CEND-3107-238
Joint U.S. -Euratom
UC-25, Metals, Ceramics
and Materials
TID-4500, 38th Edition

PROGRESS REPORT

BURNABLE POISON ADDITIONS TO UO_2

For Period
January 1 to March 31, 1965

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Contract AT (30-1)-3107
U.S. Atomic Energy Commission

Issued May 1965

Unclassified
Classification

Wm. H. Limon 5/65
Authorized Classifier Date

NUCLEAR DIVISION
COMBUSTION ENGINEERING, INC.
WINDSOR, CONNECTICUT

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BURNABLE POISON ADDITIONS TO UO_2

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FOREWORD

The United States and the European Atomic Energy Community (EURATOM) on May 29 and June 18, 1958, signed an agreement which provides a basis for cooperation in programs for the advancement of the peaceful applications of atomic energy. This agreement, in part, provides for the establishment of a Joint U.S. -Euratom research and development program which is aimed at reactors to be constructed in Europe under the Joint Program.

The work described in this report represents the Joint U.S. -Euratom effort which is in keeping with the spirit of cooperation in contributing to the common good by the sharing of scientific and technical information and minimizing the duplication of effort by the limited pool of technical talent available in Western Europe and the United States.

BURNABLE POISON ADDITIONS TO UO_2

By

R. C. Brayer and W. P. Chernock

ABSTRACT

Further analyses have been performed on a $\text{UO}_2\text{-B}_4\text{C}$ fuel element which was thermally tested under both radial and axial thermal gradients. A new sampling technique was employed which permitted the division of radial samples into smaller volumes of material than was previously used for boron analyses. With this technique, the boron losses, predicted for open systems, was observed, especially in that region of the fuel which was operating above 1350°C . No significant boron migration was detected in either the radial or axial directions in the fuel element, indicating that the boron losses were carried off in the flowing helium stream, verifying earlier results in open systems.

A fabrication process for the irradiation test elements was finalized and four depleted UO_2 fuel elements have been fabricated. These rods have been evaluated and show the process to be satisfactory for the fabrication of fuel rods for irradiation testing in Saxton.

A thorough evaluation has been carried out on all materials, (304 stainless steel tubing, 6% enriched UO_2 , 308L end plugs) obtained from the vendors to ensure that all Saxton specifications for quality were achieved. The fabrication of the enriched fuel elements is currently underway.

The Saxton startup date has been delayed and the insertion of the fuel rods is now scheduled for May 10. Fuel centerline temperature calculations indicated verification of the previous values of $2215\text{--}350^\circ\text{C}$. The internal gas pressure for the 30-day and 90-day irradiation periods are 1052 psi and 1296 psi, respectively. This assumes that all the B^{10} atoms are transformed to helium, all the nitrogen is released from the fuel, and 100% of the fission gas is released during the irradiation periods.

I. SUMMARY

Further evaluations have been performed on samples taken from a $\text{UO}_2\text{-B}_4\text{C}$ fuel element which was thermally tested under radial and axial thermal gradients. The technique for radial sampling was modified to determine if boron was moving within the relatively large samples that were being analyzed. It was found that, as expected, boron was being lost by a vapor phase mechanism in portions of the fuel operating above 1350°C . The higher the operating fuel temperature the more boron was lost due to vaporization. No significant boron migration was detected in either the radial or axial directions in the fuel element. It was concluded that boron loss would occur in open systems where the reaction products are continually being removed; however this mechanism should not operate in a closed system such as an operating fuel element. The successful development of a fabrication process for the irradiation test elements was demonstrated by the production of four depleted $\text{UO}_2\text{-B}_4\text{C}$ fuel elements for evaluation. The densities obtained in these elements were 86.3, 86.4, 86.9 and 86.3% of theoretical. These rods were destructively evaluated prior to fabrication of the enriched fuel elements for Saxton.

Calculations of fuel centerline temperatures by CEND and Westinghouse were in agreement and a temperature of $2215 \pm 350^\circ\text{C}$ is expected. The internal gas pressures calculated for 30 and 90-day irradiation periods were 1052 psi and 1296 psi, respectively.

