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ISSUE SUMMARY

TITLE: TEST REPORT: NONDESTRUCTIVE EXAMINATION OF FSV FUEL TEST ELEMENT FTE-1

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☒ DV & S
☐ DESIGN

APPROVAL LEVEL 5

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RELEASED

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NEXT INDENTURED DOCUMENTS

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1. INTRODUCTION

Fuel test element FTE-1 is one of eight fuel test elements (Ref. 1) inserted into the FSV reactor during the first refueling. The element was irradiated in core location 25.07.F.06 (region 25, column 7, core layer 6) for 189 effective full power days (EFPD).* As shown below, there were a number of major differences between FTE-1 and regular FSV fuel elements.

	FTE-1	Regular FSV Fuel Elements
Graphite (grade)	H-451	H-327
Fuel	UC_2 TRISO/ThO_2 TRISO	$(Th,U)C_2$ TRISO/ThC_2 TRISO
Curing process (fuel rods)	Cure-in-place	Cure-in-bed

A nondestructive examination of FTE-1 was performed in the Hot Service Facility (HSF) at FSV on April 16, 1982. The examination included:

- A visual inspection for corrosion, cracks, scratches, and other abnormalities.
- Dimensional measurements (with the metrology robot)
  - Across-flats dimensions
  - Element length
  - Coolant hole diameters
  - Distance between coolant holes
  - Distance between fiducial holes
  - Bow
- Gamma dose rate and neutron count rate measurements.

*Cycle 2 was from May 26, 1979 to May 13, 1982.

The objectives of the examination were to:

- Verify the structural integrity and dimensional stability of the H-451 graphite block.
- Obtain H-451 graphite dimensional change and bow data for comparison with HTGR design code strain and bow calculations.
- Obtain a gamma dose rate and a neutron count rate for comparison with HTGR computer code calculations.

2. IRRADIATION HISTORY

The following LHTGR design codes were used to simulate the irradiation history for FTE-1.

GAUGE (Ref. 2) - used to calculate radial power and flux distributions for FSV cycle 2.

GATT (Ref. 3) - used to calculate the power history for FTE-1.

SURVEY (Ref. 4) - used to calculate the temperature and fast neutron fluence histories for FTE-1 at 35 local points (Fig. 1).* The results from GATT indicated that during cycle 2, FTE-1 produced approximately 8% more power than would have been produced by the segment 2 element that

*The original thermal analysis for FTE-1 (Ref. 1) was performed with the TREVER code. The TREVER and SURVEY codes utilize the same thermal model and fast neutron fluence calculation. Benchmark calculations have been performed which verify that, given the same input data, the two codes calculate the same temperatures and fast neutron fluences.

it replaced.* This was modeled by adjusting the appropriate axial power profiles used in SURVEY. The axial power factors for the axial locations occupied by FTE-1 were increased by 8% and the axial power profiles were renormalized. The radial power and flux distributions for core region 25 were obtained from the GAUGE analysis of cycle 2. The perturbations in the region and column average powers caused by the insertion of FTE-1 into region 25 were small and were not considered in the SURVEY analysis.

SURVEY/STRESS (Ref. 5) - used to calculate stresses, strains, and bow for FTE-1. These calculations were based on the irradiation conditions obtained from SURVEY.

The time-averaged graphite temperatures and fast neutron fluences calculated for FTE-1 are given in Tables 1 and 2. Section 4.2 discusses the results of the SURVEY/STRESS calculations.

3. TEST METHODS

In addition to FTE-1, 53 fuel and reflector elements from FSV core segment 2 were examined in the HSF at FSV in April 1982. The same test methods were employed for the examination of FTE-1 and for the examinations of the other 53 elements. Ref. 6 describes these methods.

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*According to the GATT results, the ratio of the power generated by FTE-1 and the power that would have been generated by the element it replaced, (or the axial power correction factor, as it is called in Ref. 1), decreased from about 1.09 at the beginning of cycle 2 to about 1.07 at the end of the cycle. The axial power correction factor originally reported for FTE-1 in Ref. 1 decreased from 1.30 to 1.10 during cycle 2. This correction factor was calculated with the FEVER code. As stated in Ref 1, the FEVER calculations overestimated the power perturbation effect of the test elements because of the boundary conditions assumed by the code. Because FEVER is a one-dimensional code, it must assume that the column of fuel elements being evaluated is surrounded by an infinite array of identical columns of elements. Consequently, the neutron fluxes calculated by FEVER for FTE-1 were too high since FTE-1 had a heavier fuel loading than the surrounding partially depleted fuel elements. Because it is a three-dimensional code, GATT can accurately calculate the boundary conditions, (and therefore the neutron fluxes and fission rates), for the test elements.

4. TEST RESULTS

4.1 Visual Examination

FTE-1 was in excellent condition. No cracks were observed on any of the element's surfaces; nor was there any evidence of graphite corrosion or other significant structural damage.

Figures 2 through 7 show the side faces of FTE-1. Face C had a discontinuous vertical scratch running nearly the length of the element (Fig. 4). Similar scratches were observed on face C of several other elements and are thought to have resulted during handling, possibly from contact between the elements and a storage rack in the fuel handling machine. There was also a short vertical scratch near the upper left corner of face F (Fig. 7).

Fig. 8 shows the top surface of FTE-1. Numerous dark markings (Fig. 9) were observed on this surface. Many of these were on top of, or around, fuel hole plugs and appeared to be build-ups of some substance. The metrology robot was used to demonstrate that this was not the case. The darkened areas were found to be level with adjacent non-discolored areas. Dark markings were also observed between the burnable poison holes and the adjacent coolant holes. The markings on the top surface of FTE-1 are believed to have been stains caused by outgassing from the graphite cement used to cement the fuel hole plugs in place*. These types of markings were not observed on any of the other examined fuel elements.

No unusual features were observed on the bottom surface of FTE-1.

4.2 Dimensional Measurements

FTE-1 underwent little dimensional change as a result of irradiation. The element-average axial strain ($\Delta\ell/\ell$) was -0.038%, corresponding to a length reduction of 0.31 mm. The element-average radial strain was -0.012%,

*Unlike the regular FSV fuel elements, the fuel hole plugs were cemented in place after the element was heat-cured corresponding to a shrinkage of 0.04 mm across-flats. The maximum bow was 0.08 mm.

