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SAXTON PLUTONIUM PROGRAM

MECHANICAL, THERMAL AND HYDRAULIC DESIGN OF SAXTON PARTIAL PLUTONIUM CORE

By

E. A. Bassler
D. C. Fischer
N. J. Georges
E. A. McCabe

Approved By: N R Nelson
N. R. Nelson, Manager
Saxton Plutonium Project

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WESTINGHOUSE ELECTRIC CORPORATION
Atomic Power Division
P. O. Box 355
Pittsburgh, Pennsylvania 15230

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ABSTRACT

The purpose of the Saxton Plutonium Project is to develop information concerning the utilization of plutonium enriched fuel in pressurized water reactor systems, through design, fabrication and operation of a partial core of PuO_2-UO_2 fuel in the Saxton Reactor.

Saxton Core I contained enriched UO_2 fuel in all 21 of its assemblies. This report describes the Mechanical Design and the Thermal and Hydraulics Design for Saxton Core II, which contains nine centrally located PuO_2-UO_2 fuel assemblies and twelve peripheral UO_2 assemblies. The work on the PuO_2-UO_2 portion of Core II was carried out for the Joint US-Euratom R&D Board under contract number AT(30-1)-3385 administered by the USAEC New York Operation Office.

Design guidelines for the PuO_2-UO_2 portion of Saxton Core II included: (a) 20,000 MWD/tonne peak rod average burnup, (b) 16 kw/ft maximum heat rate in the rods, (c) at end of design life, internal gas pressure to be less than external reactor operating pressure, and (d) fuel rod outside diameter, length and lattice spacing to be the same as for the UO_2 rods in Cores I and II. Related work in the fields of Nuclear Design, Materials Design and Fuel Fabrication and in Critical Experiments for the Saxton Partial Plutonium Core is described in EURAEC's 1490, 1492 and 1493 respectively.

INTRODUCTION

The primary purpose of the Saxton Reactor has been to make available a small version of modern commercial, closed cycle, chemical shim control reactors for use with a post construction research and development program directed towards further improvements in nuclear power economics. Consequently, the first core was designed, not as a physics experimental project by itself but instead as a reliable source of nuclear power using the latest design techniques available at the time as developed in conjunction with the Yankee reactor, the multi-region reactor program, the large reactor development program and the SELNI reactor.

One of the most important parts of the Saxton R&D program includes investigations in the fields of fuel development and chemical shim reactivity control to increase the power generation per unit volume of core. Consequently, the size of the core was selected to permit "pushed" operation. This consideration led to the use of an active core containing 21 assemblies even though the reactor vessel contained space for a total of 32 assemblies. Five of the 21 assemblies were made "annular" so that they could accommodate removable 3 x 3 subassemblies which can be made specially enriched and instrumented. These subassemblies may be inserted or removed through ports in the reactor vessel head. Hence experiments carried on in these five regions can be changed or examined without removing the vessel head.

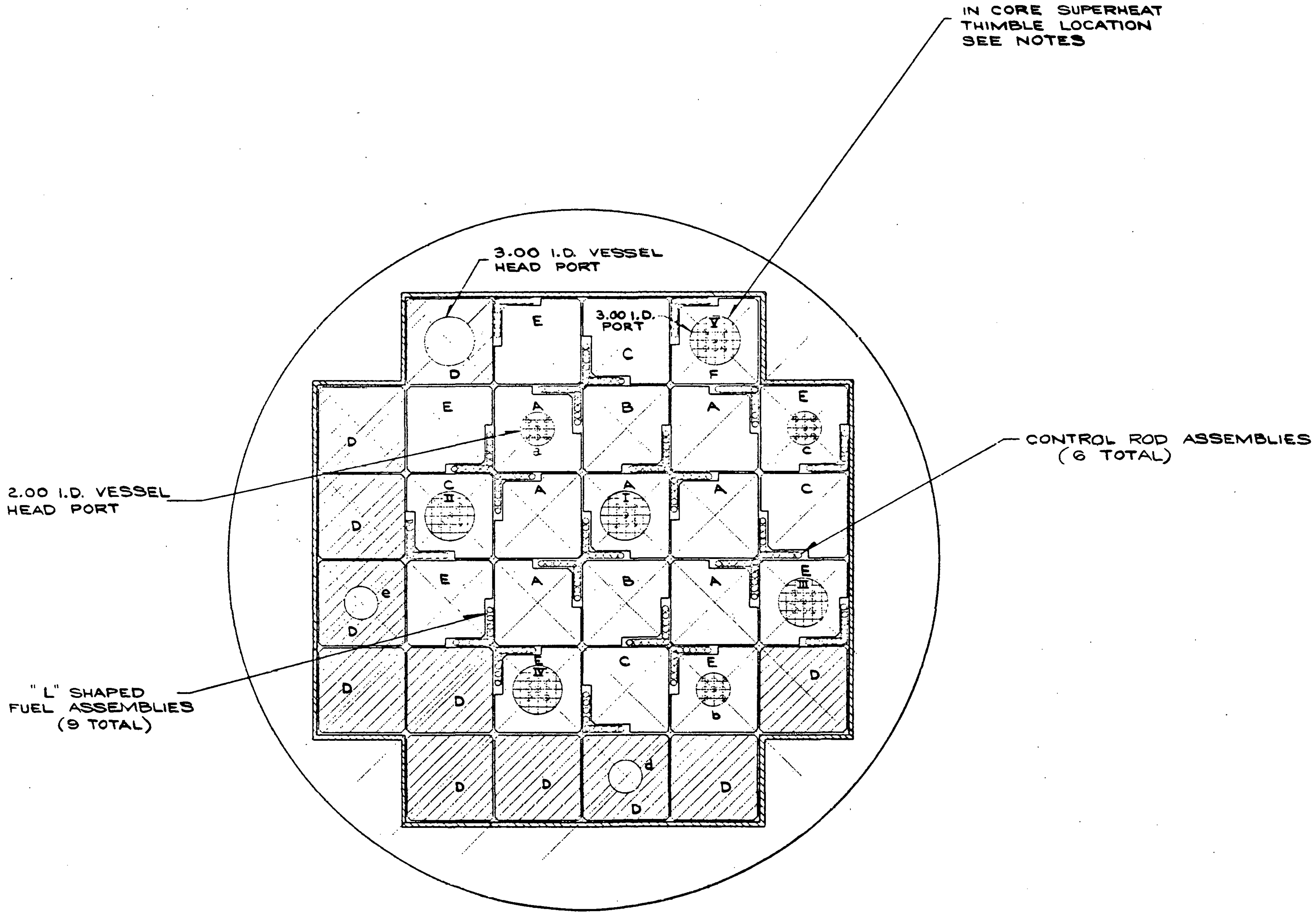
In Saxton Core I, the basic fuel element was made with .391 inch O.D. type 304 stainless steel tubes with a wall thickness of .015" containing uranium dioxide fuel as cylindrical ceramic pellets. The pellets were .357" diameter and .732" long with dished ends and fifty such pellets were placed in each tube with no discs or spacers between pellets.

Fuel follower assemblies were used to occupy the water slots created when control rods were withdrawn.

Because each fuel assembly was slotted to accommodate one half of an offset cruciform control rod, the outer nine assemblies, which do not receive control rods, were filled with "L" shaped sets of fuel rods as inserts. These insert tubes were made of type 348 stainless steel with a wall thickness of .028" giving an O.D. of .417 in. Type 348 stainless steel was used because the insert assembly was brazed together rather than assembled within spring clip grids as were the main assemblies.

The active height of the fuel was 36.6" and the equivalent diameter of the core was 28.1".

Figure 220-1 shows the Saxton Reactor Core Cross Section and shows which assemblies in Core I were replaced by PuO_2-UO_2 fuel assemblies in Core II. The control rods, followers and "L" shaped fuel assemblies used in Core I were retained and are being used again in Core II but in different locations within the core.



NOTES:

a TO e = OUTLET INSTRUMENTATION PORTS.
I TO IV = REMOVABLE FUEL SUB-ASSEMBLIES
V = SUPERCRITICAL FUEL ASSEMBLY OR INTERCHANGEABLE DUMMY ASSEMBLY.

	NO. OF ASSEMBLIES
A. PLUTONIUM PELLETIZED FUEL ASSEMBLIES.	7
B. PLUTONIUM VIBRATORY COMPACTED FUEL ASSEMBLIES.	2
C. CORE I TYPE UO_2 FUEL ASSEMBLIES.	4
D. DUMMY FUEL ASSEMBLIES	11
E. CORE II TYPE UO_2 FUEL ASSEMBLIES.	7
F. SPECIAL 49 ROD CORE UO_2 ASSEMBLY.	1
TOTAL ASSEMBLIES	32
TOTAL NO. OF AUXILIARY "L" FUEL ASSEMBLIES	9
TOTAL NO. OF FUELED FOLLOWERS	6
SECONDARY SOURCE RODS -	2
REMOVABLE FUEL RODS -	20 (11 UO_2 & 9 $PuUO_2$)

TOTAL NO. OF FUEL RODS (EXCLUDING REMOVABLE TEST ASSEMBLIES)
- CORE - 884 UO_2
- 630 $PuUO_2$
- FOLLOWERS - 108
- 1622

WEIGHT OF FUEL (EXCLUDING TEST ASSEMBLIES)
- CORE - 1182 LB. - (UO_2)
- 760 LB - ($PuUO_2$)
- FOLLOWERS - 137 LB. (ORIGINAL CORE I WT. UNCORRECTED FOR BURNUP.)
- TOTAL 2079 LB.