II. THERMAL GRADIENT STUDIES

A. Introduction

Results ⁽¹⁾ of previous studies on this phase of the program failed to indicate boron migration when an attempt was made to induce such migration by simulating open system conditions in a slowly flowing helium atmosphere. A fuel element containing UO_2 plus 0.05 wt% B_4C was vibratory compacted to 86.8% of theoretical density and was operated for 30 hours at a fuel centerline temperature in excess of 2400°C and a radial to axial thermal gradient ratio of approximately 300. The location of the samples taken for analysis of radial and axial boron migration are shown in Table I and Figure 1. The samples labeled G and K are additional samples which were evaluated during the current period and will be discussed in detail later in this report.

The analytical results, Table I, were surprising in that no discernible radial or axial variation in the boron concentration was found. The variations in the local boron concentrations, shown in the table, are attributed to the accuracy obtainable on the small samples; the average volume of the samples was slightly greater than 0.10 cm^3 .

The lack of boron migration or loss was puzzling since previous work ⁽²⁾ on this program, on identical samples tested in flowing argon at comparable temperatures, indicated significant boron loss. Since no boron migration or loss was found, it was concluded that one of the following must have occurred:

NOTE:
1. NUMBERS INDICATE POSITION OF THERMOCOUPLES ON CLADDING
2. LETTERS INDICATE AREAS WHERE FUEL SAMPLES WERE SECTIONED

Fig. 1. ESTIMATED CENTER TEMPERATURE PROFILE ON THERMAL TEST ELEMENT NUMBER 89 (T.D. = 86.8 %) and RADIAL POSITION OF CHEMICAL SAMPLES

TABLE I

Thermal Gradient Studies
B₄C Distribution (wt%)

Radial Position	Axial Position									
	A	B	A+B Avg	C	D	C+D Avg	E	F	E+F Avg	A+B+E+F Avg
a	0.030	0.032	0.031	0.038	0.052	0.045	0.037	0.034	0.036	0.034
b	0.052	0.098	0.075	0.055	0.046	0.051	0.052	0.031	0.042	0.059
c	0.048	0.040	0.044	0.041	0.036	0.039	0.031	0.052	0.042	0.043
d	0.042	0.041	0.042	0.055	0.055	0.055	0.034	0.050	0.042	0.042
e	0.032	---*	(0.032)	0.038	(0.092)**	(0.065)	0.031	0.052	0.042	0.037
Average for Axial Position			0.045			0.051			0.041	0.043
General Position	Low Temperature			High Temperature			Low Temperature		Low Temperature	

* Samples lost during analysis.

** Value for sample D_e is cumulative value for boron and impurities. Separation was attempted without success.

1. The reaction zone was limited to a volume small enough to permit boron migration within the limits of accuracy of the analytical method. In this case, radial movement of short distances within the specific radial sample width would not be detected. Axial movement would have been detected.

2. Local densification of the fuel in the highest temperature regions of the rod may have provided a local closed system in which the volatile products of the reaction reached their equilibrium pressure resulting in cessation of the reaction.

B. Current Results

Two additional samples were taken and analyzed using a modified radial sampling technique. The technique allowed us to remove a smaller volume of material for analysis; that is, the Zone a sample from the old sampling procedure was further divided into four smaller volumes of material and each individual volume was analyzed for boron. By this technique, we would be able to determine if boron was migrating very short radial distances. A comparison of the two sampling techniques is shown in Figure 2.

Boron loss was observed in the radial samples that operated above 1350°C . The boron loss increased with increasing temperature of the zone. In Zone 1, which operated in the temperature range of 1850 to 2400°C , almost 80% of the boron was lost compared to Zone 3 which operated between 1350 to 1600°C and had a 40% decrease in the boron content. The boron loss versus radial and axial positions is shown in Table II. These results

Fig. 2. OUT OF PILE THERMAL TEST RADIAL SAMPLING METHODS.