The axial strain was highest, -0.051%, adjacent to face B and lowest, -0.022%, adjacent to face E. Fig. 10 shows the axial strain distribution as determined from element length measurements. The distance-between-fiducial-hole measurements (Table 3) indicate that the axial strain was greater at the top of the block than at the bottom. The block-average axial strains between the top two, middle two, and bottom two fiducial holes were -0.065%, -0.033%, and 0.009%, respectively.

As shown in Table 4, the radial strain, (as determined from the distance-across-flats measurements), also decreased from the top to the bottom of the block (-0.032% to 0.004%). The average radial strain determined from the coolant hole diameter measurements at the top of the block (Table 5) was -0.220%, but is suspect because of the small dimensions involved. A bias of only 0.03mm in the coolant hole diameter measurements would account for the discrepancy between the radial strains determined from the distance-across-flats and coolant hole diameter measurements. The average radial strain determined from the distance-between-coolant-hole measurements at the top of the block (Table 6) was 0.028%.

Table 7 compares the strains and bow calculated by SURVEY/STRESS with the corresponding measurements. The strains were consistently overpredicted, but the absolute differences between the calculated and measured values were less than 0.100%. Although the strains were somewhat overpredicted, the strain differences within the element were well predicted. The calculated and measured top-to-bottom differences in the radial strain were 0.047% and 0.036%, respectively. The calculated and measured maximum across-flats differences in the axial strain were 0.043% and 0.022%, respectively. The calculated bow was 0.10 mm and the measured bow was 0.08 mm.

4.3 Gamma Dose Rate

At 3 feet, the gamma dose rate measured for FTE-1 was 338 R/hr. The PATH code (Ref. 7) was used to calculate the gamma dose rate for FTE-1. At 3 feet, the calculated gamma dose rate was 354 R/hr.

4.4 Neutron Count Rate

A SNOOPY-type detector system equipped with a Reuter-Stokes Model RS-P6-0305-134, fission counter was used to obtain a neutron count rate from FTE-1 for comparison with MORSE code (Ref. 8) calculations. The measured count rate was 0.00112 counts/sec. (15 counts over 224 minutes). The calculated count rate was 0.00104 counts/sec.

5. CONCLUSIONS

5.1 The structural performance and dimensional stability of FTE-1 were excellent. No cracks were observed in any of the element's surfaces. There was no evidence of graphite corrosion or other structural damage. The element underwent little dimensional change as a result of irradiation. The element-average axial strain was -0.038%, corresponding to a length reduction of 0.31 mm. The element-average radial strain was -0.012%, corresponding to a shrinkage of 0.04 mm. across flats. The maximum bow was only 0.08 mm. However, it should be noted that the fast neutron fluence for FTE-1 ($\sim 0.65 \times 10^{25}$ n/m^2, E > 29fJ$_{HTGR}$) is far less than the maximum fast fluences that will be experienced by FSV fuel elements ($\sim 8 \times 10^{25}$ n/m^2) and LHTGR fuel elements ($\sim 6 \times 10^{25}$ n/m^2).

5.2 FTE-1 shrank less than expected (based on SURVEY/STRESS dimensional change calculations). However, the absolute differences between measured and calculated strains were less than 0.100%. The measured bow (0.08 mm) and calculated bow (0.10 mm) were approximately equal.

5.3 Excellent agreement was obtained between measured and calculated gamma dose rates (358 R/hr. vs. 354 R/hr. at 3 feet), and between measured and calculated neutron count rates (0.00112 counts/sec. vs. 0.00104 counts/sec.) for FTE-1. The gamma dose rate was calculated with the PATH code. The neutron count rate was calculated with the MORSE code.

Verification of HTGR design code calculations cannot be accomplished through comparisons of calculations and experimental observations for one element. Many such comparisons for core components which have collectively experienced a wide range of irradiation conditions are required. The results of the comparisons between measurements and design code calculations (SURVEY/STRESS, PATH, MORSE) for FTE-1 should be reviewed with this in mind. Additional comparisons between measurements and calculations for FSV fuel elements are provided in Ref. 6.

6. REFERENCES

1. "Safety Analysis Report for Fort St. Vrain Reload 1 Test Elements FTE-1 Through FTE-8," General Atomic Company Report GLP-5494, June 30, 1977.
2. Wagner, M.R., "GAUGE, A Two-Dimensional Few Group Neutron Diffusion-Depletion Program for a Uniform Triangular Mesh," USAEC Report GA-8307, General Atomic Company, March 15, 1968.
3. Kraetsch, H., and M. R. Wagner, "GATT, A Three-Dimensional Few Group Neutron Diffusion Theory Program for a Hexagonal Z Mesh," USAEC Report GA-8547, General Atomic Company, January 1, 1969.
4. Georghiou, D. L., "SURVEY, A Computer Code for the Thermal and Fuel Performance Analysis of High-Temperature Gas-Cooled Reactors," General Atomic Report GA-D14869, November 1978.
5. Smith, P. D., "SURVEY/STRESS, A Model to Calculate Irradiation - Induced Stresses, Strains, and Deformations in an HTGR Fuel Block Using Viscoelastic Beam Theory," General Atomic Report GA-A13712, October 20, 1975.
6. Saurwein, J. J., "Nondestructive Examination of 54 Fuel and Reflector Elements from Fort St. Vrain Core Segment 2," DOE Report GA-A16829, General Atomic Company, to be published.
7. Clark, S. and B. A. Engholm, "PATH - A Highly Flexible General Purpose Gamma Shielding Program," General Atomic Report GA-9908, December 10, 1969.
8. Straher, E., et al., "The MORSE Code - A Multigroup Neutron and Gamma Ray Monte Carlo Transport Code," Oak Ridge National Laboratory Report ORNL-4585, September 1970.

