4 wt%) was used for Rod Types 1 through 4 (Figure 1) because the design differences do not result in significant differences in the fuel smear density hence, the power generation will be essentially the same. However, the ten percent reduction in fuel smear density for the cored pellet design (Rod Type 5, Figure 1) was compensated for by a corresponding increase in the plutonium concentration in order to make the power generation for the different rod designs comparable.

Uranium from batch UN-39 and plutonium from batch EU-24-10-7-103 were used for the fabrication of all the pellet fuel. The isotopic composition of the plutonium is as follows:

^{238}Pu - 0.0305 percent

^{239}Pu - 88.08 percent

^{240}Pu - 10.46 percent

^{241}Pu - 1.317 percent

^{242}Pu - 0.117 percent

Depleted UO_2 (0.22 percent ^{235}U) insulator pellets are located at each end of the fuel column. These pellets were fabricated by standard UO_2 cold-press-sinter techniques..

VI. Defect Testing Assembly Description

The defect testing assembly is designed to fulfill the functional requirements of the test program. This includes underwater manipulations involving element disassembly/assembly operations, fuel rod interchanges, and the irradiation testing of intentionally defected mixed-oxide fuel rods in the ETR M-3 Loop facility under the desired conditions in a safe manner.

Basically, the defect testing assembly consists of nineteen 0.565-inch diameter rods similar to the PRTR fuel element design. Twelve fueled Zircaloy-clad rods, three of which can be replaced remotely, are located in the outer twelve-rod ring of the element. Different pellet fuel rod designs will be irradiated in the twelve-rod ring of the assembly and one of the replaceable rods will be defected for each defect experiment. A flux wire assembly is located in the center support pin and the six-rod ring is composed of Zircaloy dummy rods to balance the flow distribution within the element. A stainless steel sleeve, attached by a quick-disconnect assembly, surrounds the element to protect the pressure tube from possible damage during irradiation. The design details are illustrated by the following drawings which are included in Appendix D: H-3-29631, H-3-29632, H-3-29636 Sht. 1, H-3-29636 Sht. 2, and H-3-29641. A detailed description of the major components of the assembly which include fuel rod fabrication, fuel element design, and the assembly design is presented as follows:

A. Fuel Rod Fabrication

The pellet fuel rods represent designs which are of most current interest for the commercial application of plutonium recycle. The designs include different ways of incorporating the internal void volume necessary to accommodate thermal expansion and fuel swelling, i.e., different

pellet-clad diametral gaps, pellet dishing, and pellet coring. Detailed mixed-oxide pellet designs are shown in H-3-29632. The mixed-oxide pellet fuel was fabricated utilizing standard cold-press-sinter techniques and employing thorough quality control and quality assurance procedures.

Well characterized and certified ceramic grade natural UO_2 was ball-milled 24 hours to permit attaining the desired green density and uniform sinterability. The PuO_2 was obtained by calcining plutonium oxalate in air at 425°C and recalcining at 900°C to eliminate volatile impurities. A homogeneous mixture of the UO_2 and -325 mesh PuO_2 powders was obtained by a layering and raking technique followed by mixing in a V-blender for 15 minutes. Two percent Carbowax binder was added to the powder mixture in the form of a 16 wt% solution of Carbowax and water. After drying each 4 Kg batch at 70°C for 10 hours, the powder was granulated through a 20 mesh screen and 0.25 wt% Sterotex was added as a die lubricant. The granulated powder was pressed into pellets at a pressure of 35 tsi to yield a green pellet density of 60-61% TD. Sintering was conducted in an 8% H_2 -92% A gas mixture at 1650°C for 10 hours to yield a sintered density of $93.5 \pm 1.5\%$ TD. The pellets were then centerless ground to final diameter, cleaned, and inspected prior to loading. The different pellet designs are shown in H-3-29632.

The depleted UO_2 insulator pellets at each end of the fuel column were fabricated by standard UO_2 , cold-press-sinter techniques.

The cladding used for the fuel rods is standard pre-tested PRTR reactor grade seamless Zircaloy-2 tubing. The cladding tubes for the defect testing element are 31.446 inches long with a 0.404/0.405 inch I.D. and a nominal

0.030 inch wall thickness (H-3-29636 Sht. 1). After non-destructive testing, inspection, I.D. air gauging, and cleaning, the bottom end plugs (H-3-29636 Sht. 2) were welded in place in an inert atmosphere welding box. Fuel loading consisted of inserting the bottom depleted UO_2 pellet, the mixed-oxide fuel column, the top depleted UO_2 pellet, and the Inconel-X spring into the cladding tube. The spring holds the fuel column in place and prevents collapse of the cladding in the gas plenum region under loop operating pressure. The depleted UO_2 insulator pellets at each end of the fuel column prevent excessive temperatures in the bottom end plug region and at the fuel/spring interface.

After loading, the top end plugs (H-3-29636 Sht. 2) are welded in place in a helium-filled chamber and the welds are radiographed to assess weld integrity. The rods are then etched, helium leak tested, and autoclaved for 18 to 24 hours in 400°C steam at a pressure of 700 to 1000 psi. This is followed by a thorough final inspection, length measurements, profilometer scanning of the O.D., and gamma absorbtometer scanning of each rod.

Alternate rods in the twelve-rod ring are wrapped clockwise and counterclockwise with an 0.072 inch diameter autoclaved Zircaloy-2 wire on a 9.10-inch pitch to insure proper rod spacing and adequate coolant mixing within the assembly. The wires are attached to the top and bottom end plugs by spot fusion welding. The three replaceable rods in the assembly are not wire wrapped.

B. Fuel Element Description

The fuel element is a 19-rod assembly composed of a top and bottom end bracket, center support pin, six solid Zircaloy dummy rods, twelve fueled rods, flux monitor, and circumferential bands.

Top End Bracket (H-3-29636, Sheet 2)

The function of the top end bracket is to support the rods in a 19-rod array as illustrated in the cross-sectional view of the array in drawing H-3-29636, Sheet 1, align the fuel element within the coolant channel of the basket, and to provide for lifting the fuel element assembly. Attachment to the bracket to facilitate lifting is accomplished by means of the two through pins in the bracket stem. The pins, as well as the total assembly, are fabricated from Zircaloy-2. The pins in the stem are aligned with the major gusset on the bracket. Replaceable rod positions are located 45 degrees counterclockwise (viewed from top) from the major gusset, thence every 120 degrees. The first removable rod position displaced 45 degrees counterclockwise from the major gusset is identified as the defect position. Intentionally defected fuel rods will be located in this position for defect testing.

Bottom End Bracket (H-3-29636, Sheet 2)

The function of the bottom end bracket is to support the fuel assembly, align the bottom ends of the fuel rods, and align the fuel element within the coolant channel of the basket. The fuel rods are not attached to the bottom bracket but are allowed axial freedom of movement by slipping in the guide holes, thus accommodating axial thermal expansion.

Center Support Pin (H-3-29636, Sheet 2)

The function of the center support pin is to separate the top and bottom end brackets. The fuel bundle is supported by the bottom bracket through the center pin and top bracket which, in turn, supports the fuel rods. The center pin is threaded at both ends with a nut attached to secure the top and bottom end brackets. The center pin has a hole bored the full length to accommodate the flux monitor.

Dummy Rods (H-3-29636, Sheet 1)

Non-fueled, solid Zircaloy-2 dummy rods were placed in the six-rod ring in order to balance the flow distribution within the element and to achieve the desired linear rod power generation in the test rods. This was considered preferable to increasing the plutonium concentration in the test rods to undesirably high values, i.e., greater than 4 wt% PuO_2 . The external dimensions of the dummy rods are the same as the standard rods. The six dummy rods were subjected to different surface treatments in order to study the effect of these treatments on the in-reactor corrosion behavior of Zircaloy-2 in a coolant system contaminated with fissionable material. Alternate dummy rods in the ring are wrapped clockwise and counter-clockwise with an 0.072 inch diameter spacing wire.

Fuel Rods (H-3-29636, Sheet 1)

The twelve fuel rods in the test assembly are described in Section A: Fuel Rod Fabrication.

Flux Monitor (H-3-29636, Sheet 2)

The flux monitor assembly consists of two Al-0.1 wt% Co wires enclosed in a stainless steel tube. The flux monitor assembly is located in the center of the center support pin which positions the assembly in the center of the element. The monitor passes through a hole in the top end bracket and rests in a recess in the bottom end bracket.

Circumferential Bands (H-3-29636, Sheet 1)

Zircaloy-2 circumferential strip bands are drawn around the 19-rod bundle assembly to secure the rods in position and prevent vibration. The 3/16" wide by 0.015 inch thick bands are installed remotely with a specially designed tool which secures the bands in place under tension

with a crimping buckle. The bands are located at two positions on the fuel element where they will not interfere with the spiral rod wire wraps. This banding technique was standard practice on PRTR fuel elements.

C. Assembly Description

The major components of the defect testing assembly in addition to those described above are the quick-disconnect assembly, the basket, the coupling, and the quick release pin.