Figure 220-1 Saxton Reactor Core Cross-Section

SAXTON CORE II MECHANICAL DESIGN

CORE LOADING

The Saxton plutonium program utilizes a total of 638 plutonium fuel rods which have been loaded into nine main fuel assemblies of the standard Saxton Core II designs and one removable fuel subassembly. These fuel assemblies have been positioned centrally in the core as shown in Figure 220-1. A breakdown of the 638 rods showing the types and numbers in each fuel assembly is given in Table 220-1, arranged by groups defined in Figure 220-2.

MAIN FUEL ASSEMBLIES

The main fuel assemblies containing plutonium fuel rods are shown in Figure 220-2. In addition to the fuel rods, the basic components of each fuel assembly consist of four grids, two enclosure halves, and one each top and bottom nozzles. The grid assemblies are of brazed "egg crate" construction and are spaced axially at ten inch spans to provide lateral support for the fuel rods. The enclosure halves are welded to the peripheral straps of the grid assemblies to support the grids and to tie the fuel assemblies together. The nozzles, which provide a means of handling the fuel assemblies and of positioning the assemblies in the reactor core, are welded to the top and bottom ends of the enclosure halves. A photograph of a typical finished fuel assembly is given in Figure 220-3

The fuel rods are arranged in a square lattice in a typical main fuel assembly with nine rods in each direction on a .580 inch pitch. Of the possible 81

Table 220-1

Types and Number of Plutonium Fuel Rods in Each Fuel Assembly

Assembly Type	Number of Assemblies	Number of Fuel Rods Per Assembly: Vibratory Compacted, Zr Clad	Vibratory Compacted, SST Clad	Pelletized, Zr Clad	Pelletized, SST Clad	Total No. of Rods
Main Fuel Assemblies						
Group 1	1	62	8			70
Group 2	5			70		350
Group 3	1			52	18	70
Group 4	1		2	57	2	61
Group 5	1	70				70
Removable Fuel Assembly	1	4 (2 Removable)		4 (2 Removable)		8
Removable Fuel Rods (For Main Assemblies)		2		7		9
						638

Figure 220-2 Saxton Reactor Plant - Plutonium Fuel Assembly

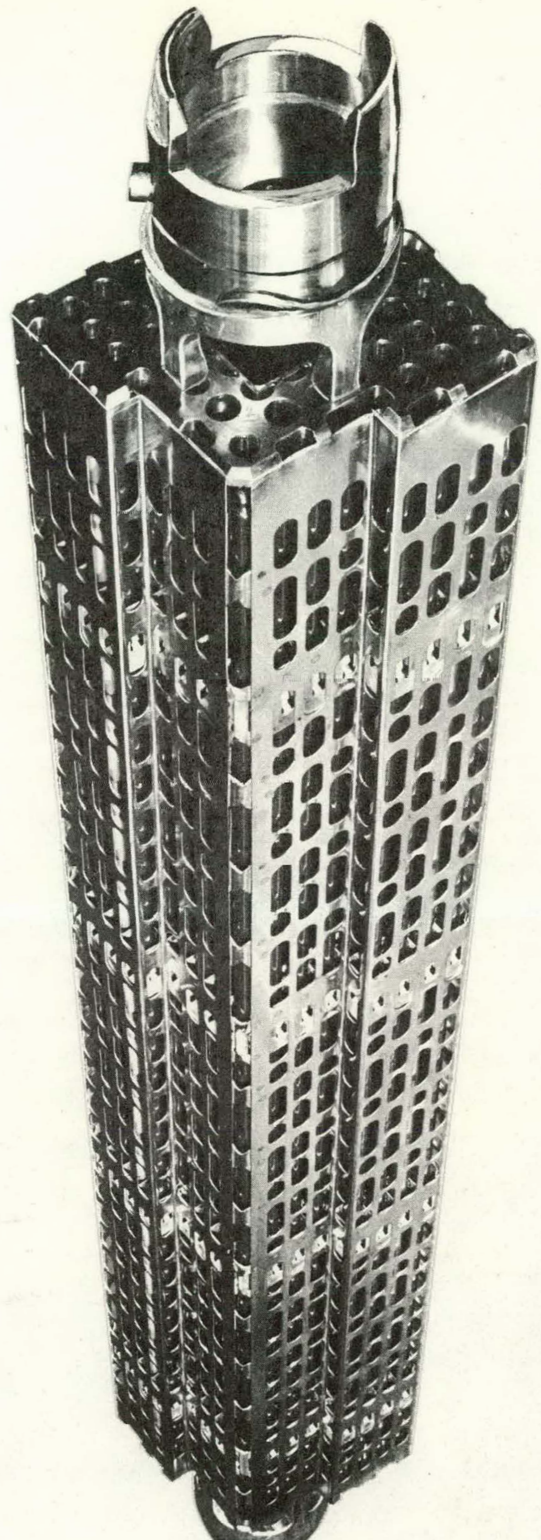


Figure 220-3 Saxton Plutonium Fuel Assembly

lattice locations in this pattern, 9 locations at one outer corner are eliminated from each fuel assembly to provide room for cruciform shaped control rods which are positioned between the assemblies. In addition, two lattice locations in each fuel assembly are left vacant during fabrication to allow for insertion of flux wire thimbles, source rods, or removable fuel rods in the assembly when the assembly is installed in the reactor.

It will be noted from Figure 220-2 that the group 4 plutonium fuel assembly has a square axial hole in place of the nine center fuel rods. This provision is made to allow for insertion of the removable plutonium fuel subassembly.

REMOVABLE FUEL SUBASSEMBLY

The removable plutonium fuel subassembly is shown in Figure 220-4. The subassembly contains eight fuel rods arranged in a square lattice with three rods per side. The ninth lattice location in the center does not contain fuel but is occupied by a flux wire thimble when the subassembly is installed in the reactor.

As with the main fuel assembly, the removable subassembly is constructed of four grids, two enclosure halves, and top and bottom end plates. In order to provide assembly clearance for insertion of the removable subassembly into the space normally occupied by the nine central fuel rods in the main assembly and to compensate for the added thickness of the enclosure and outer grid straps in the removable subassembly, the pitch between fuel rods in the subassembly has been reduced to .538 inches.

		BILL OF MATERIAL				NO. REQ.			
NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPECIFICATION	EQUIVALENT SPECIFICATION FOR USE ONLY	GR. 1	GR. 2	GR. 3	
	1	ENCLOSURE ASSEMBLY	540F533 Gr 2			1			
	2	TOP PLATE	882D432IT 1			1			
	3	BOTTOM PLATE	679C701 IT 3			1			
	4	FUEL ROD	540F555Gr 1			2			
	5	FUEL ROD	540F555 Gr 2			2			
	6	FUEL ROD	540F555GR3			2			
	7	FUEL ROD	540F555GR4			2			