TABLE II

THERMAL GRADIENT STUDIES
B₄C DISTRIBUTION (WT%)

Radial Position	Temperature Range	Axial Position		
		G	K	G and K
1	1850 - 2425	0.006	0.010	0.008
2	1600 - 1850	0.017	0.007	0.012
3	1350 - 1600	0.023	0.026	0.025
4	1200 - 1350	0.045	0.034	0.040
5	600 - 1200	0.038	0.038	0.038
6	150 - 600	0.041	0.040	0.041

suggest that the boron is being removed by a vapor phase transport mechanism. This mechanism should only occur in an open system where the reaction products are continuously being removed; in a closed system the reaction should stop when an equilibrium pressure of the species being transported was obtained within the capsule.

The two samples evaluated during this period had overall boron contents that were lower than were found in earlier specimens taken from the same test element. The analyses are being rechecked together with the accuracy implications of +100 ppm boron on the basis of the small volume of the radial samples analyzed. However, the precision of the analysis indicates that boron loss was occurring in the hot portions of the fuel.

III. FABRICATION OF IRRADIATION TEST ELEMENTS

A. Materials Control

During the current reporting period the fuel and cladding materials were received and acceptance tests were performed to establish that all material specifications were met.

1. Stainless Steel Tubing

The 304 stainless steel tubing with 10% cold work was supplied by the Superior Tube Company. Metallographic examination of the as-received tubing revealed appreciable carbide precipitation in the grain boundaries; a photomicrograph of the microstructure is shown in Figure 3. In order to determine the effect of these carbides on the susceptibility for intergranular attack, the material was tested in accordance with ASTM-A-393 "Acidified Copper Sulfate Test for Intergranular Attack in Austenitic Stainless Steel". Cracks were not found in the exposed specimens at 25X, but at higher magnifications (400X) numerous hair line cracks were observed. The combination of carbides in the grain boundaries and the observance of cracks established some doubt concerning the in-reactor performance of this material. Since the Saxton reactor startup had been delayed, and a reorder of the material would not delay the program, new material was purchased. New tubing was purchased and this material passed all of the acceptance tests. The vendor certification was verified; and these results are shown in Table III.

2. UO₂ Evaluation

The 6% enriched fused UO₂ was purchased from the Nuclear Materials and Equipment Corporation, Apollo, Pennsylvania. The enrichment,

Figure 3

304 Stainless Tubing (10% Cold Work) Lot 251094
Carbide Precipitation in Grain Boundaries

TABLE III

WELD DRAWN 304 STAINLESS TUBING

		CEND Specification	Vendor Certification	CEND Verification
I.	<u>Dimensions</u>			
	Inside Diameter (inch)	.552-.533	.5521-.5530	.552-.554
	Wall Thickness (inch)	.0223-.0247	.0228-.0237	.0224-.0230
	Straightness in/ft	.010 max	.001-.007	.003-.006
	Ovality I.D. (inch)	±.0010	.0002-.0005	.0000-.0003
	O.D. (inch)			
II.	<u>Surface Finish</u> (Micro-inch)	<125	O.D. 18-21 I.D. 28-38	.0000-.0002
III.	<u>Mechanical Properties</u>			
a)	Room temperature			
	.2% yield strength (psi)	75,000-100,000	84,100-86,500	
	U.T.S. (psi)	>100,000	109,000-108,700	
	% elongation	> 35.0	41.0-39.0	
b)	Elevated temperature (725°F)			
	.2% yield strength (psi)	50,000-75,000	64,000-64,500	62,100-62,100
	U.T.S. (psi)	>75,000	83,800-84,800	82,000-82,000
	% elongation	> 15.	20 26	21 22

and O/U ratio certified by the vendor was 6.09% and 2.003, respectively, and these values were verified by CEND: (Enrichment $6.12 \pm .2$ and O/U $2.017 \pm .005$). The purity of the UO_2 was within specification for all constituents. As shown in Table IV, as-received particle size distributions were slightly out of specification. However, this material was used and densities in excess of 86% of theoretical were attained.

Visual and metallographic examination of the UO_2 revealed that the particles were free of pores or cracks. Although no second phases or impurities were found in more than 75% of the material examined, a small unidentified phase was seen in some of the powders. Figures 4 and 5 are photomicrographs of the UO_2 showing the second phase in the microstructure. This second phase material is typical of that previously observed in some grades of fused UO_2 . Its presence, in the small quantity noted, will not influence the irradiation test results.