Quick-Disconnect Assembly (H-3-29641)

The function of the quick-disconnect assembly is to provide the lifting ring for removal of the assembly from the M-3 Loop and engage the protective basket. The quick-disconnect assembly is welded onto the coupling during assembly with the correct orientation with respect to the major gusset and defected fuel rod. The spring loaded pins in the quick-disconnect assembly engage the basket by being forced outward by the cam action which is activated by turning the double lead threaded nut. The nut may be rotated until it is stopped by engaging the head or the body without disengaging the threads. Full counterclockwise rotation of the double lead nut disengages the pins. The proper position of the quick-disconnect assembly within the basket is established by the lips on the lifting yoke and the slots on the upper end of the basket. That is, the fuel element and quick-disconnect mechanism may be inserted into the basket and rotated until the lips engage the slots on the basket and the assembly will drop approximately 1/4" into the basket. The quick-disconnect nut may then be rotated clockwise until a significant increase in torque is encountered. The quick-disconnect will then be engaged with the basket.

A 1/4" diameter hole, 1/8" deep is located on the top of the lifting yoke to facilitate proper orientation of the assembly in the M-3 Loop. The position of the intentionally defected fuel rod is in line with this mark. When engaging the fuel element lifting pins with the coupling on the bottom of the quick-disconnect mechanism, it is possible to misalign the defect position by 180 degrees. Verification of proper installation may be made by assuring that the major gusset is located 45 degrees clockwise from the mark on the lifting yoke. If not properly oriented, the major gusset will be located 135 degrees counterclockwise from the mark. Under these circumstances, the major gusset will not be visible when viewed from above and a gusset will be located 115 degrees clockwise from the mark indicating incorrect installation.

Basket (H-3-29631)

The purpose of the basket is to provide mechanical and coolant isolation between the fuel element assembly and reactor pressure tube. The basket is aligned within the pressure tube by the spacer pins which are electron beam welded to the outer diameter of the basket. The basket does not provide support for the fuel element assembly other than lateral support. The fuel element assembly bottom end bracket supports the hydrodynamic load and weight of the fuel element assembly and quick-disconnect mechanism. The bottom end bracket is supported on the pedestal assembly at the bottom of the basket which in turn rests upon the shelf at the neck of the pressure tube. When the defect testing assembly is lifted from the pressure tube, the fuel element and quick-disconnect assembly are first lifted approximately 1/8" before engagement of the basket by the three pins on the quick-disconnect. An orifice is attached to the bottom of the basket that was properly sized by hydraulic testing to provide a bypass coolant flow of 20 gpm at 150 gpm total flow in the annular space between the reactor pressure tube and basket outer diameter.

Coupling (H-3-29631)

The coupling is the latching mechanism that engages the lifting pins on the top end bracket of the fuel element assembly to the quick-disconnect assembly. The coupling is engaged by rotating while applying a light downward force until the coupling grooves are engaged by the lifting pins. The fuel element is held stationary during this operation. The coupling will then slip onto the top end bracket stem about two inches until it bottoms out against the top lifting pin. It is then rotated 90 degrees and lifted approximately 1/4".

Quick Release Pin (H-3-29631)

The quick release pin prevents inadvertent release of the coupling from the fuel element assembly. It is inserted after full engagement of the coupling as described above. The pin contains two detent balls which prevent inadvertent release of the pin. During insertion, the balls will retract when slight pressure is applied to the surface of the pin. Removal is accomplished by the same combination of forces except with slight outward force. The pin is sufficiently long that it cannot become disengaged within the basket.

VII. Basin Manipulations

The defect testing assembly is designed to facilitate underwater manipulations involving disassembly/assembly operations, fuel rod interchanges, and defecting which are essential to the conduct of the testing program in the ETR M-3 Loop facility.

The element will be sent to the ETR completely assembled and ready for insertion into the M-3 Loop facility. Test BNW-5-1 will be a one cycle, non-defect irradiation test. After irradiation of BNW-5-1, the

the element will be removed from the loop, disassembled, rods exchanged, a rod defected, reassembled, and reinserted into the loop for the next scheduled irradiation test. The sequence of manipulations will be basically the same for subsequent tests except that they will be performed with defected rods.

The steps involved in the underwater manipulations with the defect testing assembly are envisioned to be as follows:

- ° Removal of the element from the M-3 Loop following a defect test will be accomplished in the normal manner. A cooling period of 24 to 48 hours may be advisable to avoid off-gassing activity during discharge. Once the element is located in the basin work area, the temporary exhaust system will be utilized to vent any off-gas from the defected rod.
- ° The assembly will then be placed in the vertical holder on the work platform where the quick-disconnect is loosened.
- ° The fuel element will then be lifted from the basket and supported in the vertical position on the work platform. The quick release pin will be withdrawn and the quick-disconnect assembly removed from the element at the coupling. Remove the flux monitor assembly.
- ° A special handling tool will be attached to the top end bracket of the element and the element will be placed in a horizontal position on the support brackets on the work platform.
- ° The circumferential strip bands will be removed from the element and the prescribed rod exchanges made. A special rod handling tool is provided to facilitate these manipulations.

- ° The element is then rebanded with the specially designed underwater banding tool (Figure 3). (This operation has been successfully performed numerous times on PRTR elements and the satisfactory performance of the remotely installed bands has been amply demonstrated in PRTR.)
- ° Defecting of the rods with a 0.015 inch diameter hole is performed with the specially designed defecting tool (Figure 4). The depth of penetration of the drill is adjustable and rods will generally be defecting in the peak axial flux position.

As the cladding of pre-irradiated rods is penetrated with the drill, a gas bubble will emanate from the hole (Figure 5). This gas will be directed, by means of the underwater funnel, through a flexible tygon tube into a one quart collection bottle. The tygon tubing passes through a ball valve that is attached to the collection bottle. After the gas has been collected, the tubing is withdrawn from the bottle and the ball valve is closed. The bottle containing the collected fission gas is then removed from the defecting tool for storage and ultimate disposal.

- ° The element is then raised into the vertical position and a new flux monitor assembly is placed in the center rod position.
- ° The quick-disconnect assembly is then attached to the element and the quick-release pin reinserted.
- ° The element is then reinserted into the basket and the quick-disconnect assembly secured.

Figure 3 - Underwater Banding Tool

Figure 4 - Photograph Showing the Head of the Underwater Defecting Tool

Figure 5 - Underwater Fuel Rod Defecting Arrangement

The element can now be stored under the temporary gas exhaust system or charged into the loop for the subsequent irradiation test.

VIII. Irradiation Conditions

The operating conditions for these experiments, i.e., coolant pressure, coolant velocity, and power generation were chosen to approximate as nearly as possible the expected operating conditions in advanced commercial power reactors and still fulfill the operating requirements of the M-3 Loop facility.

It is important that the defect testing assembly is always positioned in the M-3 Loop facility with the same orientation relative to the center of the reactor core. An orientation hole drilled in the lifting yoke is in line with the defect rod position in the element. This fiducial mark, which is visible on the top of the element, should always be pointed toward the center of the reactor core so that the defect position is always located in the highest flux region within the loop.

Operate the loop at a pressure of 2000 psi and an inlet water temperature of 480°F at a total flow of 150 gpm. Do not exceed a total loop power output of 410 KW.

Maintain the coolant pH between 9.8 and 10.2 with ammonia control. Proper pH control should be verified by extracting a coolant sample at least once per day during operation. The oxygen level in the coolant should be maintained at less than 0.05 ppm.

To avoid potential waterlogging and excessive activity release from defected fuel rods, the reactor power during the initial startup and subsequent startups from zero power should be increased at not more than 10 MW per 5 min. up to a power level of 50 MW. Above 50 MW, the power level increases should be at a rate not to exceed 10 MW per 10 min.

IX. Data Requirements

The following data should be obtained during each experiment:

1. A plot of reactor power level versus time.
2. A plot of loop power versus time.
3. Equivalent full power days of operation.
4. Accumulated burnup.
5. Vertical flux profile.
6. A plot of loop activity versus time.

Coolant samples will be extracted periodically during defect testing depending upon the reactor operating mode, i.e., startup, steady-state or shutdown, and analyzed for uranium, plutonium, noble gas isotopes, and iodine isotopes in order to evaluate the behavior of the particular defect under study.

X. Special Considerations

A. Defect Testing Experience Summary

The postulated catastrophic defect behavior of ceramic fuels, and in particular vibrationally compacted fuels, has not materialized⁽⁵⁻⁷⁾. In general, the defect behavior of oxide fuels operating at heat ratings currently employed in power reactors has been satisfactory. These results have been corroborated by in-service fuel defect experience and intentionally defected experiments which included single rod tests, prototype elements, and full-size PRTR fuel elements tested over a wide range of conditions⁽⁵⁾.

Intentionally defected UO_2 pellet and vipac fuels have been irradiated under conditions which produce fuel melting or near melting in the ETR P-7 loop⁽¹⁻⁴⁾, the Fuel Element Rupture Test Facility (FERTF) in PRTR⁽⁵⁾, and in the X-6 loop of the NRX Reactors in Canada⁽⁷⁾. An intentionally defected Zircaloy-clad element containing Al-Pu alloy fuel was successfully irradiated in the ETR P-7 Loop⁽¹⁾.