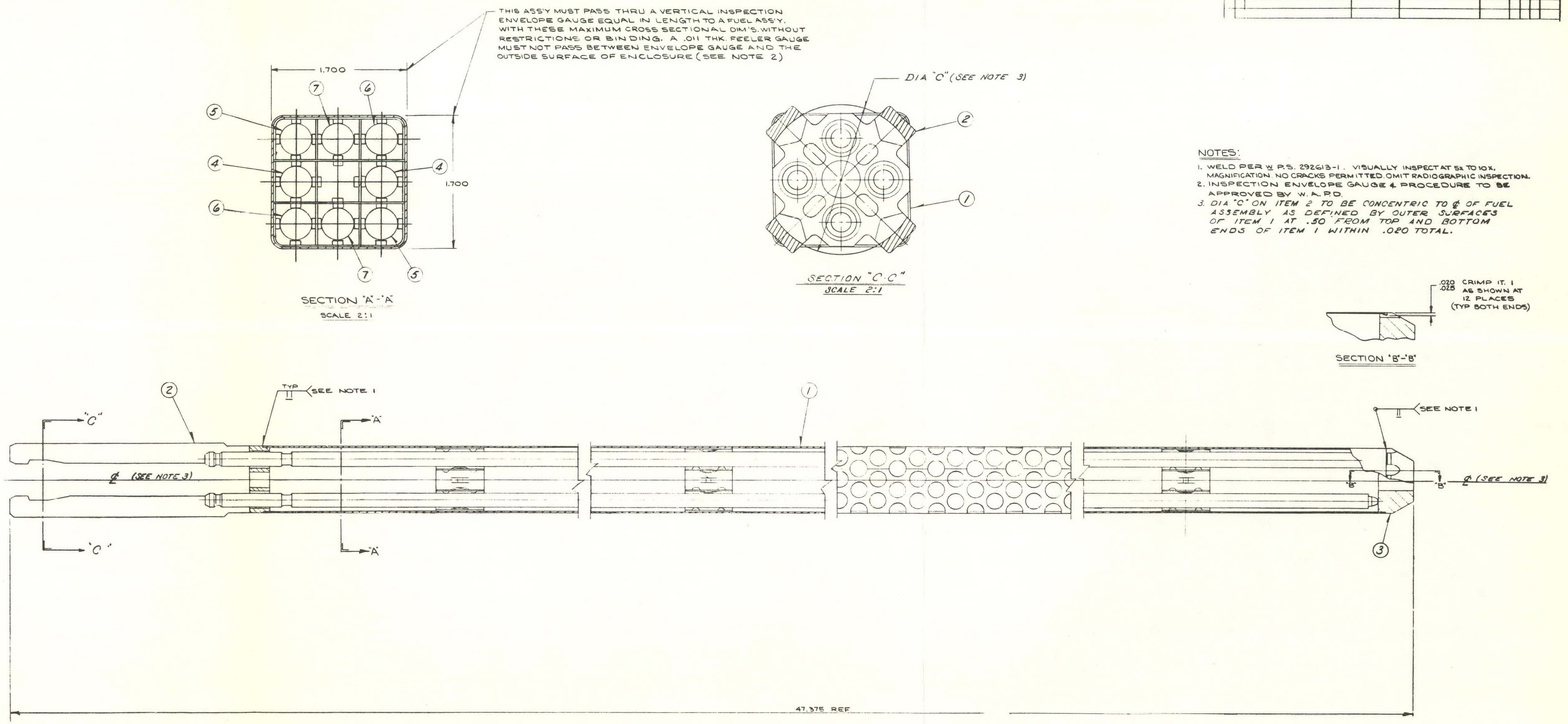


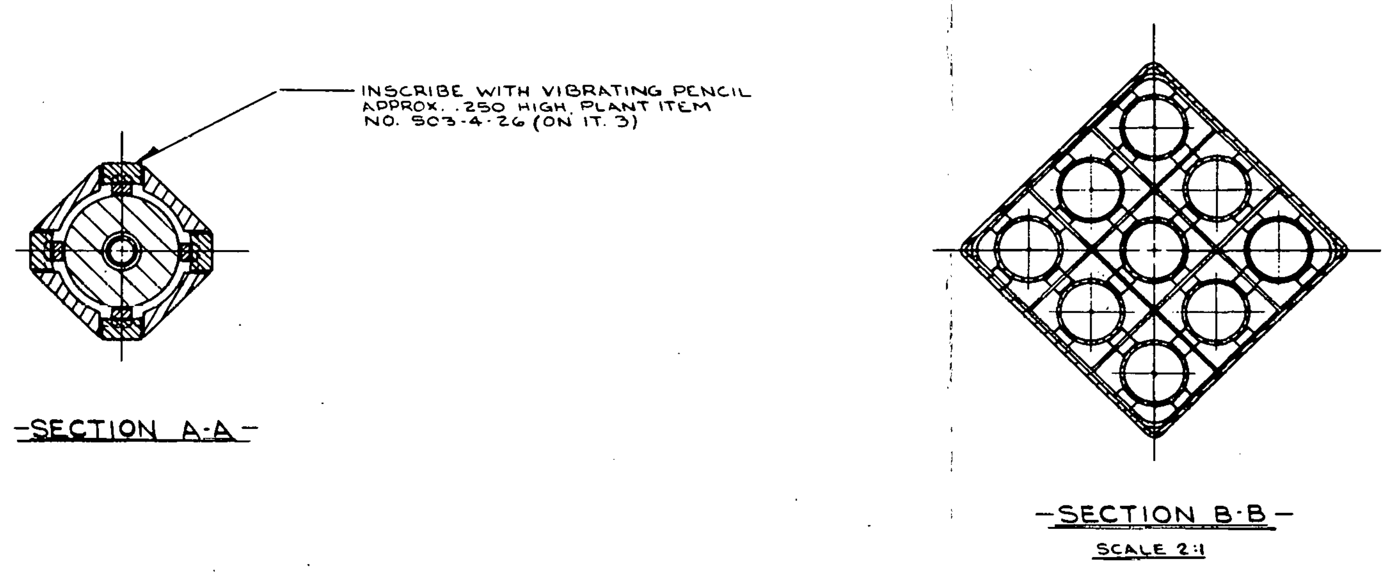
Figure 220-4 Removable Plutonium Fuel Subassembly

The removable subassembly was provided in the plutonium program to allow for periodic removal and visual inspection of typical plutonium fuel rods without the necessity of removing the reactor vessel head. The subassembly is inserted into and removed from the reactor through a port in the reactor top head. A removable support tube and latch assembly is used to handle the removable subassembly and to support it axially in the reactor. The support tube latches to the fingers on the top plate of the removable subassembly and hangs from a conoseal joint in the reactor head port to axially position and support the removable subassembly within the main fuel assembly. The complete removable fuel assembly with the support tube attached is shown in Figure 220-5.

In order to allow for periodic visual inspection of the plutonium fuel rods, four of the eight rods in the removable subassembly are of the removable type. The top end plug of the removable rods protrudes through the top plate on the subassembly, so that when the support tube is unlatched from the subassembly, the top end of the removable rods are accessible. Tooling is available at Saxton to grasp the top end plug of each removable rod,to remove the rods from the subassembly, and to position them before an underwater periscope for visual inspection.

The flux wire thimble which occupies the center position in the removable subassembly is actually a part of the support tube and latch assembly, and is supported axially from the conoseal adaptor at the top of the support tube. Thus, when the support tube is disconnected from the fuel subassembly to gain

BILL OF MATERIAL						NO. REQ.			
NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPECIFICATION	[illegible]	GR. 1	GR. 2	GR. 3	
	1	CONNECTOR	[illegible]			1			
	2	LATCH ASS'Y	[illegible]			1			
	3	FUEL SUB ASS'Y.	[illegible]			1			



NOTES:

1- WELD PER W P.S. 292613-1. OMIT RADIOGRAPHIC INSPECTION.

2- RECORD PLANT ITEM NO. OF EACH ASSEMBLY IDENT. NO. AND POSITION OF FUEL RODS, FUEL WEIGHT, & METAL WEIGHT.

3- *ASSEMBLY TO BE STRAIGHT WITHIN .200 AFTER LATCHING. METHOD AND PROCEDURE FOR CHECKING STRAIGHTNESS TO BE APPROVED BY WAPD DESIGN ENGINEER.*

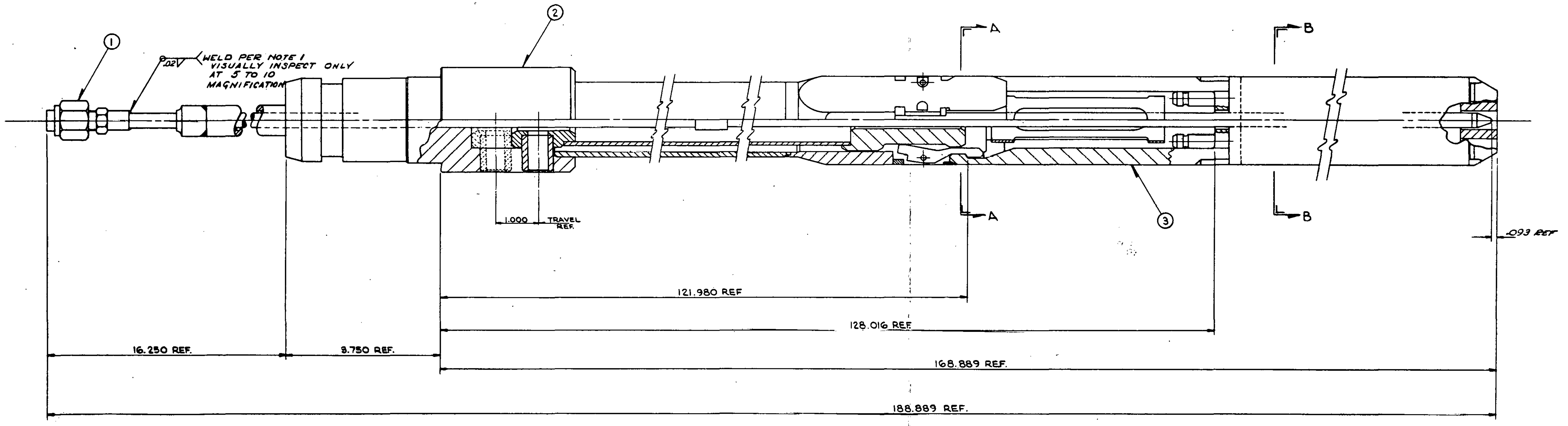


Figure 220-5 Removable Fuel Assembly

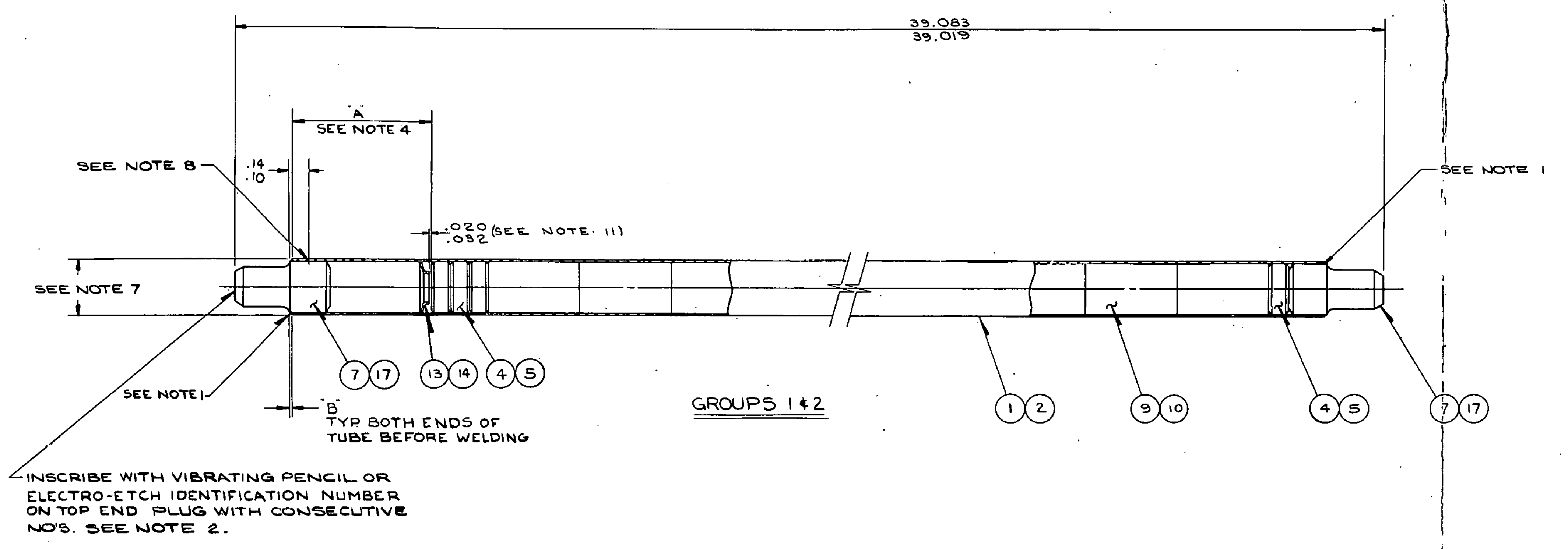
access to the removable fuel rods, the thimble will be withdrawn from the subassembly. A special tool has been designed and built to aid in guiding the thimble back into the subassembly when the support tube is reconnected.