3. Welding Qualifications

Welding parameters for the top and bottom end plugs and the vent hole were established on six, six inch long assemblies. All the components in the assemblies were identical to those being used in the full size elements. Magnesium oxide was used to simulate the fuel and a high purity aluminum oxide disc was substituted for the UO_2 pellet. Figure 6 shows a cross section of a weld evaluation assembly.

Prior to welding, all components were cleaned with acetone and the bottom end plug was press fitted into the six inch length of cladding. The assembly was then placed in a lathe chuck and TIG (Tungsten-inert-gas) welded using an automatic programmer and a variable speed drive mechanism for rotating the fuel tube.

TABLE IV

UO₂ PARTICLE SIZE DISTRIBUTION

Particle	U.S. Mesh Size	CEND Specifications Wt%	Vendor Certification Wt%	CEND Verification Wt%
Group I	6-10	55-60	52.88	42.6
	10-16	40-45	41.93	54.4
	-16		5.19	3.0
Group II	+30		.57	
	30-60	40-60	57.75	53.4
	60-100	40-60	41.30	45.3
	-100		.38	.8
Group III	+200		.48	1.2
	200-325	15	12.01	25.2
	<325	85	87.51	73.5

Figure 4

18X

Figure 5

250X

UO_2 Microstructure Showing an Impurity Phase

Fig. 6. WELD EVALUATION ASSEMBLY

The top end plug was welded in an identical fashion. The element was then placed in a dry box, evacuated, and back-filled with helium; a 1/16 inch diameter 308L filler wire was inserted into the vent hole and fusion welded. The welds were evaluated radiographically and metallographically. All welds having a penetration less than the thickness of the tube wall, cracks, non-metallic inclusions or porosity larger than .010 inch diameter were rejected. In addition, the assemblies were placed in a helium atmosphere for twelve hours at 15 psi pressure and helium leak checked. The maximum acceptable leak rate was 10^{-6} cc/sec.

B. Fabrication Development

The fabrication process for the enriched fuel elements is outlined in Figure 7; the general sequence of operations are listed below:

1. Shell assembly, inspection, and welding of bottom end plug
2. Shell assembly preparation prior to loading
3. Loading and vibratory compaction
4. Hardware insertion and element closure

The tubes were cut to the specified length and inspected for camber, outside diameter and ovality, wall thickness at both ends and surface finish. The inspected plugs were then cleaned and the bottom end plug was welded in place. The end plug welds were inspected by dye penetrant inspection and radiography; the completed shell assembly was then leak checked under a 15 psi helium pressure. The maximum allowable leak rate acceptable was 10^{-6} cc/sec. After inspection a measurement of the volume of the shell was made and the assembly was weighed prior to mounting in the loading fixture.

FIGURE 7

Detailed Fabrication Process For Enriched Fuel Elements

I. Shell Assembly Fabrication

1. Inspect tubing for length, camber, O.D. and ovality every 6 inches at 0° and 90°, I.D. at both ends, end configuration dimensions and surface finish.
2. Clean inside of tube--Alconox wash, water rinse, alcohol rinse, inert gas drying.
3. Weld tubes to bottom end plugs.
4. Radiograph end plug weld (2 exposures at 0° and 90°).
5. Helium leak check shell assembly under 15 psi helium atmosphere (Maximum allowable leak rate 10^{-6} cc/sec).

II. Shell Assembly Preparation Prior to Loading

1. Weigh shell assembly (average of three weighings)
2. Measure shell assembly volume (average of three determinations)
3. Dry shell assembly
4. Mount shell assembly in loading fixture

III. Loading in Glove Box

1. Weigh total amount of UO_2 of each particle size
2. Weigh 30 incremental mixtures of UO_2
3. Weigh 30 incremental charges of B_4C
4. Prepare blend of UO_2-B_4C
5. Spatula load the increment
6. Repeat 4 and 5, 30 times
7. Lower the compaction ram on top of fuel
8. Transfer the element to the compaction area

IV. Compaction

1. Mount the assembly on compaction table
2. Compact for 25 minutes
3. Measure UO_2 height, weigh element
4. Radiograph top of element

V. Top Hardware Insertion

1. Insert UO_2 pellet, stainless spacer, lower marker ring, marker body, top marker ring
2. Radiograph top of element to verify the position of hardware
3. Insert bushing and spring
4. Weld top end plug

An incremental loading technique was used to obtain a boron distribution with $\pm 20\%$ of the nominal composition. Thirty individual loadings of the pre-blended fuel were needed to obtain this distribution. Once the UO_2 -0.050 wt% B_4C mixture was loaded, the ram was set on the top of the fuel and the assembly was transferred to the compaction area.