In addition, defective Zircaloy-clad oxide fuels have been irradiated under PWR conditions in Shippingport, Westinghouse MTR, NPD, Saxton, and CVTR⁽⁷⁾. Defected and defective fuels were irradiated under BWR conditions in VBWR, Dresden, Big Rock Point, and the Halden BWR⁽⁷⁾.

The experience that is most pertinent to the tests included in this proposal are the intentional defect tests conducted in the FERTF in PRTR⁽⁵⁾, the recent ETR P-7 Loop test⁽⁴⁾, and PRTR in-service defects⁽⁶⁾. The results of these experiments are summarized as follows:

1. Intentionally Defected FERTF Experiments

Irradiation testing of deliberately defected fuel rods was performed in the FERTF in PRTR using full-size PRTR fuel elements^(5,6). The types of defects studied have included 0.060-in. diameter holes and slits up to 6.5-in. long. Peak rod power generations in pre-irradiated and non-irradiated intentionally defected experiments were in the range of 6.1 to 27 kW/ft with a considerable portion of the fuel molten at the higher powers. A summary of the packed particle element defect tests performed in the FERTF is presented in Table I.

Release characteristics for the different types of defects studied were determined by analyzing the ratios of nuclides released to the coolant. Generally, the defect behavior of ceramic packed particle elements has been excellent. No appreciable fuel washout, fuel rod deformation, waterlogging effects, or excessive hydriding of the Zircaloy cladding was observed in any of the packed particle fuel element defect tests. The only exception was the rupture of a deliberately defected $\text{UO}_2\text{-PuO}_2$ rod operating with considerable fuel melting at the plane of the defect (Test I, Table I)⁽⁸⁾. There was evidence of fuel oxidation and an unexpected large amount of fuel melting in the

Table 1

COOLANT ACTIVITIES FROM DEFECTED PRTR ELEMENTS IRRADIATED IN THE FERTF

Test No.	Element Type	Cladding Defect	Maximum Rod Power, (kW/ft)	Loop Activity, (cpm)			Type of Release
				Steady State GM ($\beta\gamma$)	DN (Delayed Neutron)	Burst Activity GM ($\beta\gamma$)	
A	Al-Pu FE-5008	6 in. Cracks	~7.0	15,000	13,000	-----	Recoil ^(a)
B	Swaged UO ₂ FE-1069	0.06 in. diam Hole	7.3	8,000	4,000	220,000	Equilibrium ^(b)
C	Swaged UO ₂ FE-1039	0.62 in. long Longitudinal Slit	7.0	~23,000 ^(c)	~13,000 ^(c)	700,000	-----
D	Swaged UO ₂ FE-1039	3.2 in. long Longitudinal Slit	7.0	23,000	13,000	100,000	Diffusion ^(d)
E	Swaged UO ₂ FE-1030	6.5 in. long Longitudinal Slit	6.1	68,000	38,000	150,000	Diffusion
F	Vibrationally Compacted UO ₂ FE-1067	3.0 in. long Longitudinal Slit	7.3	47,000	26,000	170,000	Diffusion
G	Vibrationally Compacted UO ₂ FE-1067	2.9 in. long Longitudinal Slit	7.3	60,000	31,000	-----	Diffusion
H	Vibrationally Compacted UO ₂ - 2 wt% PuO ₂ FE-6004	0.06 in. diam Hole	24	80,000	4,000	660,000	Equilibrium
I	Vibrationally Compacted UO ₂ - 4 wt% PuO ₂ FE-6504	0.06 in. dia Hole	27	7.7 x 10 ⁶ ^(c)	1.6 x 10 ⁵ ^(c)	-----	(Ruptured)

(a) Good communication between fuel and coolant - zero delay time.

(b) Poor communication between fuel and coolant - long delay time.

(c) Test was discontinued before stable form of release was established.

(d) Good communication between fuel and coolant - short delay time.

ruptured rod which may have been the underlying cause of failure. Excessive melting could have been caused by a decrease in the effective thermal conductivity of the fuel, a decrease in the melting temperature, a higher than expected power generation due to venting of the fission gas poisons through the defect, or a combination of these effects.

2. Intentionally Defected ETR Experiment

A defect test of a prototype element in ETR provides a demonstration of the satisfactory defect behavior of vibrationally compacted fuel under extreme operating conditions⁽⁴⁾. In this experiment one rod of a seven-rod cluster containing vibrationally compacted UO_2 fuel was defected with a 0.060-inch diameter hole. The defected fuel rod operated at 27 kW/ft with as much as 60% of the fuel radius molten with very little fuel washout and no clad distortion occurring. This defect test was irradiated for nine effective full power days during which time it was subjected to three programmed reactor shutdowns and startups and one scram recovery startup. During the scram recovery, power was increased from 0 to 50% of full power in less than 3 minutes. In spite of this sudden power increase there was no excessive fuel washout. The release of gaseous fission products to the coolant sharply increased during the scram recovery but soon returned to the release rate normally obtained during steady-state operation.

3. PRTR In-Service Defects

Thirty-seven in-service defects of packed particle oxide fuel rods have occurred in PRTR without causing serious operating problems⁽⁶⁾. These include one swage compacted UO_2 rod out of 1254 irradiated, 20 swage compacted UO_2 - PuO_2 rods out of 1824 irradiated, and 16 vibrationally compacted UO_2 - PuO_2 rods out of 2242 irradiated. It is important to realize that all but two of the defects are attributed to impurities in the fuel material.

The defects attributed to impurities in the fuel material occurred over a relatively short period of time and were caused by 1) fluoride contamination in the plutonium, 2) sorbed moisture on the oxide fuel particles, and 3) traces of hydrocarbons in the form of oil inadvertently introduced by the failure of mechanical processing equipment. These fuel rod defects characteristically occurred at low burnups (approximately 300 MWd/MTM) because of severe internal hydriding and embrittlement of the cladding by a gas phase hydriding mechanism. A predominance of the PRTR defects occurred in the end plug crevice region. Laboratory investigations have shown that this crevice region is particularly susceptible to gas phase hydriding attack. Gas phase hydriding type defects have been experienced by other investigators in pellet-containing fuel rods.

One of the two defects not attributed to fuel impurities occurred in a swage compacted UO_2 rod and was attributed to cladding damage during swaging. The cause of the other defect in a swage compacted $\text{UO}_2\text{-PuO}_2$ rod after a peak burnup of 5800 MWd/MTM has not been determined. This defect was characterized by corrosion on the inside surface of the Zircaloy cladding and a pin-hole penetration through the cladding.

Although severe localized embrittlement and loss of cladding fragments occurred in some instances, little or no fuel loss into the coolant resulted and no severe reactor operating difficulties were experienced. Fuel washout or erosion resistance is improved by in-reactor sintering of packed particle fuel which occurs during the early stages of irradiation at the higher fuel operating temperatures. Waterlogging has not been observed in fuel elements that either failed during normal reactor operation or were initially intentionally defected. In many instances, the release of fission product activity was first detected during postshutdown depressurization.

B. Fission Gas Release When Defecting Irradiated Fuel Rods

When mixed-oxide pellet fuel is irradiated at specific powers sufficient to produce central fuel temperatures near melting, about 25 percent of the noble gas generated is released from the fuel material. When an irradiated rod is purposely defected under water, a major fraction of this noble gas may be released from the fuel rod. If the noble gas from short cooled rods is allowed to rise to the surface of the water, high beta-gamma dose rates may result.

A 25 percent noble gas release from a fuel rod 26.8 inches long that has been irradiated at a peak power of 23 kW/ft for 4 weeks is estimated to be 900 curies after a decay time of 8 hours, 700 curies after 1 day, and 20 curies after 4 weeks. These amounts of noble gas in a bubble would produce gamma dose rates in air at a distance of three feet of 100 R/hr, 45 R/hr, and 0.5 R/hr, respectively. The beta-gamma dose rates (primarily beta) from these amounts of noble gas inside a cloud of volume 1000 ft³ would be 10,000 Rad/hr, 8,000 Rad/hr, and 160 Rad/hr.

Radioactive iodine may also be released during the defecting of irradiated fuel rods. Provided the fuel is cold (70 to 100°F) only a small fraction of the radioactive iodine in a fuel rod is believed to be released in the bubbles of noble fission gases when a fuel rod is defected. One percent of the iodine-131 in a fuel rod 26.8 inches long that has been irradiated at a peak power of 23 kW/ft for 4 weeks is estimated to be 11 curies after a decay time of one day, and 0.1 curies after a decay time of 4 weeks. The allowable body burden of I-131 in the thyroid gland is 7×10^{-7} curies.

To minimize personnel exposure from the release of radioactive noble gas and iodine, an underwater hood will be utilized to collect the released gas when the fuel rod

is purposely defected. The collected fission gases will be transferred to an underwater collection bottle for storage to permit decay of the noble gas and iodine activity before the gas is released. It is estimated that a three-month decay period will reduce the noble gas beta dose rate by a factor of 400 and the iodine - 131 activity by a factor of 2000.