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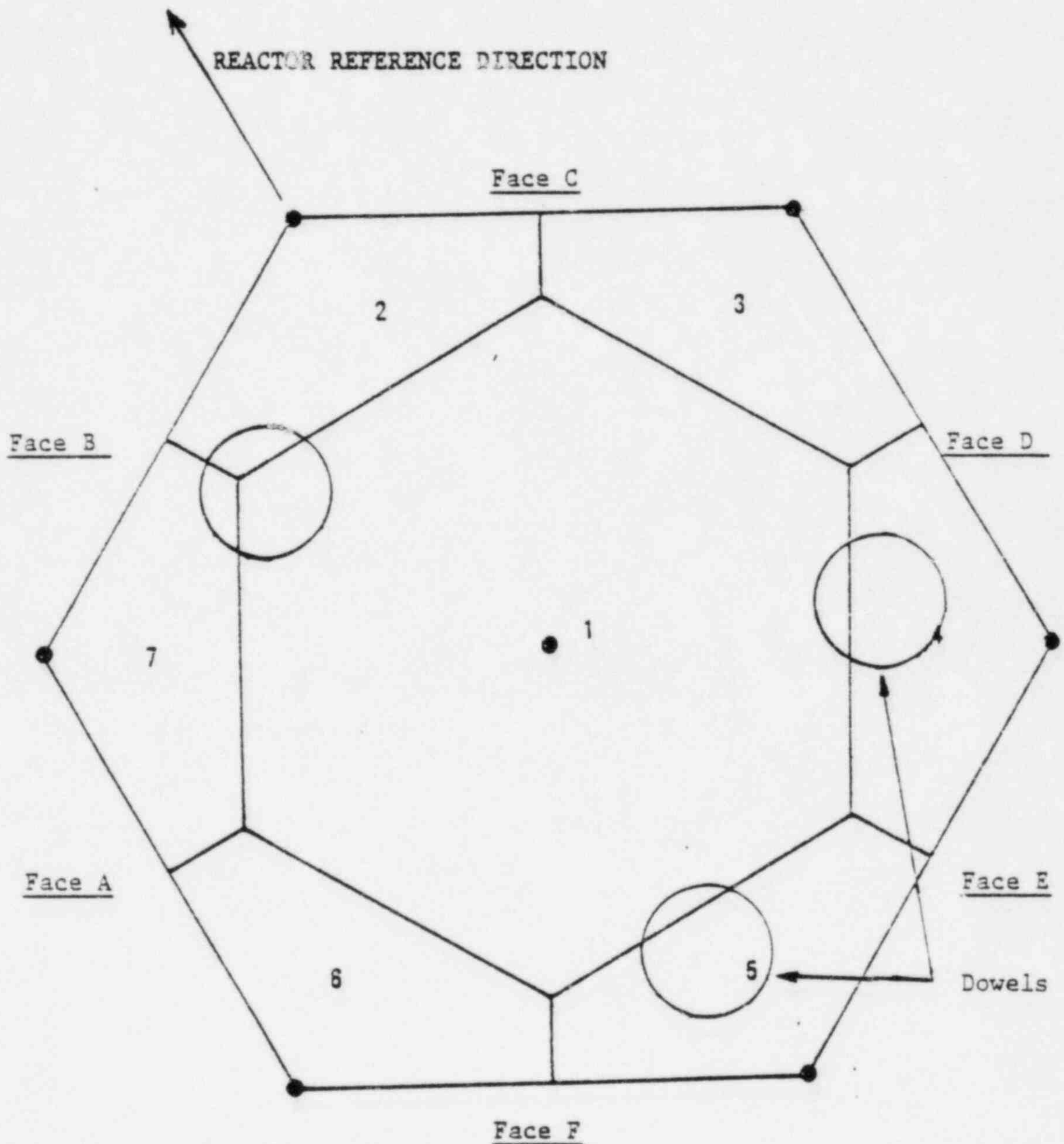

SURVEY code calculations were performed for 7 local points at 5 axial positions (35 points). The axial positions were at the top and bottom of the block and at 1/4 block intervals.

Fig. 1. Local point numbering - SURVEY code analysis of FSV FTE-1

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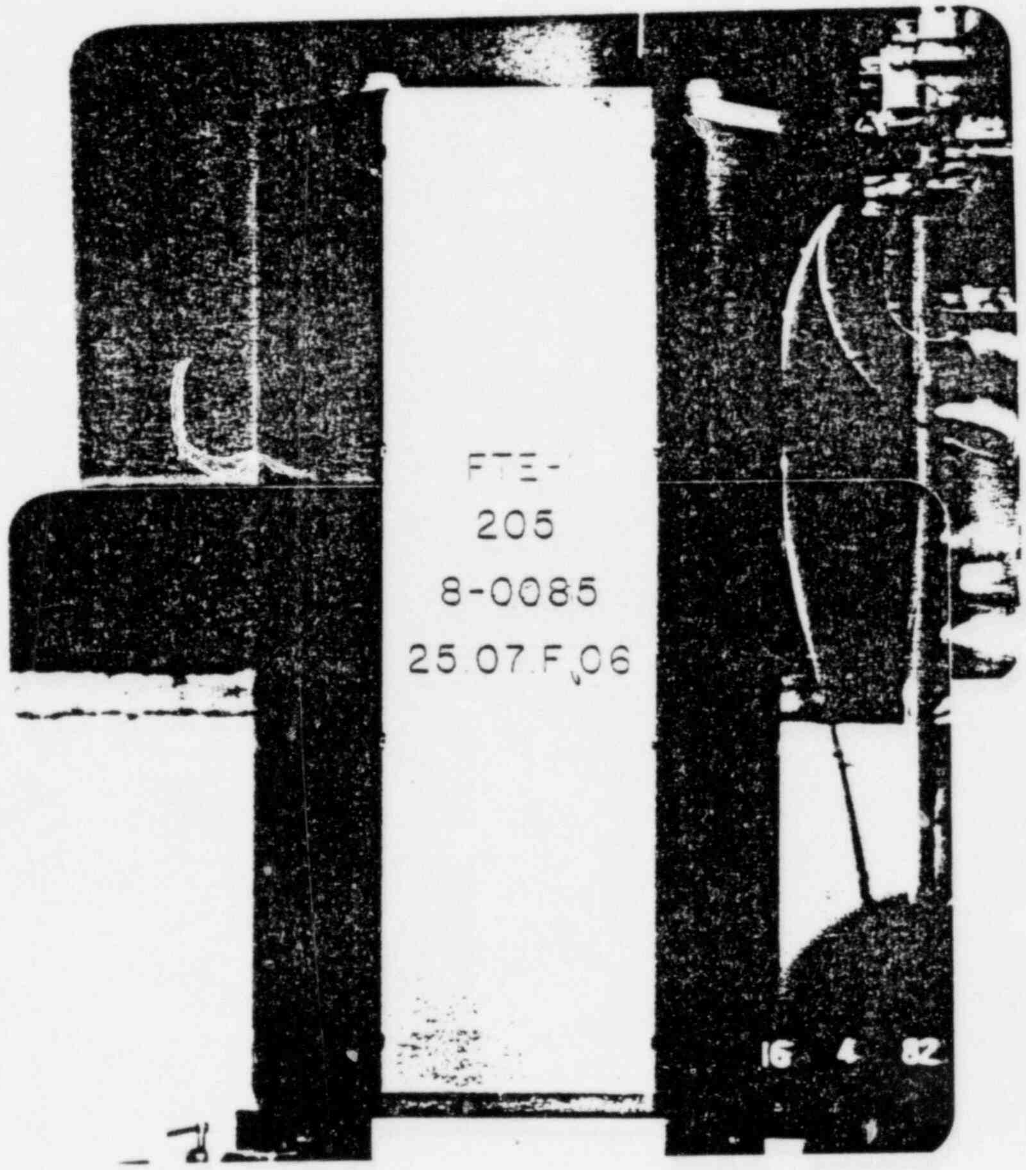