The fuel column was vibratory compacted on a Model C-10 Vibration Exciter, driven by a Model T-35 Power Amplifier. During compaction the fuel column was axially restrained by a 7 kg (15 lb) spring-loaded ram. An acceleration force of about 45G's, with a frequency continuously oscillating between 200 and 500 cycles per second, was applied for a period of approximately 15 to 25 minutes until the element was compacted to about 85%-88% theoretical density. After compaction, a sintered pellet of depleted UO_2 was placed on the top of the vibratory compacted fuel column. The pellet was used to prevent the possibility of reactions taking place between the UO_2 - B_4C mixture and the stainless steel marker, which is being used to measure axial movement of the fuel during irradiation. The top end plug was press fitted into place and TIG welded; all welds were non-destructively tested by radiography and dye penetrant. A cross section of the fuel element is shown in Figure 8: the dimensions and densities of the completed rods are given in Table V. The completed fuel element undergoes a complete non-destructive evaluation which includes; dimensional measurements, radiography, dye penetrant and helium leak check prior to shipping. A total of four fuel elements were fabricated using the described fabrication procedures.

The densities obtained on the compacted fuel elements were 86.3, 86.4, 86.9 and 86.3% of the theoretical. One rod will be evaluated at

TABLE V
DIMENSION OF COMPLETED FUEL ELEMENTS

Identification		A-2	B-1	B-2	B-3
Rod length (inches)		40.958	40.980	40.985	40.995
UO ₂ Density (%TD)		86.3	86.4	86.9	86.3
Outside Diameter (inches)					
A. Top	0°	.5992	.5989	.5989	.5985
	90°	.5985	.5990	.5990	.5988
B. Center	0°	.5985	.5990	.5990	.5983
	90°	.5988	.5990	.5988	.5985
C. Bottom	0°	.5988	.5989	.5990	.5982
	90°	.5985	.5988	.5990	.5986

The wall thickness on the rods measured 0.0234 to 0.0240 inches.

Fig.8 . FUEL ROD DESIGN

Westinghouse, APD and two rods will be destructively evaluated at CEND. The fourth rod will be kept as a backup in case further studies are required.

The destructive evaluation of the fuel elements at CEND will include the determination of the moisture content in the fuel and the axial and radial boron distribution. Samples will be taken along the length of the fuel element and from the plenum. The sampling of the element will be done in accordance with Figure 9. Six axial and two radial specimens will be analyzed using the techniques described under Section A of this report.

Fig. 9. POSITION OF SAMPLES FOR BORON ANALYSIS

IV. IRRADIATION

A. Introduction

Startup of the Saxton Reactor has been delayed and is presently scheduled for May 10. The burnable poison fuel elements are scheduled for insertion into the reactor during the month of April. The enrichment for the fuel elements has been fixed at 6% and was based on a fuel centerline temperature of $2200 \pm 350^{\circ}\text{C}$ and a peak power rating of 17.8 kw/ft. The fuel elements will be irradiated to burnups of 2000 and 5000 MWD/MTU.

The safeguards report for the experiment has been completed for review by the Saxton Safety Committee.

B. Current Results

1. Fission Gas Release Calculations

Calculations of the fuel centerline temperature have been made at CEND and Westinghouse APED; the values obtained were in agreement ($2200 \pm 350^{\circ}\text{C}$) and are presented in detail in Appendix I.

The fuel centerline temperature is given by the relationship

$$\int_{T_s}^{T_c} K dT = \frac{Q}{4\pi} \frac{I_0(Ka) - 1}{0.5(Ka)I_1(Ka)}$$

where

Q is the linear heat rating in watts/cm

K is the thermal conductivity in watts/cm- $^{\circ}\text{C}$

and

$$\left[\frac{I_0(Ka) - 1}{0.5(Ka)I_1(Ka)} \right]$$

is the neutron flux depression factor.