C. Fission Gas Release From Defected Fuel Rods During Irradiation

A fraction of the radioactive fission product isotopes that are produced in a defected fuel rod will be released to the coolant during irradiation. These radioactive isotopes, especially the noble gases and iodines, that are in the coolant can create high gamma dose rates in the loop equipment areas and air contamination during sampling of the coolant.

Defect testing of UO_2 and $\text{UO}_2\text{-PuO}_2$ fuels in the FERTF loop in PRTR and the X-2 loop of the NRX Reactor in Canada has provided information on the release characteristics of the fission product isotopes from defected fuel rods to the coolant during irradiation. During steady state power irradiation of defected rods, the concentrations of noble gases and iodine in the coolant will generally reach a steady state value. Provided the loop water is not degassed, the amount of long half life Xe-133 and I-131 steady state activity in the coolant system is approximately five percent of the amount of these isotopes generated in a fuel rod operating at centerline fuel temperatures near melting. This amounts to approximately 130 curies of Xe-133 and 55 curies of I-131.

For the shorter half life fission gas isotopes, the amount of activity in the coolant is less than one percent of the activity produced in a fuel rod. If the

coolant is passed through a cleanup ion exchange system, a significant part of the iodine activity will be deposited on the resins. The noble gas isotopes will generally stay in the coolant.

During reactor startups, reactor power changes, large coolant temperature or pressure changes, and reactor shutdowns, bursts of fission gas activity may be released that will increase the net coolant activity by a factor of 10 to 1000 above the steady state activity. These bursts of activity are mostly short half life isotopes and will decay in a fairly short period of time.

D. Fission Gas Release From Defected Fuel Rods During Charge-Discharge

When defected fuel is discharged from or charged into the reactor, fission gas will generally not be released provided the fuel is kept cold (70-100°F). If the fuel is allowed to heat up, fission gas including radioactive iodine will be released from defected or failed fuel.

E. Fuel Release From Defected Fuel Rods During Irradiation

In the event of extensive splitting or loss of cladding during defect testing, it possible to release irradiated fuel to the coolant. A loss of cladding during irradiation can be identified by a large sudden increase in the steady state gamma or delayed neutron activity in the coolant. This increase in activity is a result of the direct fission product recoil-release to the coolant from the exposed fuel.

The release of a large amount of fuel from a rod can also be identified by an increase in the gamma dose rate where the irradiated fuel would tend to settle out in the loop or by an unexplained decrease in power generation or reactivity.

If a fuel rod is defected by drilling a small hole in the cladding, loss of fuel from the rod does not occur unless gross failure of the cladding takes place. Under certain conditions gross cladding failure can be caused by a) steam pressure buildup inside the fuel rod on reactor startups caused by rapid heatup of water that entered the fuel rod when the fuel was cold (waterlogging), b) overheating the cladding from localized flow blockages caused by fuel swelling, rod bowing or foreign objects becoming lodged in the fuel element, and c) rapid corrosion and massive hydriding of the zirconium cladding as a result of localized overheating. Several intentional defect tests have been performed with Zircaloy-clad pellet and vibrationally compacted UO_2 and $\text{UO}_2\text{-PuO}_2$ fuel rods defected with holes ranging from 0.013 to 0.060 in. in diameter and irradiated at specific powers sufficient to produce fuel melting. There has been no evidence of failures caused by waterlogging, localized overheating, or accelerated corrosion. For cladding defects between a size sufficiently large to let water enter the fuel rod and about 5 mils, waterlogging may occur, but experimental confirmation is lacking.

The rupture of a defected $\text{UO}_2\text{-4 wt% PuO}_2$ fuel rod during irradiation in the FERTF loop in PRTR with considerable fuel melting resulted in the expulsion of about 700 grams of fuel from the rod⁽⁸⁾. Of the fuel that was released, an estimated 300 grams remained in the loop piping and heat exchangers, 200 grams was removed by the loop cleanup ion exchange system (100 percent cleanup flow) and 200 grams were released outside of the loop through a hole in the pressure tube that occurred at the time of the fuel failure.

The FERTF loop piping and equipment was successfully decontaminated by a) using the OPG solution to remove the $\text{UO}_2\text{-4 wt% PuO}_2$ fuel material, (The OPG solution dissolves the UO_2 in mechanically mixed $\text{UO}_2\text{-PuO}_2$ fuels leaving the finely

divided PuO_2 particles in a solution that flushed from the system. OPG dissolves solid solution $\text{UO}_2\text{-PuO}_2$ completely.); b) using an alkaline permanganate solution to condition the film on the stainless piping; and c) using an inhibited citrate-oxalate solution to remove the oxide film on the piping⁽⁹⁾. During the decontamination any leakage of the decontamination solutions was immediately cleaned up to prevent the spread of plutonium contamination.

The fission product and plutonium contamination that was released through the hole in the pressure tube was deposited throughout the reactor complex making cleanup difficult. After the iodine-131 activity in the contamination decayed, air and bioassay sampling results indicated that the beta-gamma and alpha activity remained in particulate form and did not create an airborne contamination problem.

The FERTF fuel rupture would have been relatively innocuous had the pressure tube not failed. In the proposed defect tests in ETR, the fuel element is surrounded by a basket assembly to protect the pressure tube.

F. Fuel Release From Nonirradiated $\text{UO}_2\text{-PuO}_2$ Fuel Rods During Handling

Nonirradiated $\text{UO}_2\text{-PuO}_2$ fuel material may be released as a result of accidental rod damage during handling. This potential can be minimized by a) handling fuel with care to prevent damage, b) routinely surveying unirradiated plutonium containing fuels and personnel for alpha contamination, c) wearing protective clothing and assault masks into an area where unirradiated mixed-oxide fuel has been released from a rod, and d) cleaning up spilt fuel as soon as possible to minimize the spread of contamination.

G. Contamination Control From Irradiated Mixed-Oxide Fuel

Contamination control considerations with irradiated plutonium have been studied at PNL in connection with the postirradiation examination of plutonium-containing fuels⁽¹⁰⁾. Control of alpha contamination by means of controlling the associated beta contamination is practical if the composition and concentration of the plutonium in the fuel and the irradiation history of the fuel under consideration are known. A re-evaluation of the concentrations to which beta activity must be controlled to assure that the combined dose from the beta and associated alpha emitters are within permissible limits was performed.

The computer program ISO-GEN was used to calculate the radionuclide inventory in a defect test fuel rod at the end of each of eight steps which occur in a sequence similar to what is anticipated during the conduct of the proposed defect test series. The steps are as follows:

- Step 1 - Thirty-four (34) days irradiation at 40 watts/fuel rod (1.36 MWd total); flux of 5.95×10^{12} nv; flux ratio of 0.2.
- Step 2 - Twenty-eight (28) days decay following discharge after Step 1.
- Step 3 - Thirty-four (34) days additional irradiation at conditions similar to Step 1 except the flux is 6.16×10^{12} nv.
- Steps 4, 5, 6, 7, and 8 correspond to 10, 25, 50, 150, and 300 days decay time following Step 3.

From the calculated radionuclide composition and assuming no fractionation upon release from the fuel, an MPCa for the mixture was calculated by the methods described in Reference 11. The MPCa mixture is the total airborne

concentration which corresponds to the maximum permissible dose to the "critical" organ for which it is calculated. For the radionuclide mix in the subject fuel, bone was deemed to be the critical organ.

The results of the analysis are summarized in Table II and Figure 6. The percentage contribution of alpha emitters to both the radiation dose and the activity present are indicated in the Table. As indicated, even though about 2/3 of the dose is contributed by the alpha emitters, effective alpha control can be exercised by limiting the concentration of beta emitters in air to a value which corresponds to their share of the MPCa mixture. The principal alpha emitters contributing to the radiation dose to bone are ^{238}Pu , ^{239}Pu , and ^{240}Pu which have an MPCa of $2 \times 10^{-12} \mu\text{Ci/cc}$. ^{241}Pu with an MPCa of $9 \times 10^{-11} \mu\text{Ci/cc}$ contributes the majority of the beta radiation dose.

Similar considerations could be applied to surface contamination provided no fractionation occurred. If the degree of fractionation were known, it could be taken into account and a new MPCa mixture could be calculated from the fractionated composition.

PAGES 38 to 39
WERE INTENTIONALLY
LEFT BLANK

Table II

MPCa*For the Mixture of Radionuclides Present in
Mixed-Oxide Fuel and the Relative Contributions of
 α and β Emitters (Bone as Critical Organ)

<u>Step</u>	<u>40-Hr. MPCa of Mixture (μCi/cc)</u>	<u>Corresponding Concentration of</u>		<u>% Contribution of Alpha Emitters to</u>	
		<u>Beta Emitters</u>	<u>Alpha Emitters</u>	<u>Total Activity</u>	<u>Radiation Dose to Bone</u>
1	7.1×10^{-9}	7.1×10^{-9}	1.4×10^{-12}	0.020	69
2	1.1×10^{-9}	1.1×10^{-9}	1.3×10^{-12}	0.12	69
3	7.4×10^{-9}	7.4×10^{-9}	1.5×10^{-12}	0.020	67
4	2.7×10^{-9}	2.7×10^{-9}	1.5×10^{-12}	0.055	67
5	1.6×10^{-9}	1.6×10^{-9}	1.5×10^{-12}	0.095	67
6	9.9×10^{-10}	9.9×10^{-10}	1.5×10^{-12}	0.15	67
7	3.9×10^{-10}	3.9×10^{-10}	1.5×10^{-12}	0.38	68
8	1.9×10^{-10}	1.9×10^{-10}	1.5×10^{-12}	0.80	69

* MPCa = maximum permissible concentration in air, i.e., that concentration which if inhaled continuously during a 40-hour work week would result in the maximum permissible dose to the bone (30 rem).