Fig. 2. FSV FTE-1, side face A.

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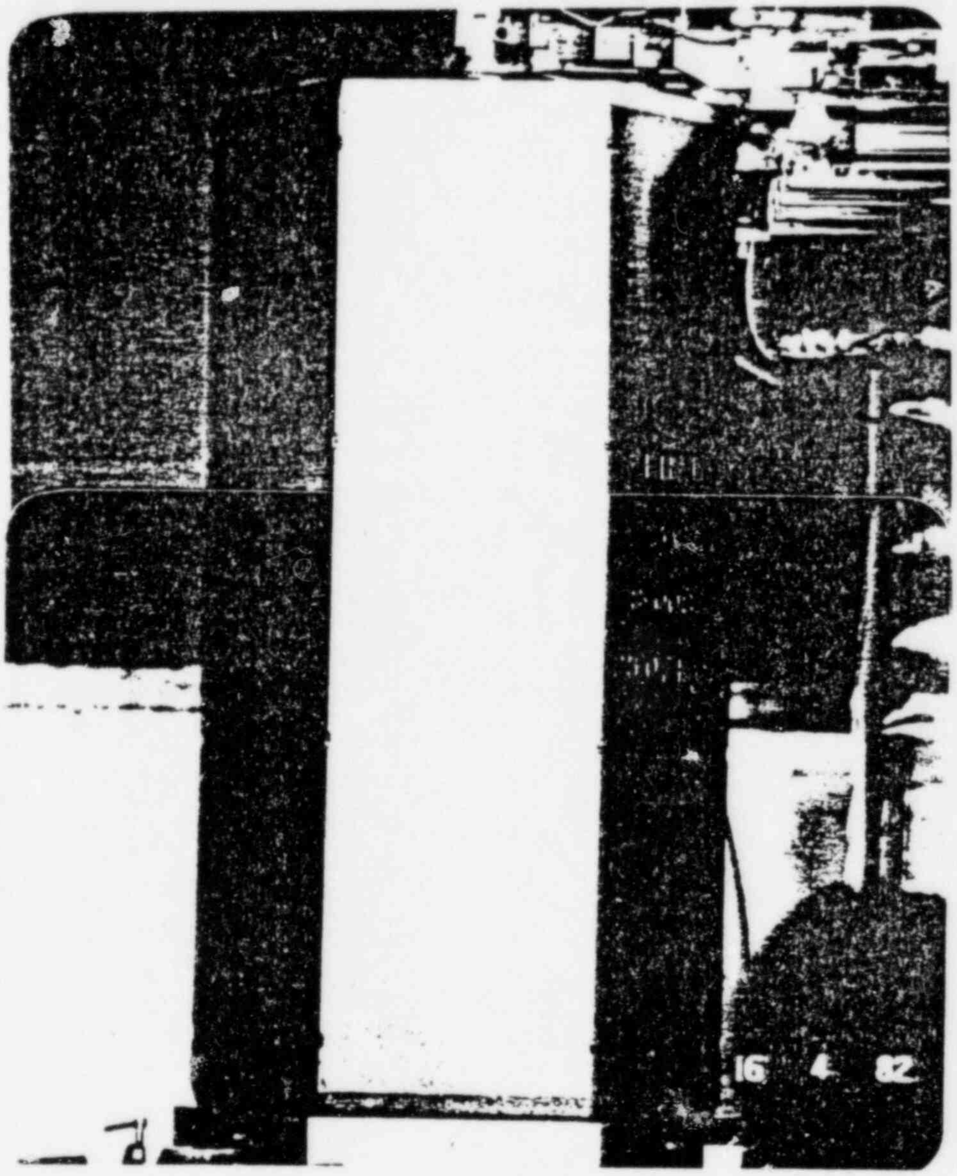