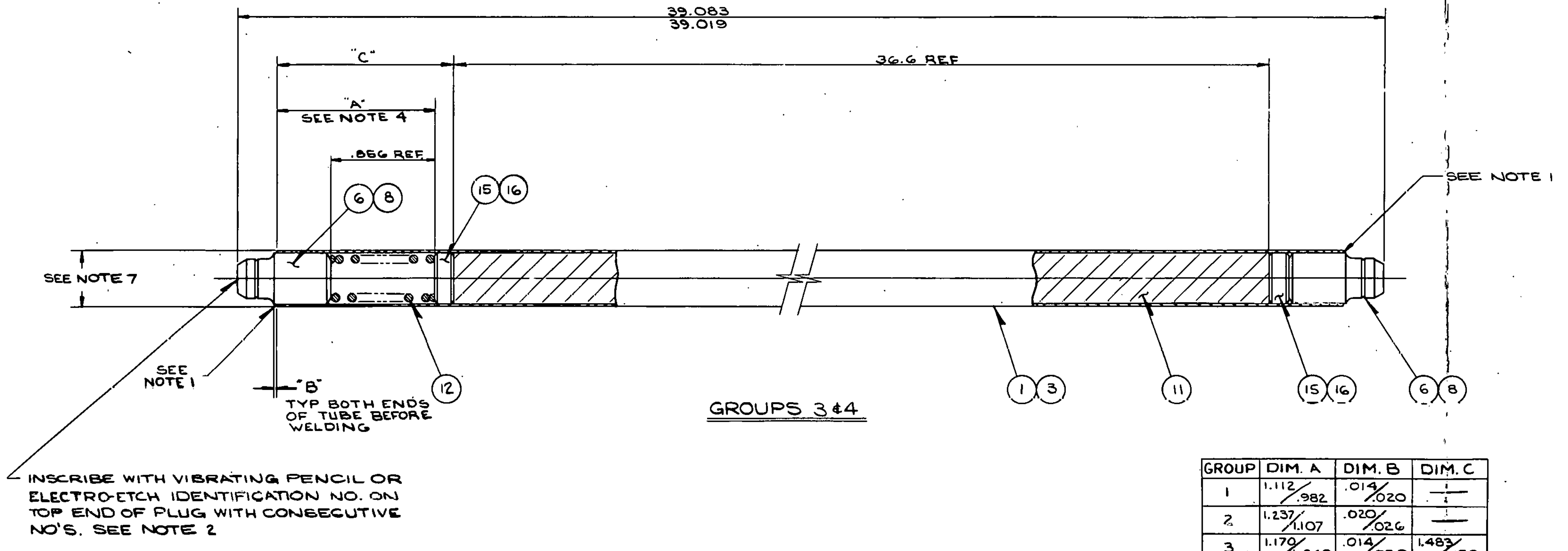
FUEL RODS

General Description

The fuel rods utilized in the Saxton plutonium program are shown in Figures 220-6, 220-7, and 220-8. The basic features of construction for the various types of rods are similar; i.e., they all consist of fuel encased in tubular cladding with welded end plugs and a hold down spring at the top of the fuel column to restrict axial motion of the fuel within the cladding. The fuel in all cases is composed of a mixture of natural uranium and plutonium dioxide powders in either the pelletized or vibratory compacted (VIPAC) form. Zircaloy-4 tubing is used primarily for the fuel rod cladding although a few 304 stainless steel clad fuel rods (30 total) have been included in the program for test comparison purposes.

Although Figures 220-6, 220-7, and 220-8 show ten different groups of fuel rods, the differences between groups, except for fuel and cladding, are confined to the configuration of the various end plugs external to the cladding. Thus, four basic types of fuel rods, as determined by the fuel and cladding, are utilized in the plutonium program. The design data for these four types of fuel rods are given in Table 220-2.





GROUP	DIM. A	DIM. B	DIM. C
1	1.112/.982	.014/.020	—
2	1.237/1.107	.020/.026	—
3	1.170/1.040	.014/.020	1.483/1.170
4	1.295/1.165	.020/.026	1.608/1.295

BILL OF MATERIAL						NO. REQ.				
NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPECIFICATION	EQUIVALENT SPECIFICATION FOR Ⓦ USE ONLY	GR. 1	GR. 2	GR. 3	GR. 4	GR. 5
	1	FUEL TUBE	499B772 IT. 4			1	—	1	—	
	2	FUEL TUBE	499B772 IT. 6			—	1	—	—	
	3	FUEL TUBE	674C863 IT. 1			—	—	—	1	
	4	FILLER	498B981 IT. 1			AS REQ	—	—	—	
	5	FILLER	498B981 IT. 2			—	AS REQ	—	—	
	6	END PLUG	674C886 IT. 2			—	—	2	—	
	7	END PLUG	674C886 IT. 3			—	2	—	—	
	8	END PLUG	674C886 IT. 1			—	—	—	2	
Y	9	PELLET STACK	500B071 GR. 2			1	—	—	—	
Y	10	PELLET STACK	500B071 GR. 3			—	1	—	—	
X/Y	11	FUEL COLUMN				—	—	AS REQ	AS REQ	
	12	SPRING	~~500B086 IT. 1~~			—	—	1	1	
	13	SPRING	500B157 IT. 1			1	—	—	—	
	14	SPRING	500B157 IT. 2			—	1	—	—	
	15	FILLER	500B158 IT. 1			—	—	AS REQ	—	
	16	FILLER	500B158 IT. 2			—	—	—	AS REQ	
	17	END PLUG	674C886 IT. 4			2	—	—	—	

"X" - FUEL COLUMN TO CONSIST OF A MIXTURE OF URANIUM DIOXIDE & PLUTONIUM DIOXIDE POWDER VIBRATORY COMPACTED IN ACCORDANCE WITH WAPD APPROVED PROCEDURE TO A DENSITY OF 87 ± 1% OF THEORETICAL. POWDER TO BE PREPARED FOR COMPACTION BY DYNAPAK OR EQUIVALENT PROCESS. ENRICHMENT OF UO_2, OXIDE COMPOSITION, PuO_2 CONTENT, AND PARTICLE SIZE & DENSITY FOR POWDER TO BE SPECIFIED ON P.O.

"Y" - GAS & VAPOR CONTENT OF FUEL TO BE LIMITED TO FOLLOWING:

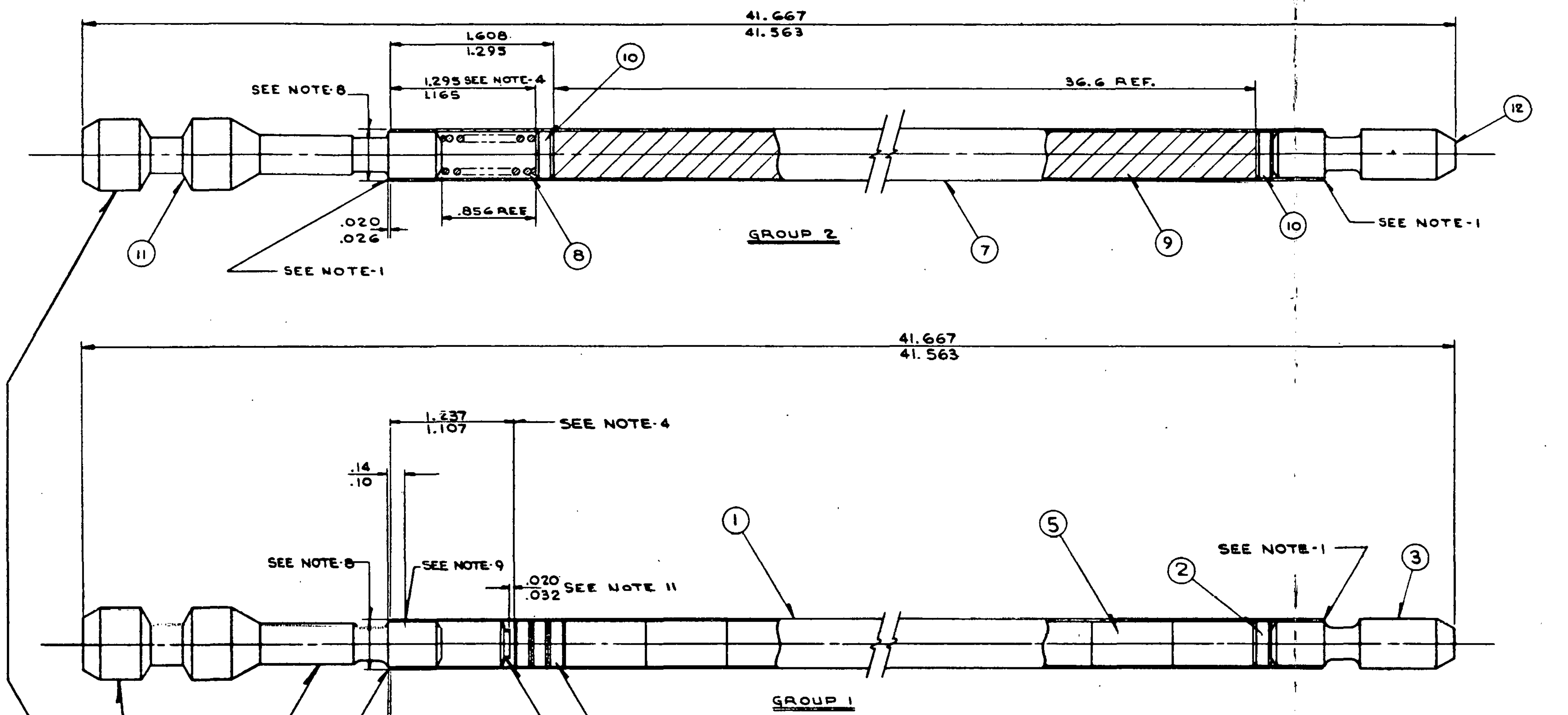
GROUPS 1 & 2 — H_2O - 30 PPM
N_2 - 75 PPM
H_2 - 15 PPM
TOTAL GAS - .05 scc/gm (EXCLUSIVE OF H_2O)