Figure 6 - Forty-hour MPC_a for the Mixture of Radionuclides Present in a UO₂-4 wt% PuO₂ Defect Fuel Rod Versus Decay Time Subsequent to the Second Discharge or Step 3. (Bone as Critical Organ).

XI. References

1. M. D. Freshley and F. E. Young, "Proposal for Rupturing an Al-Pu Seven-Rod Cluster (GEH-11-5) in the ETR 3x3P7 Facility," HW-67366, Nov. 8, 1960.
2. L. A. Pember, "Proposal to Irradiate a Defect Test, GEH-12-38," BNWC-52, March 13, 1965.
3. F.E. Panisko and L. A. Pember, "Proposal to Irradiate a Defect Test, GEH-12-39," BNWL-CC-433, Nov. 18, 1965.
4. M. D. Freshley, "Proposal to Irradiate an Intentionally Defected UO_2 Element in the ETR P-7 Loop: GEH-12-40," BNWL-CC-1181, April 24, 1967.
5. M. D. Freshley and F. E. Panisko, "The Irradiation Behavior of UO_2 - PuO_2 Fuels in PRTR," BNWL-366, March 1967.
6. M. D. Freshley and T. B. Burley, "Vibrationally Compacted Ceramic Fuels," BNWL-SA-2412, February 1969.
7. D. H. Locke, "Defected Zircaloy Fuel," Nuclear Engineering International, August 1969, pp. 648-652.
8. M. D. Freshley, R. G. Wheeler, J. M. Batch, and G. M. Hesson, "Investigation of the Combined Failure of a Pressure Tube and Defected Fuel Rod in PRTR," BNW-272, May 1966.
9. J. A. Ayres, L. D. Perrigo, and R. D. Weed, "Decontamination of a PWR," Nucleonics, April 1967.
10. L. G. Faust, J. G. Bradley, and C. L. Band, "Criteria for Safe Handling of Irradiated Plutonium in Beta-Gamma Cells," ANS Proceedings of the 11th Hot Lab. Conference, Nov. 1963, p. 437.
11. International Commission on Radiological Protection, Report of Committee No. II, Permissible Dose for Internal Radiation, ICRP Publication 2, Pergammon Press, 1959.

XII. Acknowledgements

The authors wish to acknowledge the assistance rendered by the following individuals in the preparation of this proposal:

C. E. Leach	-	Mechanical Design Thermal Hydraulic Testing
J. W. Kutcher	-	Physics Calculations
J. L. Maryott	-	Evaluation of Activity Releases Associated With Defect Testing
J. K. Soldat	-	Evaluation of Contamination Control From Irradiated Mixed- Oxide Fuel

XIII. Approvals

Proposed by:

M.D. Freshley, Tech Ldr.
PUP Fuels Evaluations

Date: 2/11/70

Recommended by:

J.A. Christensen, Mgr.
Fuels Research Section

Date: 2/12/70

Fuels & Materials Dept.
Approval for Transmittal by:

A.L. Bement, Jr., Mgr.
Fuels & Materials Dept.

Date: 2/12/70

XIV. Appendix A: Power Generation and Heat Flux

A. Power Generation

Power generation calculations for the defect testing assembly with twelve fueled rods were performed by Reactor Experiment Section personnel using the THERMOS code⁽¹⁾. The results of these calculations were normalized to experimental data obtained during the irradiation of GEH-10-101-148 during ETR Cycle 85.

Measured fluxes from the Cycle 85 experiment were used to construct a map of the unperturbed flux in the M-3 Loop. This map was then superimposed on the THERMOS calculation for the Cycle 85 to provide an estimate of rod power generations. The peaking factors used were calculated from the flux map and the measured flux values. The tube power calculated in this manner was within 10% of the measured powers.

The standard annular ring geometry was used for the cell calculations. This model and the flux shape for a 19-rod element containing 12 UO_2 -4 wt% PuO_2 fuel rods are shown in Figure 1A.

The calculated maximum rod power at the peak position versus PuO_2 enrichment is shown in Figure 4A. The power generations indicated are believed to be conservative.

Calculated flux distributions in single rods are shown in Figures 2A and 3A.

B. Weight/inch of Major Test Components

1. Fuel

Assume: same density for UO_2 and PuO_2
and 94% TD pellets. Nominal ID of cladding
- 0.505 inches.

a. Solid Pellets - 0.006 in. diametral gap

$$\frac{\pi}{4} (.499)^2 \text{ in}^2 \times 10.96 \text{ gm/cm}^3 \times 0.94 \times 16.39 \text{ cm}^3/\text{in}^3 \\ = \underline{33.1 \text{ gm/in fuel}}$$

similarly

- b. Solid Pellets - 0.010 in. diametral gap
32.6 gm/in fuel
- c. Dished Pellets - 0.006 in. diametral gap
and 3 vol% dish. - 32.1 gm/in fuel
- d. Dished Pellets - 0.010 in. diametral gap
and 3 vol% dish - 31.6 gm/in fuel
- e. Cored Pellets - 0.006 in. diametral gap
and 10 vol% void - 29.8 gm/in fuel

2. Cladding (Zr-2)

$$\frac{\pi}{4} (0.565^2 - 0.505^2) \text{ in.}^2 \times 6.55 \text{ gm/cm}^3 \times 16.39 \text{ cm}^3/\text{in}^3$$

$$= \underline{5.4 \text{ gm/in (cladding)}}$$

3. Dummy Rods (Zr-2)

$$\frac{\pi}{4} (0.565)^2 \text{ in.}^2 \times 6.55 \text{ gm/cm}^{-3} \times 16.39 \text{ cm}^3/\text{in}^3$$

$$= \underline{26.95 \text{ gm/in (Dummy Rod)}}$$

4. Basket (304SS)

ID = 3.255 in.

OD = 3.317 in.

$$\frac{\pi}{4} (3.317^2 - 3.255^2) \text{ in.}^2 \times 7.92 \text{ gm/cm}^{-3} \times 16.39 \text{ cm}^3/\text{in}^3$$

$$= \underline{40.8 \text{ gm/in (Basket)}}$$

5. Pressure Tube (Zr-2)

ID = 3.441 in.

OD = 4.391 in.

$$\frac{\pi}{4} (4.391^2 - 3.441^2) \text{ in.}^2 \times 6.55 \text{ gm/cm}^3 \times 16.39 \text{ cm}^3/\text{in}^3$$

$$= \underline{628.0 \text{ gm/in (Pressure Tube)}}$$

C. Gamma Heating

Calculations are based on an average gamma heating value of one watt per gram.

$$1. \text{ Fuel (use 12 rods with maximum gm/in fuel)} \\ 33.1 \text{ gm/in} \times 0.001 \text{ Kw/gm} \times 26.8 \text{ in/rod} \times 12 \text{ rod} \\ = \underline{10.65 \text{ Kw.}}$$

2. Cladding

$$5.4 \text{ gm/in} \times 0.001 \text{ Kw/gm} \times 26.8 \text{ in/rod} \times 12 \text{ rod} \\ = \underline{1.74 \text{ Kw.}}$$

3. Dummy Rods (includes center pin)

$$26.95 \text{ gm/in} \times 0.001 \text{ Kw/gm} \times 26.8 \text{ in/rod} \times 7 \text{ rod} \\ = \underline{5.05 \text{ Kw.}}$$

4. Basket

$$40.8 \text{ gm/in} \times 0.001 \text{ Kw/gm} \times 46.75 \text{ in} = \underline{1.91 \text{ Kw.}}$$

5. Pressure Tube

$$628 \text{ gm/in} \times 0.001 \text{ Kw/gm} \times 46.75 \text{ in} = \underline{29.4 \text{ Kw.}}$$

D. Surface Heat Flux1. Surface area of cladding per inch of cladding

$$\pi(.565 \text{ in}) \times 1 \text{ in.} \times \frac{1 \text{ ft}^2}{144 \text{ in.}^2} = 1.232 \times 10^{-2} \text{ ft}^2/\text{in}$$

2. $(Q/A)_{\text{max}} = (\text{fission heat fuel} + \text{gamma heat fuel} + \text{gamma heat cladding}) \div (\text{surface area cladding})$

$$(Q/A)_{\text{ave}} = \frac{(Q/A)_{\text{max}}}{(P_A)(P_C)}$$

$$P_A = \text{axial peaking factor} = 1.32^{(*)}$$

$$P_C = \text{circumferential peaking factor} = 1.26^{(*)}$$

* Personal communication-JW Kutcher (Jan. '70).