Fig. 3. FSV FTE-1, side face B

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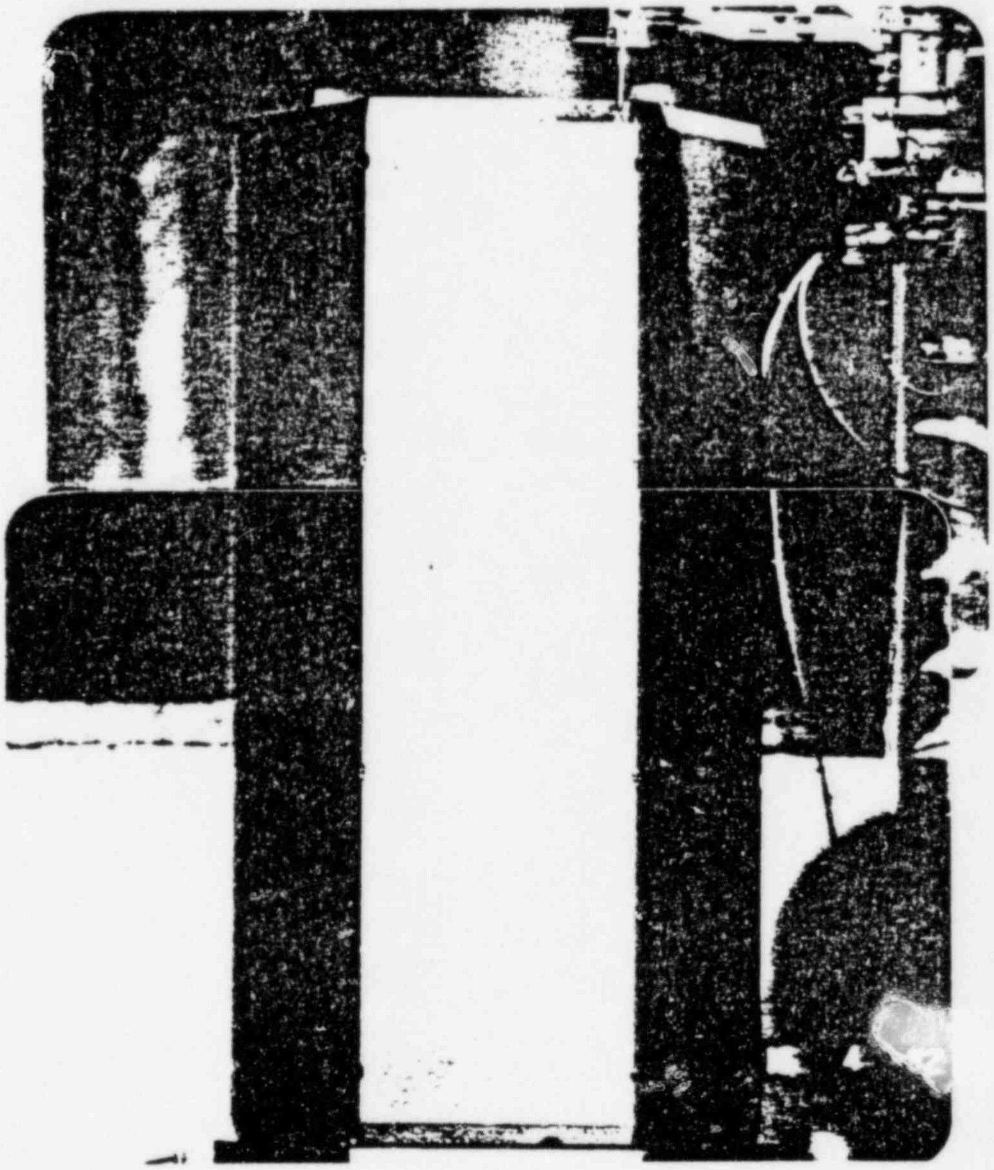

Fig. 4. FSV FTE-1, side face C: discontinuous vertical scratch starting near top center and extending down face.

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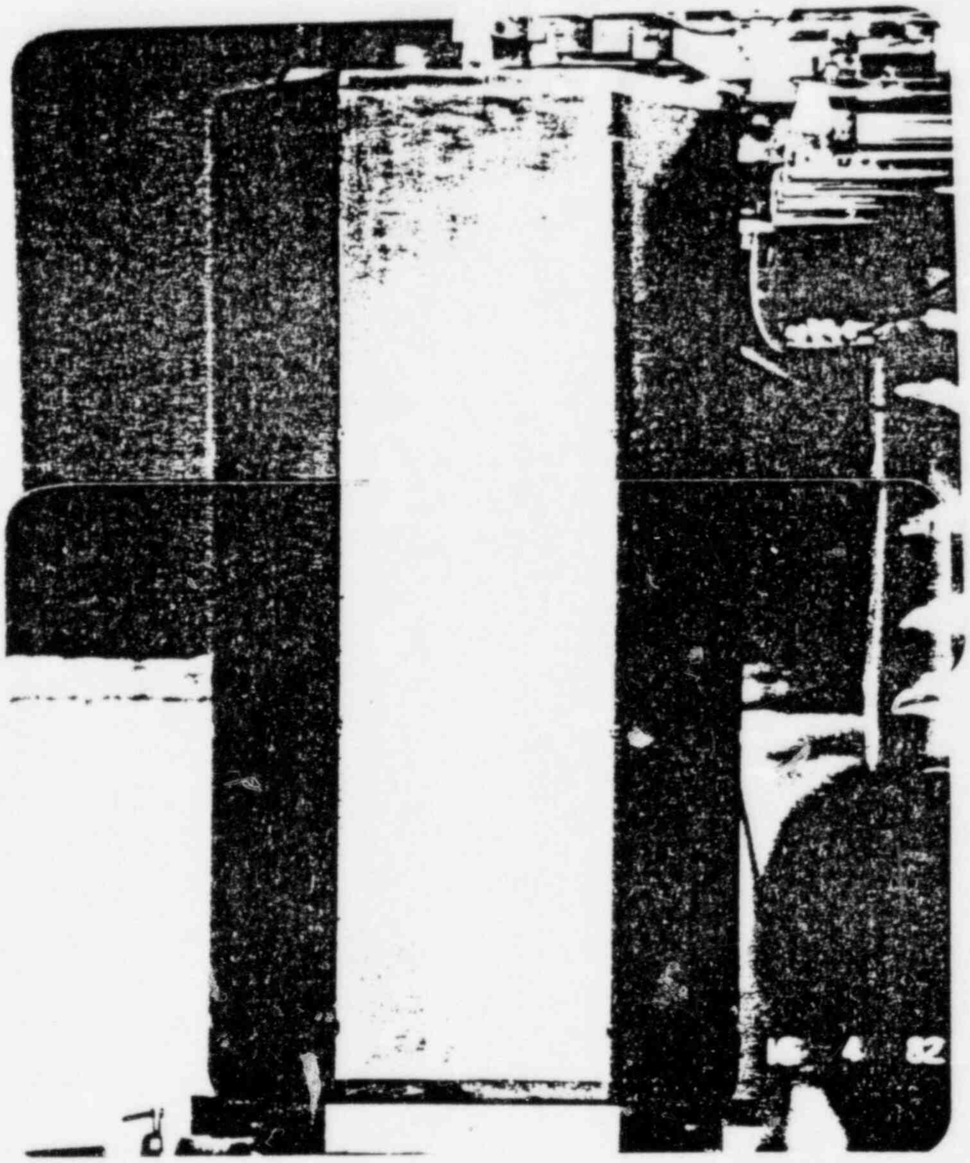

Fig. 5. FSV FTE-1, side face D.