The variation of thermal conductivity of vibratory compacted uranium dioxide, 88% of theoretical density as a function of temperature, is given in Figure 1 of Appendix I along with the corresponding values of $\int_0^T k dT$. The curve is based on Westinghouse irradiation tests in GETR.

The Wigner "rational approximation" was employed for calculating the flux depression factor for 95% theoretical dense UO_2 and modified on the basis of Saxton cross-sections for UO_2 of 88% theoretical density. In the calculation of flux depression, the addition of 500 ppm of natural boron carbide to the fuel is equivalent, on the basis of absorption cross-section, to an increase in enrichment of 0.12 wt.%. The variation of the flux depression factor with fuel density is shown in Figure 2 of Appendix I.

It should be noted that the parameters used in the derivation of the fuel center temperature lack precision. The thermal conductivity of vibratory compacted UO_2 is derived from a limited number of results, the reported values of the contact conductance vary over a wide range and calculations of specific power includes a factor of 1.15 on the nominal value to allow for nuclear uncertainties and the engineering hot channel factor. Also, the possible influence of gas dilution in the colder unsintered regions of the fuel on thermal conductivity was not considered in the calculation.

2. Internal Gas Pressures

Calculations (Appendix II) have been performed on the magnitude of the internal pressures that will be generated in a fuel element due to fission gas release and helium. Helium will be generated by the reaction $B^{10} + n \rightarrow He + Li^7$. Nitrogen pressure was also taken into consideration since the fuel contained 100 ppm of nitrogen that could be liberated during irradiation. The following factors were assumed:

- 1) 100% fission gas release
- 2) 100% nitrogen release from the fuel
- 3) All B^{10} atoms are transferred to helium

For irradiation periods of 30 and 90 days the total gas released (cc at STP) from the fuel was 2.7788 and 3.424 cc, respectively. This results in total pressures of 1052 psi for the 30-day irradiation and 1296 psi for the 90-day irradiation. These pressures are low and should not affect the performance of the fuel element.

3. Spectrographic Determination of Boron in Uranium Dioxide

Since September 1963, Westinghouse has determined boron in UO_2 by optical emission spectrography in the range of 10 to 1,000 parts per million. The precision and accuracy are estimated to be 5% to 8% relative for homogeneous materials. Experience has shown that when the particle size of the added boron compound exceeds 80 microns, the precision approaches 30% relative.

The representative sample selected for analysis is crushed in a tool steel Plattner type mortar to less than 60 mesh and oxidized in a platinum crucible at $500^{\circ}C$ for one hour. A soft uranium oxide is obtained which is blended in a plastic vial mechanically. Five hundred (500) milligrams of this oxide are blended with 250 milligrams of less than 100 mesh graphite powder. Fifty (50) milligrams of this blend are tamped into a preformed graphite electrode (UCP #7175). Three electrodes are loaded per sample.

The samples are excited in a direct current arc for 15 seconds at 4.5 amperes followed by 105 seconds at 9.0 amperes. The sample to graphite ratio, the charge size, the geometry of the electrode and the

programmed excitation are so related that all the boron of whatever compound and all the uranium is converted to a carbide in a melt as the walls of the electrode crater are consumed.

Spectra of samples and standards are recorded on Eastman SA-1 plates with a 3.4 meter Ebert JACO spectrograph in the second order with a 10 micron slit. The reciprocal linear dispersion is 2.59 Å per mm. The following boron and uranium lines are measured on a microdensitometer to give a relative line intensity ratio from an emulsion calibration curve.

<u>Analytical Time Pair</u>	<u>Range in ppm B</u>
B 2496.78 U 2498.83	40 to 400
B 2497.73 U 2498.83	10 to 200
B 2497.73 U 2500.86	100 to 1,000

The boron concentration is determined from an analytical curve relating log intensity ratio to log concentration. Standards are prepared by blending and solid dilution of an analyzed boron compound into a spectrographically pure U_3O_8 base material. The standards are prepared in the following concentrations: Blends, 10, 20, 40, 60, 100, 200, 400, 600 and 1000 ppm.