3. Solid or dished fuel (use maximum power rods as base).

From Figure 4A rod power at 4 wt% PuO₂

$$= \underline{22.6 \text{ Kw/ft}}$$

$$= \underline{1.885 \text{ Kw/in}}$$

$$(Q/A)_{\text{max}} = (1.885 \text{ Kw/in} + 0.033 \text{ Kw/in} + 0.005 \text{ Kw/in})$$

$$3413 \text{ Btu/hr-ft}^2\text{-Kw} / 1.232 \times 10^{-2} \text{ ft}^2/\text{in.}$$

$$= \underline{533,000 \text{ Btu/hr-ft}^2}$$

$$(Q/A)_{\text{ave}} = \frac{533,000 \text{ Btu/hr-ft}^2}{(1.32)(1.26)} = 321,000 \text{ Btu/hr-ft}^2$$

4. Cored Fuel

From Figure 4A and the single rod flux distribution calculations (Figures 2A & 3A),

$$\text{Rod Power at 4.4 wt\% PuO}_2 = \underline{23.3 \text{ Kw/ft}}$$

$$= \underline{1.940 \text{ Kw/in}}$$

$$(Q/A)_{\text{max}} = (1.940 \text{ Kw/in} + 0.0298 \text{ Kw/in} + 0.005 \text{ Kw/in})$$

$$3413 \text{ Btu/hr-ft}^2\text{-Kw} / 1.232 \times 10^{-2} \text{ ft}^2/\text{in}^{-1}$$

$$= \underline{546,000 \text{ Btu/hr-ft}^2}$$

$$(Q/A)_{\text{ave}} = \frac{546,000 \text{ Btu/hr-ft}^2}{(1.32)(1.26)} = \underline{329,000 \text{ Btu/hr-ft}^2}$$

Figure 1A - Results of THERMOS Calculations for the Defect Testing Assembly in the ETR M-3 Loop Facility

Figure 2A - Results of THERMOS Flux Depression Calculations for Solid Pellet Fuel Rods

Figure 3A - Results of THERMOS Flux Depression Calculations for the UO_2 -4.4 wt% PuO_2 Cored Pellet Fuel Rods

Figure 4A - Rod Power Versus Enrichment

Appendix A References

1. Letter from J.W. Kutcher to M.D. Freshley,
dated 12/22/69, "Physics Calculations on
M-3 Loop in ETR."

XV. Appendix B: Fluid Flow and Coolant Temperatures

A. Fluid Flow

A mockup of the test assembly was subjected to hydraulic testing in the thermal-hydraulic facilities at PNL. The dimensions of the test assembly were within the tolerance levels of the as-built drawings. The purpose of the hydraulic tests was to 1) verify pressure drops across the assembly and 2) determine the proper size of the flow orifices in order to provide at least 20 gpm flow in the annulus between the basket and the reactor pressure tube at conditions of 480°F and 150 gpm total flow. The results of the tests⁽¹⁾ indicate that with an orifice sized to provide a minimum of 20 gpm flow through the annulus at 480°F and 150 gpm, the pressure drop will range between 4.44 and 5.54 psi depending upon the final physical dimensions of test assembly⁽²⁾.

1. Fluid Flow

$$\begin{aligned}\text{Flow area} &= \frac{\pi}{4} [3.255^2 - 19(0.565)^2 - 12(0.072)^2] \\ &= \underline{3.51 \text{ in.}^2} \\ &= \underline{0.0244 \text{ ft.}^2}\end{aligned}$$

$$\begin{aligned}\text{Wetted Perimeter} &= \pi [3.255 + 19(0.565) + 12(0.072)] \\ &= \underline{46.7 \text{ in.}}\end{aligned}$$

$$\begin{aligned}\text{Equivalent Diameter} = D_e &= \\ &= \frac{4 \times \text{area of cross-section}}{\text{wetted perimeter}}\end{aligned}$$

$$\begin{aligned}&= \frac{4(3.51)}{46.7} = \underline{.301 \text{ in.}} \\ &= \underline{.0251 \text{ ft}}\end{aligned}$$

2. Coolant Velocity

$$\begin{aligned}V &= \text{flow rate (ft}^3/\text{sec)} \div \text{flow area (ft}^2\text{)} \\ &= \frac{(130 \text{ gal/min})}{(7.48 \text{ gal/ft}^3)(60 \text{ sec/min})(0.0244 \text{ ft}^2)} \\ &= \underline{11.85 \text{ ft/sec}}\end{aligned}$$

B. Coolant Temperatures

1. Heat Released to Coolant

- a. Assume fuel loading of BNW-5-1 for maximum fuel inventory, i.e., 10 solid rods and 2 cored rods.

- b. $(Q_T) = Q_1 + Q_2 + \text{gamma heat from dummy rods, basket, and pressure tube}$

Q_1 = heat from solid and dished rods

Q_2 = heat from cored rods

Q_1 or $Q_2 = (Q/A)_{\text{ave}} \times \text{total surface of rod} \times \text{number of fuel rods}$

$$\begin{aligned} Q_1 &= (321,000 \text{ Btu/hr-ft}^2)(1.232 \times 10^{-2} \text{ ft}^2/\text{in}) \\ &\quad (26.8 \text{ in})(10 \text{ rods}) \\ &= \underline{1.06 \times 10^6 \text{ Btu/hr}} \end{aligned}$$

$$\begin{aligned} Q_2 &= (329,000 \text{ Btu/hr-ft}^2)(1.232 \times 10^{-2} \text{ ft}^2/\text{in}) \\ &\quad (26.8 \text{ in})(2 \text{ rods}) \\ &= \underline{.217 \times 10^6 \text{ Btu/hr}} \end{aligned}$$

$$\begin{aligned} \text{gamma heat} &= 5.05 \text{ Kw} + 1.91 \text{ Kw} + 29.4 \text{ Kw} \\ &\quad 3413 \text{ Btu/hr-ft}^2 - \text{Kw} \\ &= \underline{.124 \times 10^6 \text{ Btu/hr}} \end{aligned}$$

$$\begin{aligned} Q_T &= 1.401 \times 10^6 \text{ Btu/hr} \\ &= \underline{410 \text{ Kw}} \end{aligned}$$

2. Bulk Water Temperature Rise

$$\Delta T_B = \frac{\Delta Q_T}{\dot{m} C_p}$$

ΔT_B = bulk water temperature rise

Q_T = heat transferred to coolant

$\dot{m}$ = mass rate of flow, lb_m/sec

C_p = specific heat of water @ 2000 psi
and 480°F

$$\dot{m} = \frac{R_L}{v}$$

R_L = flow rate, gpm

v = specific volume H_2O (ft^3/lb) @ 2000 psi and 480°F

$$\dot{m} = \frac{(150 \text{ gal/min})(60 \text{ min/hr})}{(0.0197 \text{ ft}^3/LB_M)(7.48 \text{ gal/ft}^3)} = \frac{6.1 \times 10^4 \text{ LB}_M/\text{hr}}{}$$

$$\Delta T_B = \frac{1.401 \times 10^6 \text{ Btu/hr}}{(6.1 \times 10^4 \text{ LB}_M/\text{hr})(1.131 \text{ Btu/LB}_M\text{-}^\circ\text{F})} = \underline{20.3^\circ\text{F}}$$

3. Water Temperature Rise Within the Basket

$$\Delta T = \frac{Q}{\dot{m}C_p}$$

Q = fission heat + gamma heat of fuel rods + 1/2 gamma heat of basket = $1.308 \times 10^6 \text{ Btu/hr}$

$$\dot{m} = \frac{(130 \text{ gal/min})(60 \text{ min/hr})}{(0.0197 \text{ ft}^3/LB_M)(7.48 \text{ gal/ft}^3)} = 5.28 \times 10^4 \text{ LB}_M/\text{hr}$$

$$\Delta T = \frac{1.308 \times 10^6 \text{ Btu/hr}}{(5.28 \times 10^4 \text{ LB}_M/\text{hr})(1.131 \text{ Btu/LB}_M\text{-}^\circ\text{F})} = \underline{21.9^\circ\text{F}}$$

4. Maximum Bulk Water Temperature (Outlet)

$$\begin{aligned} T_B (\text{max}) &= T_{\text{inlet}} + \Delta T_{\text{Basket}} \\ &= 480^\circ\text{F} + 21.9^\circ\text{F} = \underline{502^\circ\text{F}} \end{aligned}$$

with hot channel factor

$$\begin{aligned} \text{Adj. } T_B &= (\text{calc } \Delta T_B) F_B + \Delta T_{\text{add}} + T_{\text{inlet}} \\ &= (21.9^\circ\text{F})(1.14) + 6^\circ\text{F} + 480^\circ\text{F} \\ &= \underline{511^\circ\text{F}} \end{aligned}$$

5. Maximum Cladding Surface Temperature

from CHEAT computer code (Appendix C), $T = \underline{603^\circ\text{F}}$

with hot spot factor

$$\begin{aligned} (T_c)_{\text{max}} &= T_B + \Delta T_B \text{ at hot spot} + \Delta T_{\text{film}} F_h + \Delta T_{\text{add}} \\ &= 480^\circ\text{F} + 11^\circ\text{F} + (103^\circ\text{F})1.39 + 6^\circ\text{F} \\ &= \underline{640^\circ\text{F}} \end{aligned}$$

C. Hot Spot and Hot Channel Factor Analysis⁽³⁾

The results of the hot channel and hot spot analysis indicate the saturation temperature of the loop may be exceeded under reactor operating conditions of 480°F inlet and 150 gpm flow. Recent tests in the ETR P-7 loop⁽⁴⁾ operated with clad surface temperatures above the saturation temperature with no adverse effects. The peak rod powers calculated for the defect test are believed to be conservative⁽⁵⁾, i.e., peak rod powers are likely to be less than those indicated in Figure 4A. Also, additional flow capacity of 16 gpm is available to suppress nucleate boiling should maximum water temperatures become a limiting consideration⁽¹⁾.