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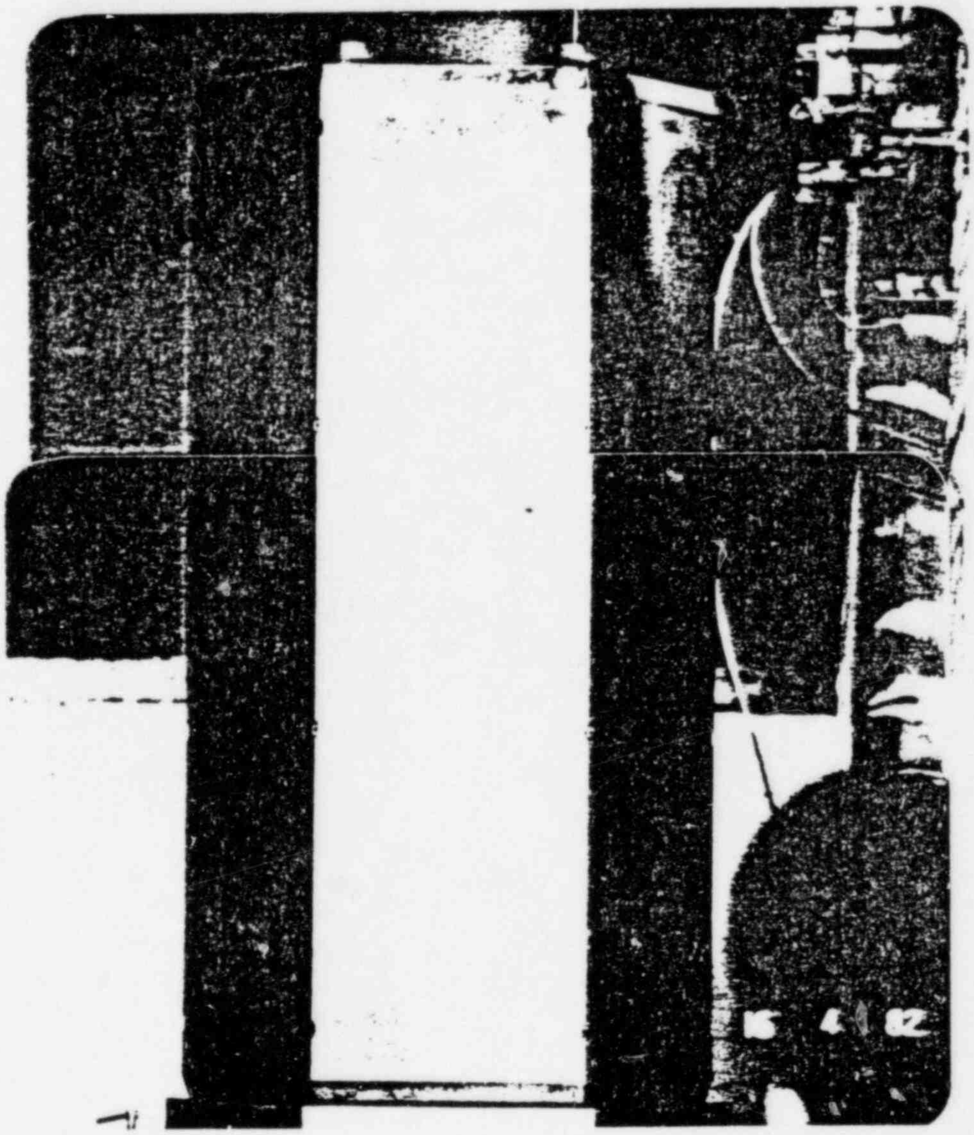

Fig. 6. FSV FTE-1, side face E.

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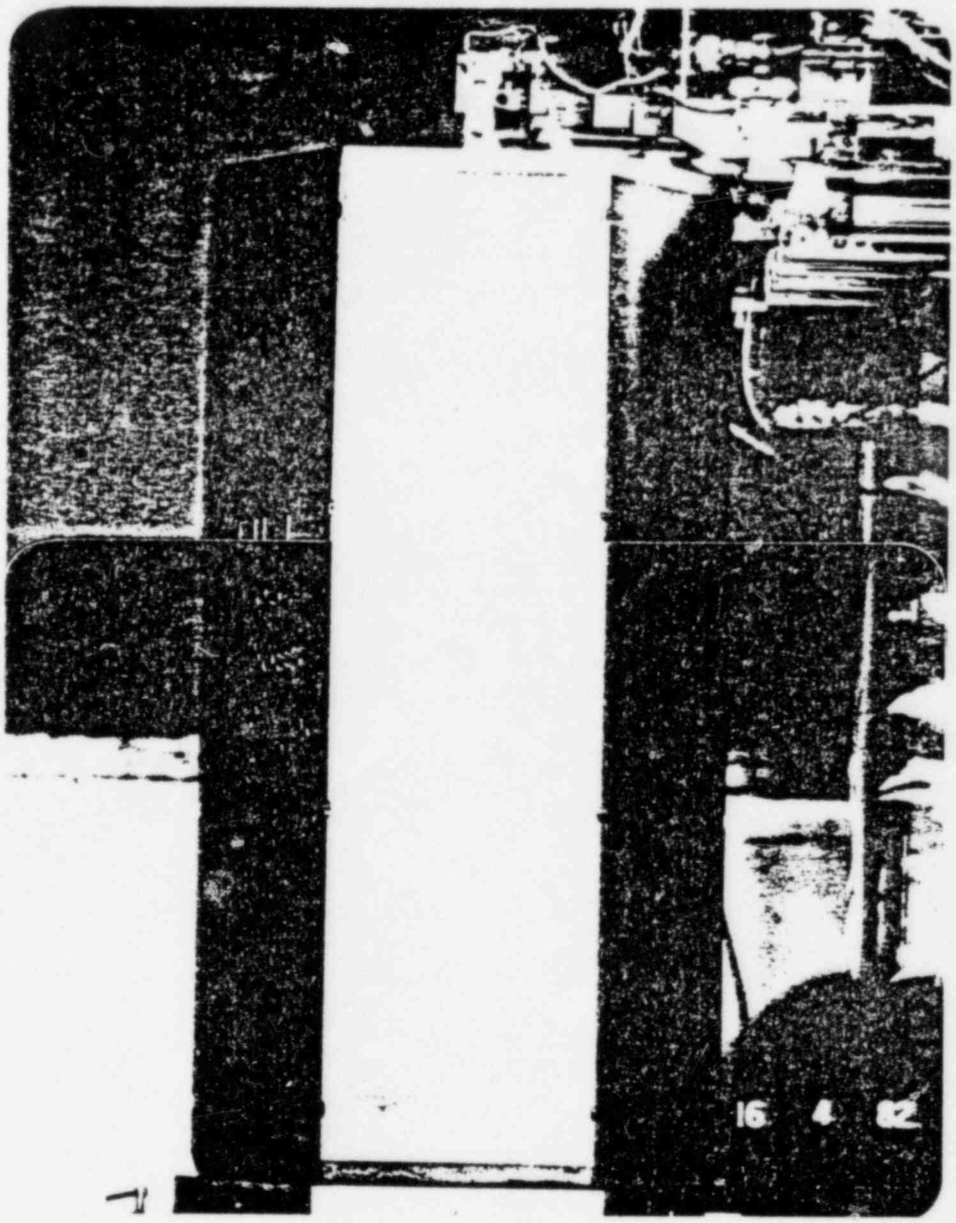