GROUPS 3 & 4 — H_2O - 100 PPM
N_2 - 100 PPM
H_2 - 20 PPM (Si-H & C-H BONDS)
TOTAL GAS - .06 scc/gm (EXCLUSIVE OF H_2O)

NOTES:

1. WELD PROCEDURE & INSPECTION TO BE APPROVED BY WAPD.
2. RECORD IDENT. NO., FUEL WT., & METAL WT.
3. MOISTURE CONTENT OF FUELS PRIOR TO WELDING TO BE AS SPECIFIED IN NOTE "Y". TUBE MUST BE DRY.
4. USE ONE FILLER AT BOTTOM OF FUEL TUBE AND ADD FILLERS TO TOP END AS NECESSARY (MIN. OF ONE ADDITIONAL AT TOP END TO OBTAIN THIS DIMENSION.
5. ROD ASS'Y MUST BE STRAIGHT WITHIN .010 PER FT. BETWEEN END PLUG WELDS WHEN LYING ON A SURFACE PLATE.
6. PRIOR TO ASS'Y, WIPE INSIDE & OUTSIDE SURFACES OF TUBE, SPACER, END PLUGS, & SPRING (USE ALCOHOL ONLY AS A SOLVENT) WITH SWABS TO REMOVE ALL FOREIGN MATTER. WIPE WITH DRY SWAB AFTER CLEANING.
7. MAX. DIA. PERMITTED AFTER WELDING AS FOLLOWS: GROUPS 1 & 3 - .395 DIA. GROUPS 2 & 4 - .399 DIA (BEFORE PICKLING) ALL SURFACES IN WELD AREA TO HAVE SMOOTH TRANSITION. CAUTION! NO GRINDING OF TUBE WALL PERMITTED ROLLING OF WELDS PERMITTED PROVIDED THE OPERATION DOES NOT EXTEND OVER THE UNSUPPORTED CLAD AREA. THE ROLLING PROCEDURE IS TO BE APPROVED BY W APD ENGINEERING
8. PRICK PUNCH END PLUGS OF GROUPS 1 & 2 AT 3 EQUALLY SPACED LOCATIONS TO OBTAIN .001/.003 INTERFERENCE FIT WITH TUBE. CAUTION! REMOVE ALL SHARP EDGES IN RAISED METAL CAUSED BY PRICK PUNCHING SO AS NOT TO SCORE I.D. OF TUBE. (TYP. BOTH ENDS)
9. FUEL ROD ASS'YS (GRS. 2 & 4) TO BE CORROSION TESTED AFTER WELDING IN ACCORDANCE WITH W APD APPROVED PROCEDURE. MAX. TUBE & WELD DIA. AFTER PICKLING TO BE .397
10. THIS DIMENSION TO BE ESTABLISHED BY SPRING INSERTION TOOL & IS NOT AN INSPECTION REQUIREMENT.

Figure 220-6 Non-Removable Rods for Main Fuel Assembly



BILL OF MATERIAL						NO. REQ.		
NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPECIFICATION	EQUIVALENT SPECIFICATION	GR 1	GR 2	GR
	1	FUEL TUBE	499B772 IT.6			1	–	
	2	FILLER	498B981 IT.2			AS REQ	–	
	3	END PLUG	675C067 IT.1			1	–	
	4	END PLUG	675C067 IT.3			1	–	
Y	5	PELLET STACK	500B011 GR.3			1	–	
	6	SPRING	500B151 IT.3			1	–	
	7	TUBE	674C868 IT.1			–	1	
	8	SPRING	500B086 IT.1			–	1	
X/Y	9	FUEL COLUMN				–	AS REQ	
	10	FILLER	500B758 IT.2			–	AS REQ	
	11	END PLUG	675C067 IT.4			–	1	
	12	END PLUG	675C067 IT.2			–	1	

"X": FUEL COLUMN TO CONSIST OF A MIXTURE OF URANIUM DIOXIDE & PLUTONIUM DIOXIDE POWDER VIBRATORY COMPACTED IN ACCORDANCE WITH WAPD APPROVED PROCEDURE TO A DENSITY OF 87±1% OF THEORETICAL. POWDER TO BE PREPARED FOR COMPACTION BY DYNAPAK OR EQUIVALENT PROCESS. ENRICHMENT OF UO_2, OXIDE COMPOSITION, PuO_2 CONTENT, AND PARTICLE SIZE & DENSITY FOR POWDER TO BE SPECIFIED ON P.O.

"Y": GAS & VAPOR CONTENT OF FUEL TO BE LIMITED TO FOLLOWING:

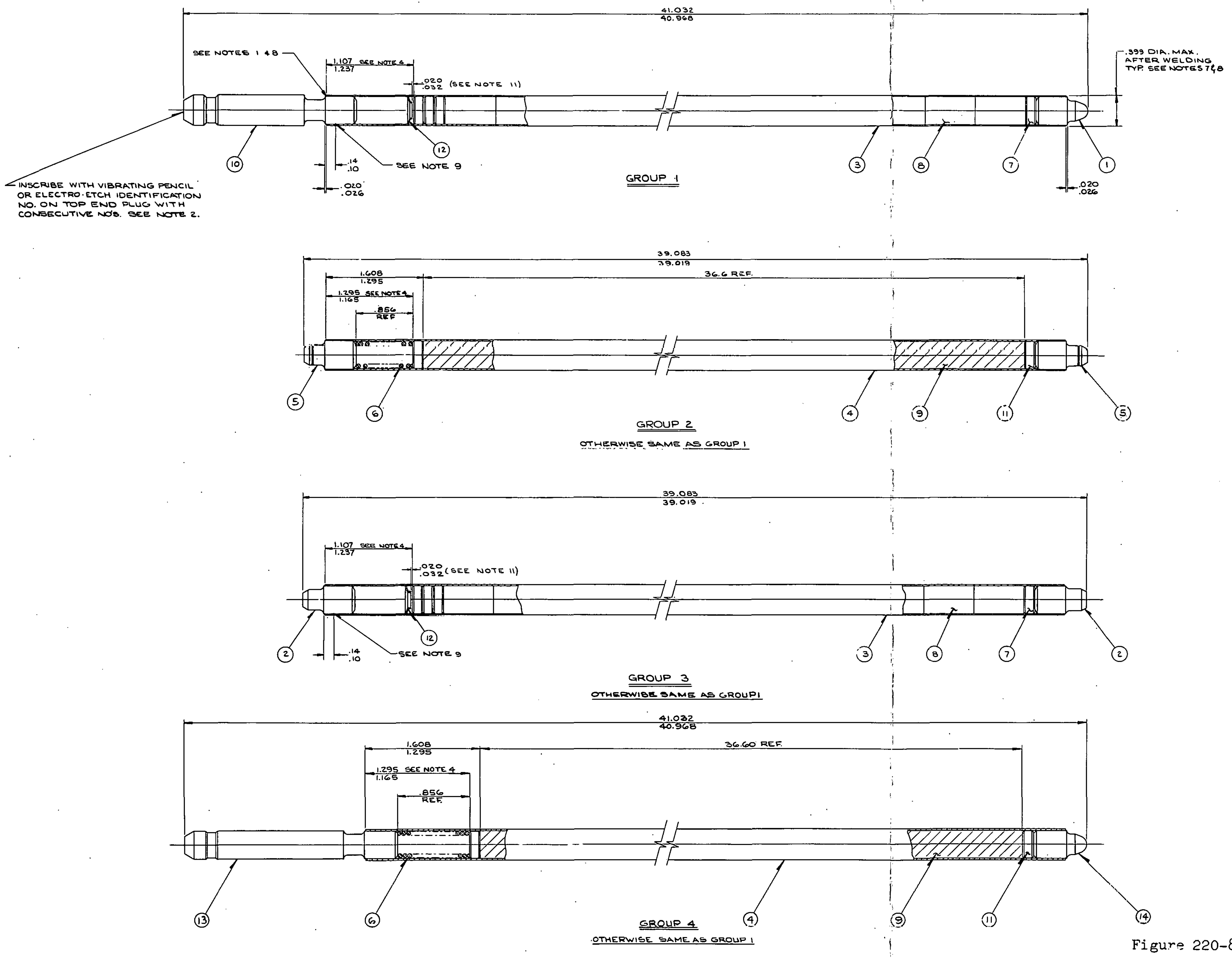
GROUP 1- H_2O - 30 PPM
N_2 - 75 PPM
H_2 - 15 PPM
TOTAL GAS - .05 scc/gm (EXCLUSIVE OF H_2O)

GROUP 2- H_2O - 100 PPM
N_2 - 100 PPM
H_2 - 20 PPM (Si-H & C-H BONDS)
TOTAL GAS - .06 scc/gm (EXCLUSIVE OF H_2O)