Hot Spot and Hot Channel Factors

	<u>Hot Channel Factor, F_b</u>	<u>Hot Spot Factor F_H</u>	<u>Additive Factors $\Delta T_{Add}, ^\circ F$</u>
1. Mechanical Tolerances			
a. Effect on Mass Flow- External Effects	1	1	
b. Effect on Mass Flow- Entrance and Exit Effects	1	1	
c. Effect on Mass Flow- Channel Effects	1	1	
d. Effect on D Variation	1	1.01	
2. Mechanical Distortion			
a. Effect on Mass Flow- External Effects	1	1	
b. Effect on Mass Flow- Entrance Effects	1	1	
c. Effect on Mass Flow- Channel Effects	1	1	
d. Effect of D Variation	1	1	
3. Correlation Equation on h	1	1.20	

	Hot Channel Factor, F_b	Hot Spot Factor F_H	Additive Factors $\Delta T_{Add}, ^\circ F$
4. Determination of Critical Parameters			
a. Flow Measurement			
b. Temperature Measure- ments (metal)]	1.03	1.03	1.4
c. Pressure Measurements]			
d. Temperature Measure- ments (coolant)	1	1	4.3
5. Flow Distribution Effects			
a. Plenum Effects	1	1	
b. Edge Effects	1	1	
6. Neutron and Gamma Flux Distribution			
a. Vertical Distribution	1	1	
b. Horizontal Distribution	1	1	
c. Internal Distribution	1	1	
7. Neutron and Gamma Flux Uncertainty			
a. Unperturbed Flux Source	1	1	
b. Average Depression	1	1	
c. Average Distribution	1	1	
d. Experimental power generation variation timewise	1.1	1.1	
8. Heat Flux Over Element Surface			
a. Fuel Homogeneity	1	1.01	
b. Mechanical Effects (Meat & Clad thicknesses)	1	1	
	<u>1.14</u>	<u>1.39</u>	<u>6°F</u>

Appendix B: References

1. Letter from C.E. Leach to C.R. Hann, dated 12/30/69, "Hydraulic Tests of ETR-M3 Loop 19-rod Defect Test Assembly."
2. C.E. Leach, "Flow Pressure Drop Characteristics of: 19-Rod Wire Wrapped Fuel Bundle, Annular Orifice, Annular Flow Passage and In-Pile Tube Simulating the ETR-M3 Loop," BNWL-CC-2431, January 1970.
3. R.J. Nertney, "Calculated Surface Temperatures For Nuclear Systems and Analysis of Their Uncertainties," IDO-16343, June 1957.
4. R.K. Marshall, "Proposal to Irradiate UO_2 Fuel Elements in the ETR P-7 Loop: GEH-12-44-100," Battelle-Northwest Report, BNWL-CC-1789, August 1968.
5. Letter from J.W. Kutcher to M.D. Freshley, dated 12/22/69, "Physics Calculations on M-3 Loop in ETR."

Appendix C: Fuel Temperature Calculations

A. Calculated Fuel Temperatures:

Fuel temperatures were calculated using the CHEAT heat transfer code which calculates fuel temperatures using a film coefficient obtained from the Modified Colburn Equation. The effects of self-shielding across the fuel rod and temperature dependent thermal conductivity are also considered. Typical CHEAT outputs are shown in Figures 1C, 2C, and 3C.

Input:

1. Rod outside diameter	0.565 in.
2. Rod inside diameter	0.505 in.
3. Coolant surface velocity	12 ft/sec.
4. Equivalent diameter	0.342
5. Coolant Temperature	500°F
6. Core-cladding coefficient	1000, 1500, & 2000 Btu/hr-ft ² -°F
7. Zircaloy thermal conductivity	8.4 Btu/hr-ft ² -°F
8. Radial power generation: from THERMOS	(Figures 2A & 3A)
9. Thermal conductivity from GEAP-5100-1(1)	

Summary of Calculated Fuel Temperatures for 23 Kw/ft (Figures 4C, 5C, and 6C):

	<u>Small Gap - Solid Pellets*</u>	<u>Large Gap - Solid Pellets**</u>	<u>Small Gap - Cored Pellets***</u>
1. Film Coefficient, Btu/hr-ft ² -°F	5169	5169	5169
2. Cladding Outside Surface Temp., °F	603	603	603
3. Cladding Inside Surface Temp., °F	770	770	770
4. Fuel Surface Temp., °F	1066	1165	1066
5. Fuel Center Temp., °F	4550	4695	3800

* (hg = 2000 Btu/hr-ft²-°F)

** (hg = 1500 Btu/hr-ft²-°F)

*** (hg = 2000 Btu/hr-ft²-°F)

B. Calculated Cladding & Fuel Temperatures with Hot Spot Factor, F_h

1. Cladding Temperature:

$$\begin{aligned} T_{\max} &= 770^{\circ}\text{F} + \Delta T_{\text{add}} + (\Delta T_f F_h - \Delta T_f) \\ &= 770^{\circ}\text{F} + 6^{\circ}\text{F} + 40^{\circ}\text{F} \\ &= \underline{816^{\circ}\text{F}} \end{aligned}$$

2. Fuel Surface Temperature:

Consider large gap-solid pellet and $h_{\text{gap}} = 1000$
Btu/hr-ft²-°F

Extrapolation of CHEAT calculations made at

$h_{\text{gap}} = 1500$ & 2000 Btu/hr-ft²-°F yields

$$T = 1215^{\circ}\text{F}$$

$$\text{with } F_h, T = 1215^{\circ}\text{F} + 46^{\circ}\text{F} = \underline{1260^{\circ}\text{F}}$$

3. Maximum Fuel Temperatures:

a. Small gap - solid pellet and $h_{\text{gap}} = 1500$ Btu/hr-ft²-°F
from Figure 4C, $T_{\max} = 4620^{\circ}\text{F}$ at 22.6 kW/ft

$$T_{\max} = 4620^{\circ}\text{F} + 46^{\circ}\text{F} = \underline{4665^{\circ}\text{F}}$$

b. Large gap - solid pellet and $h_{\text{gap}} = 1000$ Btu/hr-ft²-°F
from Figure 5C, $T_{\max} = 4855^{\circ}\text{F}$ at 22.6 Kw/ft

with F_h

$$T_{\max} = 4855^{\circ}\text{F} + 46^{\circ}\text{F} = \underline{4900^{\circ}\text{F}}$$

c. Small gap - cored pellet and $h_{\text{gap}} = 1500$ Btu/hr-ft²-°F
from Figure 6C, $T_{\max} = 3815^{\circ}\text{F}$ at 23.3 Kw/ft

$$T_{\max} = 3815^{\circ}\text{F} + 46^{\circ}\text{F} = 3860^{\circ}\text{F}$$

Figure 1C

SAMPLE CALCULATION FOR SOLID AND DISHED UO₂-4 WT% PuO₂ PELLET FUEL RODS OPERATING AT 23 kW/FT WITH A FUEL/CLAD CONTACT COEFFICIENT OF 2000 BTU/HR-FT² °F

Figure 2C

SAMPLE CALCULATION FOR SOLID AND DISHED UO_2 -4 WT% PuO_2 PELLET FUEL RODS OPERATING AT 23 KW/FT WITH A FUEL/CLAD CONTACT COEFFICIENT OF 1500 BTU/HR-FT² °F

Figure 3C

SAMPLE CALCULATION FOR A CORED UO₂-4.4 WT% PuO₂ PELLET FUEL ROD OPERATING AT 23 kW/FT WITH A FUEL/CLAD CONTACT COEFFICIENT OF 2000 BTU/HR-FT² °F

Figure 4C - Maximum Fuel Temperature Versus Peak Rod Power for Small Gap
UO₂-4 wt% PuO₂ Pellet Fuel

Figure 5C - Maximum Fuel Temperature Versus Peak Rod Power for Large Gap
 UO_2 -4 wt% PuO_2 Pellet Fuel

Figure 6C - Maximum Fuel Temperature Versus Peak Rod Power For Small Gap Cored UO_2 -4.4 wt% PuO_2 Pellet Fuel

Appendix C: References:

1. Lyons, M. F. et al, "UO₂ Powder and Pellet Thermal Conductivity During Irradiation," General Electric Company Report, GEAP-5100-1, March 1966.