Fig. 7. FSV FTE-1, side face F: short vertical scratch near the upper left corner.

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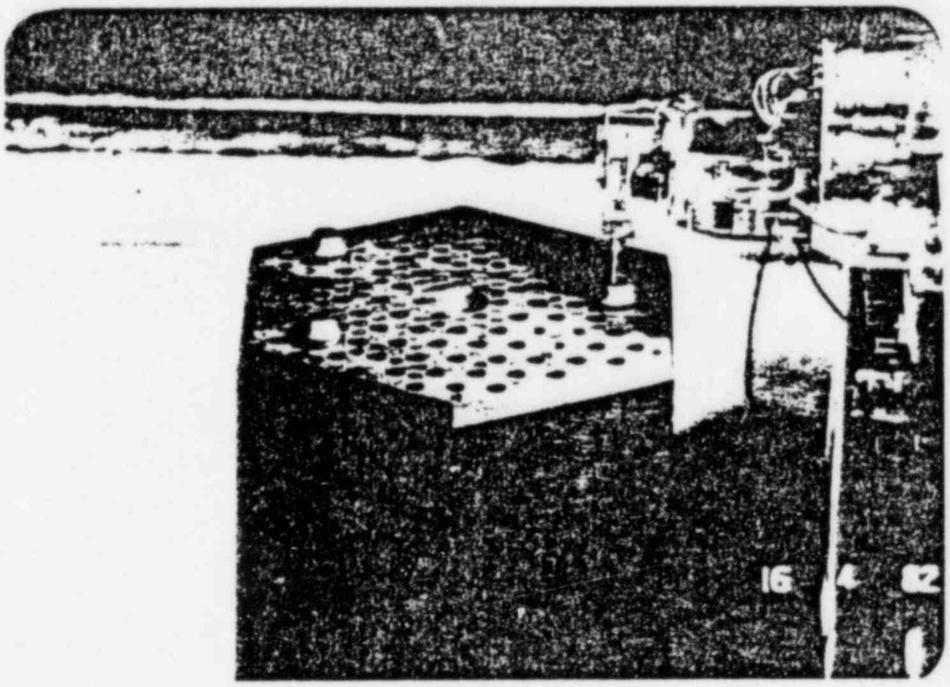
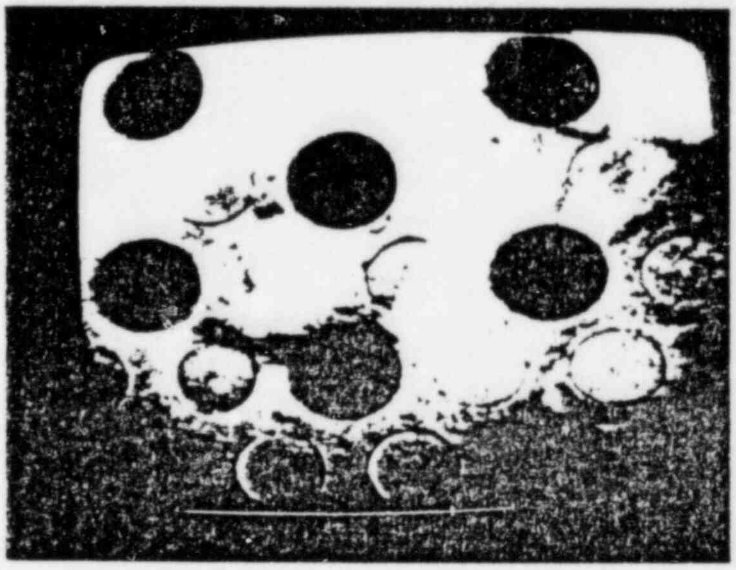


Fig. 8. FSV FTE-1, top surface.



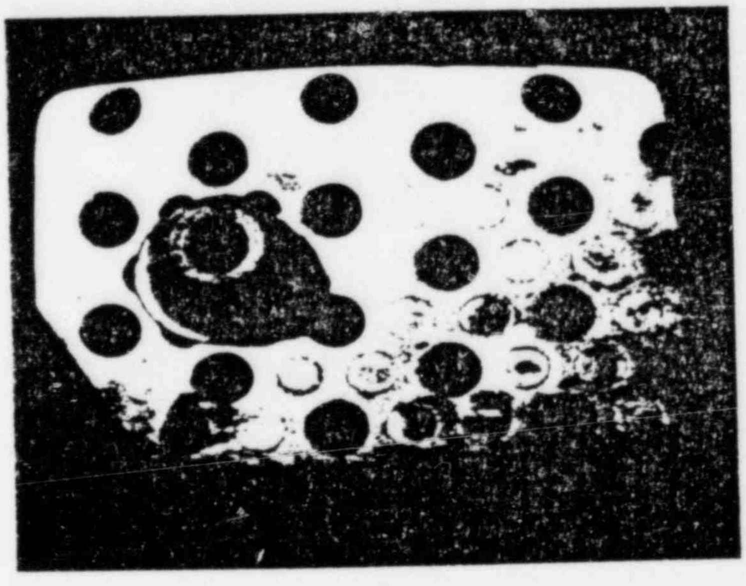


Fig. 9. Examples of disclorations observed on the top surface of FSV fuel test element FTE-1