NOTES:

1- WELD PROCEDURE & INSPECTION TO BE APPROVED BY WAPD
2- RECORD IDENT. NO., FUEL WT., & METAL WT.
3- MOISTURE CONTENT OF FUELS PRIOR TO WELDING TO BE AS SPECIFIED IN NOTE "Y" TUBE MUST BE DRY.
4- USE ONE FILLER AT BOTTOM OF FUEL TUBE AND ADD FILLERS TO TOP END AS NECESSARY (MIN. OF ONE ADDITIONAL AT TOP END TO OBTAIN THIS DIMENSION.)
5- ROD ASS'Y MUST BE STRAIGHT WITHIN .010 PER FT. BETWEEN END PLUG WELDS WHEN LYING ON A SURFACE PLATE.
6- PRIOR TO ASS'Y, WIPE INSIDE & OUTSIDE SURFACES OF TUBE, SPACER, END PLUGS, & SPRING (USE ALCOHOL ONLY AS A SOLVENT) WITH SWABS TO REMOVE ALL FOREIGN MATTER. WIPE WITH DRY SWAB AFTER CLEANING.
7- END PLUGS TO BE CONCENTRIC WITH TUBE ENDS WITHIN .010 TOTAL AFTER WELDING.
8.- MAX. DIA. PERMITTED AFTER WELDING AS FOLLOWS: .399 DIA (BEFORE PICKLING) ALL SURFACES IN WELD AREA TO HAVE SMOOTH TRANSITION. CAUTION! NO GRINDING OF TUBE WALL PERMITTED ROLLING OF WELDS PERMITTED PROVIDED THE OPERATION DOES NOT EXTEND OVER THE UNSUPPORTED CLAD AREA. THE ROLLING PROCEDURE IS TO BE APPROVED BY WAPD ENGINEERING.
9.- PRICK PUNCH END PLUGS AT 3 EQUALLY SPACED LOCATIONS TO OBTAIN .001/.008 INTERFERENCE FIT WITH TUBE. CAUTION! REMOVE ALL SHARP EDGES IN RAISED METAL CAUSED BY PRICK PUNCHING SO AS NOT TO SCORE I.D. OF TUBE. (TYP. BOTH ENDS)
10.- FUEL ROD ASS'Y TO BE CORROSION TESTED AFTER WELDING IN ACCORDANCE WITH WAPD APPROVED PROCEDURE. MAX. TUBE & WELD DIA. AFTER PICKLING TO BE .397
11.- THIS DIMENSION TO BE ESTABLISHED BY SPRING INSERTION TOOL & IS NOT AN INSPECTION REQUIREMENT.

Figure 220-7 Removable Rods for Main Fuel Assembly



BILL OF MATERIAL						NO. REQ.				
NOTE	ITEM	TITLE	DRAWING & GR. OR IT.	MATERIAL SPECIFICATION	EQUIVALENT SPECIFICATION FOR ® USE ONLY	GR. 1	GR. 2	GR. 3	GR. 4	GR. 5
	1	END PLUG	674C887 IT.4			1	-	-	-	
	2	END PLUG	674C886 IT.3			-	-	2	-	
	3	TUBE	499B772 IT.6			1	-	1	-	
	4	TUBE	674C863 IT.1			-	1	-	1	
	5	END PLUG	674C886 IT.1			-	2	-	-	
	6	SPRING	500B086 IT.1			-	1	-	1	
	7	FILLER	498B981 IT.2			AS REQ	-	AS REQ	-	
Y	8	PELLET STACK	500B071 GR.4			1	-	1	-	
X/Y	9	FUEL COLUMN				-	AS REQ	-	AS REQ	
	10	END PLUG	674C887 IT.3			1	-	-	-	
	11	FILLER	500B158 IT.2			-	AS REQ	-	AS REQ	
	12	SPRING	500B157 IT.2			1	-	1	-	
	13	END PLUG	674C887 IT.1			-	-	-	1	
	14	END PLUG	674C887 IT.2			-	-	-	1	

"X" - FUEL TO CONSIST OF URANIUM DIOXIDE AND PLUTONIUM DIOXIDE POWDER PREPARED FOR COMPACTION BY DYNAPAK OR EQUIV. PROCESS AND VIBRATORY COMPACTED PER WAPD APPROVED PROCEDURE TO A DENSITY OF 87.0 ± 1% OF THEORETICAL. ENRICHMENT OF UO_2 OXIDE COMPOSITION, PUO_2 CONTENT, PARTICLE DENSITY & SIZE FOR POWDER TO BE SPECIFIED ON P.O.

"Y" - GAS & VAPOR CONTENT OF FUEL TO BE LIMITED TO FOLLOWING:

GROUPS 1 & 3 — H_2O - 30 PPM
N_2 — 75 PPM
H_2 - 15 PPM
TOTAL GAS — .05 cc/gm (EXCLUSIVE OF H_2O)

GROUP 2 & 4 — H_2O — 100 PPM
N_2 — 100 PPM
H_2 — 20 PPM (Si-H & C-H BOND)
TOTAL GAS - .06 cc/gm (EXCLUSIVE OF H_2O)

NOTES:

1. WELD PROCEDURE AND INSPECTION TO BE APPROVED BY WAPD.
2. RECORD IDENT. NO., FUEL WT., & METAL WT.
3. MOISTURE CONTENT OF FUELS PRIOR TO WELDING TO BE AS SPECIFIED IN NOTE "Y". TUBE MUST BE DRY.
4. USE ONE FILLER AT BOTTOM OF FUEL TUBE AND ADD FILLERS TO TOP END AS NECESSARY (MIN OF ONE ADDITIONAL FILLER AT TOP END) TO OBTAIN THIS DIMENSION.
5. ROD ASSEMBLY MUST BE STRAIGHT WITHIN .010 PER FT. BETWEEN END PLUG WELDS WHEN LYING ON A FLAT SURFACE AND END PLUGS MUST BE CONCENTRIC WITHIN .003 WITH RESPECT TO END OF TUBE.
6. PRIOR TO ASS'Y., WIPE INSIDE & OUTSIDE SURFACES OF TUBE, SPACER, END PLUGS, & SPRING (USE ALCOHOL ONLY AS A SOLVENT) WITH SWABS TO REMOVE ALL FOREIGN MATTER. WIPE WITH DRY SWAB AFTER CLEANING.
7. ALL SURFACES IN WELD AREA TO HAVE SMOOTH TRANSITION. CAUTION! NO GRINDING OF TUBE WALL PERMITTED. ROLLING OF WELDS PERMITTED PROVIDED THE OPERATION DOES NOT EXTEND OVER THE UNSUPPORTED CLAD AREA. THE ROLLING PROCEDURE IS TO BE APPROVED BY WAPD ENGINEERING.
8. FUEL ROD ASS'YS TO BE CORROSION TESTED AFTER WELDING IN ACCORDANCE WITH WAPD APPROVED PROCEDURE. MAX. TUBE & WELD DIA. AFTER PICKLING TO BE .397.
9. PRICK PUNCH END PLUGS OF GROUPS 1 & 3, AT 3 EQUALLY SPACED LOCATIONS TO OBTAIN .001/.003 INTERFERENCE FIT WITH TUBE. CAUTION! REMOVE ALL SHARP EDGES IN RAISED METAL CAUSED BY PRICK PUNCHING SO AS NOT TO SCORE I.D. OF TUBE. (TYP. BOTH ENDS)
10. THIS DIMENSION TO BE ESTABLISHED BY SPRING INSERTION TOOL & IS NOT AN INSPECTION REQUIREMENT.

Figure 220-8 Fuel Rods for Removable Fuel Assembly

Table 220-2

Plutonium Fuel Rod Data

Fuel Configuration	Pelletized		Vipac	
Clad Type	304 SST	Zircaloy-4	304 SST	Zircaloy-4
Clad Inside Dia., In.	0.361	0.3445	0.361	0.3445
Clad Wall Thickness, In.	0.015	0.0233	0.015	0.0233
Clad Outside Dia., In.	0.391	0.391	0.391	0.391
Pellet Diameter, In.	0.3558	0.3445	-	-
Pellet Length, In.	0.366	0.366	-	-
Diametral Gap, In.	0.0052	0.0071	-	-
Fuel Column Height, In.	36.6	36.6	36.6	36.6
End Gap, In.	0.797	0.797	0.855	0.855
Fuel Density, Percent of Theoretical	94	94	87	87
Fuel Enrichment, Weight Percent PuO_2	6.6	6.6	6.6	6.6

The external configurations of the end plugs are dictated by the location of the fuel rods in the fuel assemblies. The fuel rods shown in Figure 220-6 are the non-removable rods for the main fuel assemblies. These rods were loaded into the various main fuel assembly groups at the lattice locations specified in Figure 220-2.

The fuel rods shown in Figure 220-7 are the removable rods used in the main fuel assemblies. These rods were inserted into the lattice locations in the main fuel assemblies which were left vacant for this purpose. Although two such locations are provided in each of the nine main assemblies, only nine removable rods were manufactured. The remaining vacant lattice locations were utilized in the reactor for two startup neutron source rods and for seven flux wire thimbles.

The fuel rods shown in Figure 220-8 were designed for use in the removable 3 x 3 fuel assembly.

Design Criteria

The Saxton PuO_2-UO_2 fuel rods are designed to permit a peak rod average burnup of 20,000 MWD/tonne with no change in the basic configuration of the Saxton fuel assembly design. The Saxton Core II design fuel assembly enclosures and grids which were used with the UO_2 fuel rods were, therefore, used also as carriers for the PuO_2-UO_2 fuel rods. For consistency with the design of the Saxton Core II UO_2 fuel rods, and to meet the above design objectives, the following criteria were used in the mechanical design of the stainless steel clad PuO_2-UO_2 fuel rods.