Specimens from CEND have been cross-checked for boron analysis using the described technique. Two UO_2-B_4C mixtures containing 390 ppm boron were prepared in an identical manner from a single master mix of UO_2-B_4C containing 1560 ppm boron. One mixture was analyzed at Combustion and one at Westinghouse. Westinghouse made five determinations on three samples representing 63% of the mixture. Combustion analyzed the whole

mixture in three separate samples making six determinations on each sample. The results are listed below and are in excellent agreement.

WESTINGHOUSE RESULTS

Sample Identification	Sample Weight (g)	Boron Content (ppm)
A	8	306 $\pm$ 20
B	8	475 $\pm$ 30
C	4	415 $\pm$ 30
Weighted Average	20	395 $\pm$ 26

COMBUSTION RESULTS

Sample Identification	Boron Content (mg)	Boron Content (ppm)
C ₁ Average	1.40 $\pm$.40	
C ₂	4.55 $\pm$.65	
C ₃	6.43 $\pm$.30	
Total Samples	12.38 $\pm$ 2.35	387 $\pm$.73

V. FUTURE WORK

During the next reporting period four enriched fuel elements will be fabricated using the procedures established on the four dummy elements that were described in this report. The completed elements will be shipped to Westinghouse, APD and of these, two will be inserted in Saxton. The third rod will be used for evaluation studies. The fourth rod, which will be retained at CEND, will be kept as a backup for evaluation.

APPENDIX I

Fuel Center Temperatures

At the request of Combustion Engineering Corporation, Nuclear Power Division Staff, the procedure for calculation of the fuel center temperature has been re-examined and is here presented.

The fuel center temperatures for the vibratory compacted fuel rods depend on the power rating and thermal conductivity of the fuel. The methods used in the calculation of center temperatures of pelleted fuel are employed with modifications to the appropriate parameters.

The fuel center temperature is given by the relationship

$$\int_{T_s}^{T_c} k dT = \frac{Q}{4\pi} \left[\frac{I_0(ka) - 1}{0.5(ka)I_1(ka)} \right]$$

Where Q is the linear heat rating, watts/cm

K is the thermal conductivity of the fuel, watts/cm-°C

$$\left[\frac{I_0(ka) - 1}{0.5(ka)I_1(ka)} \right] \text{ is the flux depression factor.}$$

The variation of thermal conductivity of vibratory compacted uranium dioxide, 88% theoretical density, as a function of temperature is given in Figure 1 and is based on Westinghouse irradiation tests in GETR. The corresponding values of $\int_0^T k dt$ are given also in Figure 1.

The Wigner "rational approximation" was employed for calculating the flux depression factor for 95% theoretically dense uranium dioxide and modified on the basis of Saxton cross-sections for uranium dioxide of 88% theoretical density.

FIGURE 1 THERMAL CONDUCTITIVITY AND $\int k dT$ AS A FUNCTION OF TEMPERATURE FOR VIBRATORY COMPACTED UO_2 (88% T.D.)

To a first approximation, in the calculation of flux depression the addition of 500 ppm of natural boron carbide to the fuel is equivalent, on the basis of absorption cross-section, to an increase in enrichment of 0.12 w/o. The variation of the flux depression factor with fuel density is shown in Figure 2.

The maximum clad surface temperature is 339°C

Temperature drop across the clad is 35.5°C

Clad inner surface temperature is 374.5°C

Reported values^(1,2,3,4) of the contact conductance range from 875 to 2100 BTU/hr-ft²-°F and yield an average value of 1025 BTU/hr-ft²-°F at zero contact pressure. This value of contact conductance is used for calculation of the temperature drop between fuel and clad.

$$\Delta T_{\text{contact}} = \frac{4.212 \times 10^5}{1025} = 411^{\circ}\text{F}$$

Consequently, T_s , the fuel surface temperature is $602.8^{\circ}\text{C} \approx 600^{\circ}\text{C}$

The fuel center temperature, T_c , is given by

$$\int_{600}^{T_c} k dT = \frac{17.8 \times 10^3}{30.48 \times \pi} \times 0.857 = 39.8 \text{ watt/cm}$$

$$\int_0^{600} k dT = 12 \text{ watt/cm}$$

$$\int_0^{T_c} k dT = 51.8 \text{ watt/cm}$$

$$\text{and } T_c = 2215^{\circ}\text{C}$$

FIGURE 2. VARIATION OF FLUX DEPRESSION FACTOR WITH FUEL DENSITY.