Appendix D: Defect Test Element Drawings

QTY	PART NO.	DESCRIPTION	MATL OR REF
1	1	ARRANGEMENT	
1	2	COUPLING	
1	3	BASKET	
1	4	QUICK-RELEASE PIN (MODIFY)	
1	5	QUICK DISCONNECT ASSY	M-3-29641
1	6	FUEL ELEMENT ASSY	M-3-29636
21	7	SPACER

1 ARRANGEMENT

2 304 SST

- GENERAL NOTES**
(UNLESS OTHERWISE SPECIFIED)
1. TOLERANCES: FRACTIONAL: 1/16, DECIMAL: .005, ANGULAR: .030°
 2. ALL MACHINED SURFACES: 63 SURFACE TEXTURE SHALL BE IN ACCORDANCE WITH: USAS1 B46.1 (LATEST REV).
 3. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS.
 4. ALL MATL SHALL BE AS SPECIFIED OR APPROVED EQUAL QUALITY.

1/8 DIA, 3 SPACERS EQ SP ON PERIPHERY, TYP EVERY OTHER LEVEL ROTATED 60° TYP. (E.B. WELD BEAD, FINISH TO 3.410 DIA)

3 304 SST

NOTE: DIMENSION NOMINALLY 1/2\"/>

SECTION 'A-A'

REV	DATE	DESCRIPTION
1	1/15/67	1) ADDED PART #7 ZONES A-10
2	1/15/67	2) ADDED NOTE ZONE E-11
3	1/15/67	3) DIM WAS FRACT. ZONE C-9
4	1/15/67	4) ADDED NOTE, ZONES A-B13
5	1/15/67	5) CHANGED NOTE, ZONE A-12
6	1/15/67	6) DIM WAS 6\"/>

REV	DATE	DESCRIPTION
1	1/15/67	1) ADDED PART #7 ZONES A-10
2	1/15/67	2) ADDED NOTE ZONE E-11
3	1/15/67	3) DIM WAS FRACT. ZONE C-9
4	1/15/67	4) ADDED NOTE, ZONES A-B13
5	1/15/67	5) CHANGED NOTE, ZONE A-12
6	1/15/67	6) DIM WAS 6\"/>

U. S. ATOMIC ENERGY COMMISSION	DATE	4703
RICHLAND OPERATIONS OFFICE	BY	CE LEACH
PACIFIC NORTHWEST LABORATORY	DATE	1/15/67
OPERATED BY BATTTELLE MEMORIAL INSTITUTE	BY	CE LEACH
ETR-M3 19 ROD DEFECT TEST	DATE	1/15/67
ARRANGEMENT	DATE	1/15/67
55 AA	DATE	1/15/67
H-3-29631	DATE	1/15/67

QTY	PART NO.	DESCRIPTION	DWG NO. OR MATL
	1	ARRANGEMENT	
	2	FUEL ROD ASSEMBLY - NO WIRE	
	3	FUEL ROD ASSEMBLY - CW WIRE	
	4	FUEL ROD ASSEMBLY - CCW WIRE	
	5	FUEL ROD ASSEMBLY - REPLACEABLE	
	6	CENTER PIN	SN #2
	7	NUT	
	8	END BRACKET-TOP (MODIFY)	
	9	END BRACKET-BOTTOM (MODIFY)	
	10	FLUX MONITOR	
	11	PIN .124-.125 DIA x .436 LGNS	ZIRCALOY-2
	12	BAND $\frac{1}{8}$ WIDE x .010 THICK	ZIRCALOY-2
2 2 2 2	13	PELLET .500 DIA x .415 LONG (W-B GRN)	EXTRACTED UO ₂
1 1 1 1	14	END CAP - TOP	SN #2
1 1 1 1	15	END CAP - BOTTOM	SN #2
1 1 1 1	16	TUBE	
1	17	END CAP - TOP	SN #2
AR AR AR AR	18	FUEL	UO ₂ -2F ₂ O ₄
1 1 1 1	19	SPRING, COMPRESSION .500 ID x .575 OD x .125 DIA WIRE x 3.43 FREE LG, SQ L GRIND ENDS	INCONEL X SPRING TEMPER
AR AR	20	WIRE .072 DIA	ZIRCALOY-2

GENERAL NOTES (UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: FRACTIONAL $\pm .005$ DECIMAL $\pm .0005$ ANGULAR $\pm 0^{\circ}30'$
2. ALL MACHINED SURFACES $\sqrt{32}$ SURFACE TEXTURE SHALL BE IN ACCORDANCE WITH USAS1 B46.1 (LATEST REV)
3. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS
4. ALL MATL SHALL BE AS SPECIFIED OR APPROVED EQUAL QUALITY.

U. S. ATOMIC ENERGY COMMISSION RICHMOND OPERATIONS OFFICE PACIFIC NORTHWEST LABORATORY OPERATED BY BATTTELLE MEMORIAL INSTITUTE	19 ROD FUEL ELEMENT ARRANGEMENT
ETD-83 19 ROD DEFECT TEST	55 88 9703
H-3-29836	1 2

Technical drawing of a fuel element assembly, showing three views: a side elevation, a cross-section 'A-A', and a cross-section 'B-B'.

Side Elevation View:

- Dimensions: 3.240 DIA, 3.236 DIA, 3.343 DIA, 3.250, 2 (REF), 3 PLACES EQ SP AT ASSY, SEE SECTION 'A-A' FOR ORIENTATION.
- Callouts: 3, 4, 2, 5, 6.

Cross-section 'A-A':

- Dimensions: .567 DIA THRU, .565, 16 GAGE (.062), 45°, 60°, 60°, 15 R, 11 R, 16 R, 1/4, 1/8 (TYP), 1/4 DIA, 3 RODS EQ SP ON 1 1/2 DIA BC, 1/4 DIA, 1/8 DEEP, .251 DIA THRU & CBORE, .250 DIA - .15 DEEP, .375 DIA - .32 DEEP, TYP 3 PLACES, 1/4 (TYP), 3/75, 1/4, 1/8, 1/8, 1/2, 1/32 = 45° CHAMFER.
- Callouts: MARKS DEFECT POSITION ON FUEL ELEMENT (N-3-29636), ORIENTATION OF MAJOR BUSSET & ORIENTATION MARK ON PART #2, ELECTRON BEAM WELD SOX PENETRATION (TYP), 16 BEFORE WELDING.

Cross-section 'B-B':

- Dimensions: 1.3 DIA, 1.4 DIA (TYP), 1/4 DIA, 1/8 DIA.
- Callouts: A, B, TYP.

[illegible]

Technical drawing of a mechanical part, showing a top view and a side view.

Top View: A hexagon with a central hole. The outer diameter is labeled $1 \frac{1}{16}$ HEX.

Side View: A cross-section of the part with various dimensions and features labeled.

- $1 \frac{1}{16}$ HEX
- 11UNS-28 DOUBLE TND
- $\frac{1}{16}$ 45° CHAMFER
- 10°30'
- $\frac{1}{16}$ R
- TND RELIEF
- $\frac{19}{32}$
- $\frac{9}{32}$
- .547 D/A
- .565 D/A
- .688 D/A
- .606 D/A
- .834 D/A
- .832

Technical drawing of a shaft-hub assembly. The shaft has a diameter of $\frac{249}{240}$ DIA. The hub has a bore diameter of $\frac{374}{373}$ DIA. The shaft length is $\frac{1.270}{1.268}$. The hub length is $\frac{1}{4}$. The shaft is secured with a $\frac{1}{8}$ SPN R. The hub has a $\frac{1}{16}$ keyway.

GENERAL RULES

(UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: FRACTIONAL $\pm .005$, DECIMAL $\pm .005$, ANGULAR $\pm 0'30''$
2. ALL MACHINED SURFACES SURFACE TEXTURE SHALL BE IN ACCORDANCE WITH, ASATI DAS.1 (LATEST REV).
3. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS.
4. ALL MAT. SHALL BE AS SPECIFIED OR APPROVED EQUAL QUALITY.

[illegible]

- GENERAL NOTES
(UNLESS OTHERWISE SPECIFIED)
1. TOLERANCES: FRACTIONAL 1/1000, DECIMAL .0005, ANGULAR 0°30'
 2. ALL UNFINISHED SURFACES
 3. MAX RADIUS OR CHAMFER = 0.005"
 4. ALL PELLETS SHALL BE UO₂-PuO₂ (94% T.O.)

U. S. ATOMIC ENERGY COMMISSION RICHLAND OPERATIONS OFFICE PACIFIC NORTHWEST LABORATORY OPERATED BY BATTELLE MEMORIAL INSTITUTE			
DRAWING NO. 1-3 29632		SHEET 1	
TITLE: MIXED OXIDE FUEL PELLETS DETAILS		SCALE: NONE	
DATE: 11-28-64		BY: J. B. B.	
CHECKED BY: J. B. B.		DATE: 11-28-64	
APPROVED BY: J. B. B.		DATE: 11-28-64	
REVISIONS		DESCRIPTION	
1	GEN REVISION		
2	GEN REVISION		
REFERENCE DRAWINGS		DRAWING STATUS	
NEXT USED ON		DATE	