1. The fuel rod outside diameter and overall length will conform to those used in the design of the Core II UO_2 fuel rods.

2. The fuel rod cladding will be free standing under reactor design pressure and temperature conditions.

3. Diametral contact between the fuel pellets and cladding will occur only under the worst expected tolerance, power, and burnup combinations.

4. Internal gas pressure in the fuel rods at the end of life will be less than the reactor operating pressure.

5. The fuel rod design must be such that axial movement of the fuel will be restricted during normal handling and shipping loads.

6. The maximum heat rate in the rods will be limited to the present maximum of 16 kw/ft for Saxton.

The design of the Zircaloy clad fuel rods was established using criteria similar to those for the stainless rods with the exception that diametral contact between the pellets and cladding is not limited entirely to the worst tolerance, power, and burnup combinations. Because of the creep properties of Zircaloy-4 at high temperatures, it is expected that some reduction in clad diameter may occur in the hot zone of the fuel rods. However, since this creep will be limited to the high temperature region of the rod and will cease upon contact between the fuel and the clad, it will not affect the integrity of the clad nor will it limit the reactor operation.

Design Details

In the initial design of the Saxton plutonium fuel rods, some consideration was given to reactor operation at 28 MWt. At this power level, the peak heat rate in the hot rod, based on preliminary hot channel factors, was 18.6 kw/ft. Thermal and mechanical analysis of the hot rod at this heat rate showed the Zircaloy cladding and fuel center temperatures to be quite high and the end gaps required for fission gas volume, based on fuel properties data at that time, to be excessive. In view of possible variations in hot channel factors and fuel temperatures, and the lack of data at that time on the high temperature stress corrosion behavior of the Zircaloy cladding, the maximum allowable heat rate for the plutonium fuel rods was maintained at the 16 kw/ft limit established for Saxton Core I. The cladding wall thicknesses and fuel to clad radial gaps were set, therefore, based on this heat rate.

Fuel Cladding

In establishing the cladding dimensions for the plutonium rods, the outside diameter was dictated by the existing Saxton clad size. The design and analytical effort therefore centered on sizing the clad wall thickness and determining the clad stresses.

The governing conditions for sizing the clad thicknesses were the pressure stresses at the beginning of life and the thermal stresses in the clad at the hot spot. These combine to form the severest case at the beginning of life. Since it was a design requirement that contact between the pellets and cladding occur only under the worst conditions, the pellet to cladding diametral gaps were established to just allow contact with no net interference. Thus no clad stresses resulted from differential expansion between the fuel and cladding.

In specifying the clad dimensions, some allowances for ovality in the clad inside diameter and tolerances on the clad wall thickness are necessary. When calculating the pressure stresses in the cladding, it was assumed that the clad was oval within the allowances specified and that the net external pressure then induced bending stresses as well as direct stresses at the clad major and minor axes. Thus, at the beginning of life when the internal pressure in the fuel rod is essentially zero, the net external pressure on the cladding is the full reactor pressure and the induced pressure stresses are maximum. These stresses are compressive and when combined with the thermal stresses at the inner clad wall are the maximum stresses.

During operation at power, internal pressure increases in the fuel rod as a result of fission gas generation and release of water vapor and gases left in the fuel during manufacture. At the end of life, the net pressure difference across the clad thickness is low and the pressure stresses are quite small. As the pressure stresses are reduced, the thermal stresses, which remain essentially constant, will become overriding at the outer clad surface. At the end of life, the stresses on the clad outer surface will be tensile. These stresses are thermal stresses, however, and in no case will the cladding be subject to tensile pressure stresses. The maximum calculated pressure and thermal stresses for the beginning of life conditions are listed in Table 220-3.

Table 220-3

Maximum Clad Stresses at the Beginning of Core Life

Cladding Material	Condition	Avg. Clad Temp, °F	.2% Yield Strength At Temp, °F	Types of Stress	Circumferential Clad Stress Psi: Minor Axis Inside	Circumferential Clad Stress Psi: Minor Axis Outside	Circumferential Clad Stress Psi: Major Axis Inside	Circumferential Clad Stress Psi: Major Axis Outside
Stainless Steel	Avg. Rod	600	65,000	Pressure	-4950	-51,670	-52,800	-3820
				Thermal	-2800	2,800	- 2,800	2800
				Total	-7750	-48,870	-55,600	-1020
	Hot Spot	672	62,000	Pressure	-28,420	-28,200	-31,480	-25,140
				Thermal	- 9,560	9,560	- 9,560	9,560
				Total	-37,980	-18,640	-41,040	-15,580
Zircaloy	Avg. Rod	605	55,000	Pressure	-20,870	-17,260	-21,830	-16,300
				Thermal	- 2,890	2,680	- 2,890	2,680
				Total	-23,760	-14,580	-24,720	-13,620
	Hot Spot	692	51,000	Pressure	-12,330	-25,800	-27,870	-10,260
				Thermal	- 830	770	- 830	770
				Total	-13,160	-25,030	-28,700	- 9,490

Fuel Restraining Spring

In order to minimize the possibility of hot spots occurring in fuel rods as a result of axial gaps in the fuel columns, the current design practice at Westinghouse is to utilize a fuel restraining spring in the fuel rods. The spring and its methods of support are designed to prevent gross movement of the fuel within the cladding during handling and shipping but to allow differential axial expansion between the fuel and cladding during reactor operation.

In the initial stages of the program, a helical coil spring was used as the fuel restraining spring for all plutonium fuel rods. Later in the program, however, the coil spring was deleted from the pelletized rods in order to obtain additional void volume for fission gases without reducing the fuel loading. A Belleville type spring which occupied negligible volume was used in place of the coil spring in these rods and also in the UO_2 pelletized rods made for Core II.

In the vipac rods, the internal voids in the fuel resulting from its lower density were found sufficient to compensate for the volume of the fission gases. Since the volume occupied by the coil spring was not needed, in this case, to accommodate fission gases, the coil springs were retained in the vipac rods.

The Belleville type fuel retaining "spring" was originally developed for use in the Saxton Core II UO_2 fuel rods where the same problem with the end gap volume had occurred. The "springs" fit into the fuel tube with a slight diametral interference and are inserted to within .026 inch of the top of the fuel as shown in Figures 220-6, 220-7, and 220-8.

For normal handling and shipping, the springs will prevent gross movement of the fuel under loads of up to 6 g's (8.3 lb). With the differential expansion between the fuel and cladding during reactor operation, the springs will push through and allow expansion of the fuel.

Although the Belleville type restraining spring was used successfully to produce the 481 Saxton Core II stainless steel clad UO_2 rods, two problems occurred with the use of the springs in the plutonium rods. The first problem was traced to the method used for supporting the fuel rods during the final closure weld. The combination chill block and collet which held the fuel rods for welding radially compressed the rods and springs sufficiently to yield the springs and loosen them. This problem was corrected by reworking the chill block to eliminate grasping the rods in the area of the springs.

The second problem, resulted from the thermal properties of the Zircaloy cladding and the small end gap in the fuel rods. Because of the greater heat required to weld the Zircaloy cladding and the lower specific heat of the material, a temperature increase from welding was propagated at a greater distance from the weld in Zircaloy cladding than in stainless steel. With the relatively short end gap in the plutonium rods, the temperature increase in the Zircaloy cladding in the vicinity of the Belleville springs expanded the cladding sufficiently to loosen the springs. Since the welds on the plutonium rods were made with the rods standing vertical, the springs in some cases slipped down the cladding and bottomed on the fuel. Then, as a result of the combined axial and radial contraction of the cladding during cooldown, the springs buckled or were pushed through. This problem can be eliminated in future fuel rods with this type spring by utilizing either forced convection or liquid coolant and an external heat sink to provide additional cooling capacity for the chill block.

Weld Design

To avoid contact of fine PuO_2 powder with the weld area and contamination of the weld through alloying of weld metal with the plutonium, it was necessary to make the final closure weld on the PuO_2-UO_2 fuel rods with the rods in the vertical position. Because the butt weld design, which was used in the preliminary plutonium fuel rod design, is not suitable for welding in the vertical position, an alternate weld design suggested by the vipac fuel rod vendor (Hanford) was utilized for the fuel rod closure welds.

The alternate weld, which produces a convex fillet joint between the end plug and tubing, had been developed and used by Hanford on EBWR and PRTR fuel rods. Based on the low weld rejection rate experienced by Hanford for this type weld and the convenience and ease of making the weld, it was decided to use the weld for both the vipac and pelletized plutonium fuel rod closures.

In order to maintain maximum strength in the welds, minimum penetrations of 100% of cladding wall thickness and 90% of wall thickness were specified respectively for the stainless steel and Zircaloy closure welds. With these weld penetrations, fuel rod burst tests have shown that failure of the cladding as a result of excessive internal pressure would not occur in the weld zone but instead, in the parent tube material remote from the weld zone.

The above penetration requirements are consistent with standard Westinghouse weld requirements as given in WAPD specifications PS 292712 and CAP 292717-1. Other requirements for the plutonium fuel rod weld and weld inspection were given in the Saxton Plutonium Project Specifications SAX-P-003 and SAX-P-004. The materials design and fabrication of fuel rods are described further in EURAEC-1492 (WCAP-3385-53).