It should be noted that the parameters used in the derivation of the fuel center temperature lack precision. The thermal conductivity of vibratory compacted uranium dioxide is derived from a limited number of results; the reported values of the contact conductance vary over a wide range and calculation of specific power includes a factor of 1.15 on the nominal value to allow for nuclear uncertainties and the engineering hot channel factor. These uncertainties result in a variation of $\pm 315^{\circ}\text{C}$ in the fuel center temperature.

The reported fuel center temperature of $2215 \pm 350^{\circ}\text{C}$ for a specific rod power of 17.8 kw/ft is in agreement with the value reported in WCAP-3107-1.

APPENDIX II

During irradiation the inert fission gases, xenon and krypton, will be generated. Helium will be generated by the reaction $B^{10} + n \rightarrow He + Li^7$. The fuel contains an assumed 100 ppm of nitrogen which may be liberated during irradiation.

For a rod power of 17.8 kw/ft, the volumetric fission rate is

$$\begin{aligned} & 1.333 \times 10^{14} \times 17.8/182 \text{ fissions/cm}^3\text{-sec.} \\ & = 1.304 \times 10^{13} \text{ fissions/cm}^3\text{-sec.} \end{aligned}$$

The generation rate of xenon is $1.304 \times 10^{11} \times 21.83$

$$= 2.846 \times 10^{12} \text{ atoms/cm}^3\text{-sec.}$$

The generation rate of krypton is $1.304 \times 10^{11} \times 3.857$

$$= 5.029 \times 10^{11} \text{ atoms/cm}^3\text{-sec.}$$

The corresponding volumetric generation rates at S.T.P. are

$$1.058 \times 10^{-7} \text{ cc (STP)/cm}^3\text{-sec. and}$$

$$1.87 \times 10^{-8} \text{ cc (STP)/cm}^3\text{-sec.}$$

The total fission gas volumetric generation rate is

$$1.245 \times 10^{-7} \text{ cc (STP)/cm}^3\text{-sec.}$$

Assume that all the B^{10} atoms are transformed to helium, and 500 ppm of natural yields $1.685 \text{ cc (STP)/cm}^3$ of fuel.

Assume that all the nitrogen is released from the fuel, 100 ppm of nitrogen yields $0.7711 \text{ cc (STP)/cm}^3$ of fuel.

Assume 100% fission gas release and irradiation periods of 30 days (2.592×10^6 secs) and 90 days (7.776×10^6 secs). The total gas released (cc at STP) cc of fuel will be:

	<u>$t = 2.592 \times 10^6$ secs.</u>	<u>$t = 7.776 \times 10^6$ secs.</u>	
Fission Gas	0.3227	0.9681	100% release
Helium	1.6850	1.6850	All boron transformed
Nitrogen	<u>0.7711</u>	<u>0.7711</u>	100% release
	2.7788	3.4242	

To estimate the internal pressure, assume a gas temperature equal to the mean temperature of the fuel of 1407°C .

For the fuel element, the volume of uranium dioxide is 114.3 cc.

The free volume in the fuel is 15.59 cc.

The volume of the plenum is 11.73 cc.

Total free volume is 27.32 cc.

$$V_{t1}(\text{STP}) = 2.7788 \times 114.3 \text{ cc (STP)} = 317.6 \text{ cc (STP)}$$

$$V_{t2}(\text{STP}) = 3.4242 \times 11.3 \text{ cc (STP)} = 391.4 \text{ cc (STP)}$$

$$P_{t1} = \frac{14.7 \times 317.6 \times 1680}{273 \times 27.32} = \frac{7.843 \times 10^6}{7.458 \times 10^2} = 1052 \text{ psi}$$

$$P_{t2} = \frac{14.7 \times 391.4 \times 1680}{273 \times 27.32} = \frac{9.666 \times 10^6}{7.458 \times 10^2} = 1296 \text{ psi}$$

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