SAXTON CORE II THERMAL AND HYDRAULIC DESIGN

GENERAL DESCRIPTION

The peak spot thermal output for Saxton Core II was set at 16.0 kw/ft, which was the same level as set for the spiked fuel in Core I. Since the peak for the whole core occurs within the center nine fuel assemblies in the core, which contain PuO_2-UO_2 fuel in Core II, 16.0 kw/ft also was the design limit placed on operation of the plutonium fuel rods. Based on this number and the beginning of life hot channel factors, the total initial Core II steady-state power was set at 22.1 MWt. Nuclear calculations predicted a decrease in the hot channel factor with respect to time at power. (The Nuclear Design is covered in EURAEC-1490 [WCAP-3385-51].) Thus, the steady-state core power can be increased to 23.5 MWt after three months full power operation. These two factors, 16.0 kw/ft peak thermal heat rate and the variation in core power with lifetime, comprised the design framework on which the Saxton Plutonium Core II thermal and hydraulic design was based.

HYDRAULIC

The total primary coolant flow rate in the vessel is 2.94×10^6 lb/hr, which is the value measured with Core I instrumentation. An analysis was made of the coolant flow within the core to find the amount available for heat transfer in the active portions of the core. Eighty-five percent, or 2.5×10^6 lb/hr was found to be useful for heat transfer. A calculation made for Core I had determined a core pressure drop of 4.1 psi. However, because a new grid design was used in the new Core II assemblies, it became necessary to calculate a new pressure drop in the assemblies and compare it to that of the old Core I design (three Core I spares were used in Core II). The pressure drops were found to be approximately the same, thus there will not be any flow skewness caused by the different assembly designs. The Core II pressure drop was calculated to be 4.1 psi and the vessel pressure drop 11.3 psi.

HOT CHANNEL FACTORS

The engineering hot channel factors were the same as for Core I design since the vessel flow, geometry, etc. were the same. Thus design $F^E_{\Delta H}$ [1] was 1.22 and F^E_q [1] was 1.045, with the following subfactor breakdown. It should be noted that Core I assembly flow inlet measurements substantiated the 1.07 design subfactor in the engineering $F^E_{\Delta H}$.

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[1] NUCLEONICS, Vol. 20, No. 9, September 1962, "Engineering Hot Channel Factors for Open-Lattice Cores," H. Chelemer, L. S. Tong.

$F^E_{\Delta H}$ = Engineering factor based on enthalpy rise of coolant.

F^E_q = Engineering factor based on fuel rod heat output.

Subfactor	$F^E_{\Delta H}$	F^E_q
1. Pellet diameter, density enrichment, and eccentricity	1.037	1.041
2. Rod diameter, pitch and bowing	1.10	1.004
3. Inlet flow maldistribution	1.07	
4. Flow redistribution	1.05	
5. Flow mixing	0.95	
TOTAL Engineering Factor	1.22	1.045

DNB

Hot channel DNB calculations were made for the steady-state case and the over-power transient case assuming the hot channel to have the geometry of a typical unit cell. Because the nuclear hot channel factor and the core thermal power will change during Core II lifetime (See Nuclear Design, EURAEC-1490), the calculations were made for the worst combination of the two, i.e., that time in core life where the hot channel heat output was maximum. The CAT[2] code was used to supply the hot channel flow redistribution, which in turn was used in the DNB calculations. The W-2[3] correlations were used and the following minimum DNB ratios reported:

Local q" - DNBR at 100% Power, Nominal Conditions = 2.62

Local q" - DNBR at 120% Power, 1800 Psia, Maximum T_{in} = 1.87

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(2) WCAP-2059, "CAT II - An IBM 7090 Code for Predicting Thermal and Hydraulic Transients in an Open-Lattice Core," R. O. Sandberg.

(3) WCAP-1997, "New DNB Correlation," L. S. Tong, H. B. Currin, A. G. Thorp II.

x

The ΔH-DNB ratios at the above conditions were found not applicable, since there was no bulk boiling in the hot channel in either case.

FUEL RODS

The fuel rods in the plutonium assemblies were designed for compatibility with the desired burnup, power and enrichment. Four different types of rods are used in the assemblies and are described in detail in the Mechanical Design section of this report. For the rods containing pelletized fuel, calculations were made to ascertain the pellet dish dimension, the maximum end-of-life fission gas release, the minimum end-of-life hot fission gas space, and the hot spot average fuel temperature and average clad temperature. The latter determined the maximum radial thermal expansion of fuel relative to the clad and was used to set the cold diametral gap between the pellet and the clad. The fission gas release was calculated by FIGHT[4] code which is based on diffusion theory. The amount of fission gas released was combined with the intrinsic amounts of nitrogen and water vapor, the total of which constituted the hot internal fuel rod gases. The fuel pellet specifications permit certain impurities of which nitrogen and water vapor are included. The total amount of hot gases, together with the minimum hot void space and a maximum limit on internal pressure were used to determine the axial end gap in the fuel rod. The design limits ascribed to pellet-clad hot spot contact and to the maximum internal pressure again are defined in the

[4] WCAP-2518, "FIGHT - An IBM 7094 Code for Predicting Fission Gas Release," R. A. Dean, W. A. Jester, E. A. McCabe, Jr.

Mechanical Design section of this report. It was necessary also to make a study of the percentage yield of stable fission products for Pu_{239}. The percentage yield for Pu_{239} was found to be approximately the same as that for U_{235}.

Flux depression factors for the PuO_2 fuel were used in the fuel temperature calculations. The following hot spot fuel centerline temperatures were calculated for the pelletized fuel at steady-state and overpower conditions:

1.	Zircaloy clad, nominal power	3400°F
2.	Zircaloy clad, 120% power	4000°F
3.	Stainless steel clad, nominal power	3400°F
4.	Stainless steel clad, 120% power	4060°F

The vibratory compacted PuO_2 rods were designed to not exceed the same end-of-life rod internal pressure limit as set for the pelletized fueled rods. A thermal conductivity curve for the compacted fuel was developed by the Advanced Materials Group and was used for fuel temperature calculations. Again, FIGHT code was used to determine the maximum fission gas release. Six percent of the initial void volume of the 87 $\pm$ 1% dense compacted fuel was credited as space available for fission and intrinsic gases. Since this void volume is sufficient to contain the gases without exceeding the internal pressure limits, the axial end gap in this case is not determined by internal gas volume requirements. Core loading plans were such that a vibratory compacted rod would not be the highest power density rod. However, peak conditions were assumed to determine maximum centerline fuel temperatures. A maximum steady-state temperature of 4060°F was found and a maximum overpower transient temperature of 4600°F. Again, a flux depression compatible with theoretical density and enrichment was included.

THERMAL AND HYDRAULIC DATA

The following thermal and hydraulic data sheet was prepared for Saxton Core II for both the central Plutonium region and the outside UO_2 region.

HYDRAULIC AND THERMAL DESIGN PARAMETERS

TOTAL CORE

Total Heat Output	22.1 MWt
Total Heat Output	75.4×10^6 Btu/hr
Heat Generated in Fuel	97.4%
System Pressure, Nominal	2000 psig
System Pressure, Minimum Steady-State	1950 psig

COOLANT FLOW

Total Flow Rate	2.94×10^6 lb/hr
Effective Flow Rate for Heat Transfer	2.5×10^6 lb/hr
Flow Area for Heat Transfer Flow (Unit Cells)	2.51 ft^2
Average Velocity Along Fuel Rods	5.8 ft/sec

COOLANT TEMPERATURES

Nominal Inlet	520°F
Maximum Inlet, Including Instrumentation Errors and Deadband	525°F
Average Rise in Vessel	20.7°F
Average Rise in Core	24.8°F
Average in Vessel	530.7°F
Average in Core	532.5°F
Average Film Coefficient	2540 Btu/hr-ft^2-°F
Average Film Temperature Difference	58.0°F

HEAT TRANSFER

Active Heat Transfer Surface Area of Fuel Rods	498 ft^2
Average Heat Flux	147,200 Btu/hr-ft^2
Average Thermal Output	4.4 kw/ft
Maximum Clad Surface Temperature at Nominal Pressure	642°F

PRESSURE DROP

Across Core	4.1 psi
Across Vessel, Including Nozzles	11.3 psi

CENTRAL CORE REGION (UO_2-PuO_2 FUEL)

F_q Heat Flux Hot Channel Factor	3.61
$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor	2.81
Nominal Outlet Temperature of Hot Channel	586.7°F
Maximum Outlet Temperature Hot Channel	591.7°F
Maximum Outlet Enthalpy of Hot Channel	595.8 Btu/lb
Saturation Enthalpy at Minimum Steady-State Pressure	665.9 Btu/lb
Maximum Heat Flux	531,400 Btu/hr-ft^2
Maximum Thermal Output	16.0 kw/ft

DNB RATIOS - CENTRAL CORE REGION

Local q" - DNBR at 100% Power, Nominal Conditions	2.62
Local q" - DNBR at 120% Power, 1800 psia, Max. T_{in}	1.87
ΔH-DNBR at 100% Power, Nominal Conditions	N.A. (1)
ΔH-DNBR at 120% Power, 1800 or 2200 psia, Max. T_{in}	N.A. (1)

OUTER CORE REGION (UO_2 FUEL)

F_q Heat Flux Hot Channel Factor	2.04
$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor	1.59
Nominal Outlet Temperature of Hot Channel	558.9°F
Maximum Outlet Temperature of Hot Channel	563.9°F
Maximum Outlet Enthalpy of Hot Channel	558.9 Btu/lb

Saturation Enthalpy of Hot Channel	665.9 Btu/lb
Maximum Heat Flux	301,600 Btu/hr-ft^2
Maximum Thermal Output	9.05 kw/ft

DNB RATIOS - OUTER CORE REGION

Local q" - DNBR at 100% Power, Nominal Conditions	4.86
Local q" - DNBR at 120% Power, 1800 psia, Max. T_{in}	3.47
ΔH-DNBR at 100% Power, Nominal Conditions	N.A.
ΔH-DNBR at 120% Power, 1800 or 2200 psia, Max. T_{in}	N.A.