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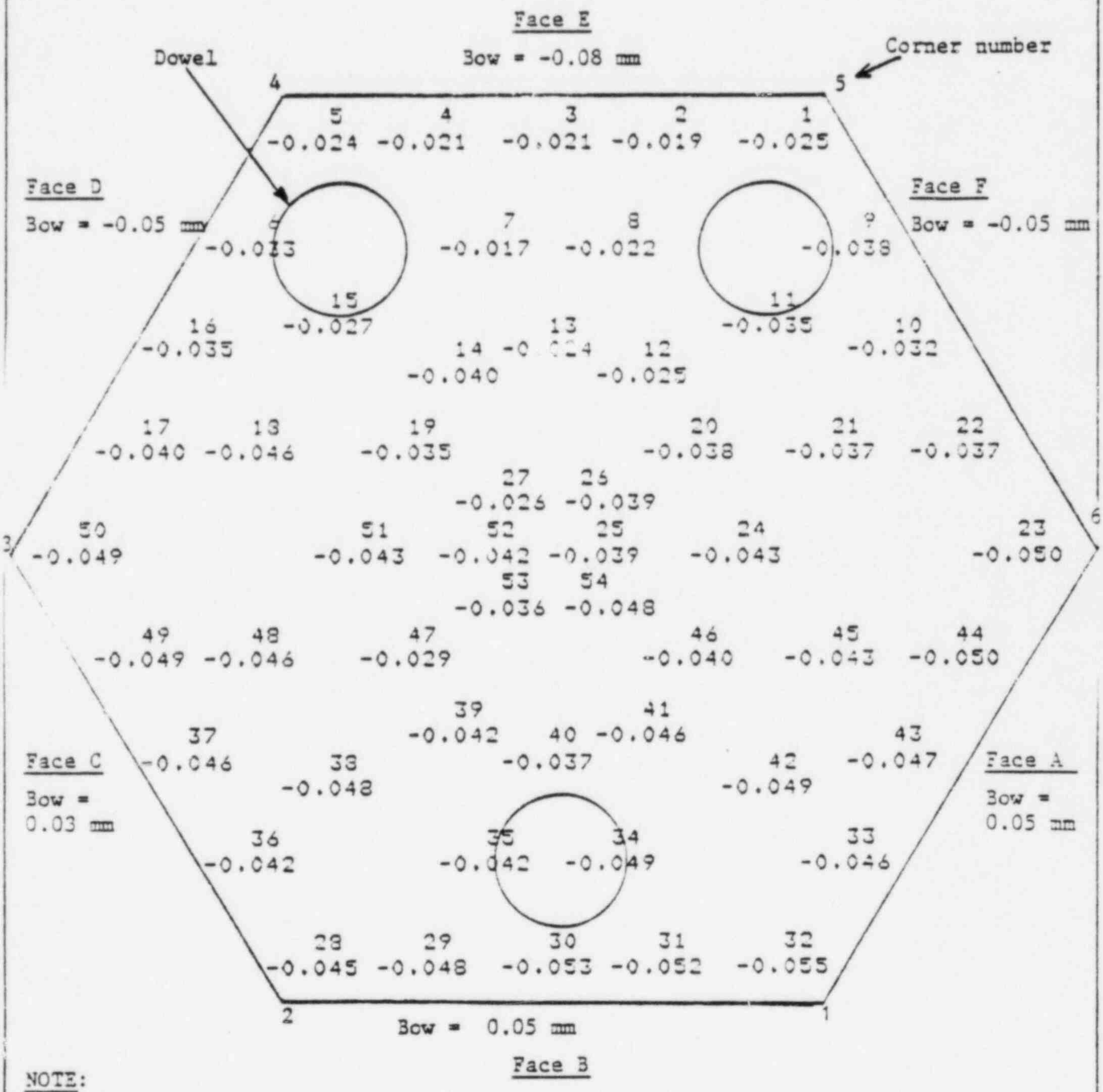

NOTE:

Positive bow - concave
Negative bow - convex
All strains ($\Delta l/l$) in %
Element average axial strain = -0.039% $\pm$0.009% (1σ)

Fig. 10 Measured axial strain and bow for FSV fuel test element FTE-1

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Table 1

TIME-AVERAGED GRAPHITE IRRADIATION TEMPERATURES FOR FSV FTE-1

Axial Point[a]	Temperature (°C) Local point [a]						
	1	2	3	4	5	6	7
1	576	557	563	578	579	572	571
2	598	578	584	600	602	595	592
3	619	597	603	621	622	615	612
4	640	616	623	641	643	636	632
5	663	637	645	665	667	658	655

[a] See Figure 1, Point 1 - top of block.

Table 2

FAST NEUTRON FLUENCES FOR FSV FTE-1

Axial Point[a]	Fast Neutron Fluence ($x10^{25}$ n/m^2, $E>29 fJ_{HTGR}$) Local point [a]						
	1	2	3	4	5	6	7
1	.63	.73	.69	.57	.55	.62	.68
2	.63	.73	.70	.58	.55	.62	.69
3	.64	.74	.70	.58	.56	.62	.69
4	.63	.73	.70	.58	.55	.62	.69
5	.60	.70	.66	.55	.53	.59	.65

[a] See Figure 1, Point 1 - top of block

Table 3. DISTANCE-BETWEEN-FIDUCIAL-HOLE MEASUREMENTS FOR FSV FUEL TEST ELEMENT 1

Dimension (a)	Corner	Preirradiation Distance (in.)	Postirradiation Distance (in.)	Strain ΔZ/Z %	Strain ± 1σ %
L	1	9.002	8.993	-0.097	0.044
L	2	9.002	8.993	-0.099	0.044
L	3	9.003	8.999	-0.046	0.044
L	4	9.004	9.000	-0.036	0.044
L	5	9.004	9.000	-0.036	0.044
L	6	9.003	8.996	-0.075	0.044
L	BLK AV	9.003	8.997	-0.065	0.044
M	1	9.002	8.999	-0.025	0.044
M	2	9.003	8.996	-0.069	0.044
M	3	9.002	8.997	-0.035	0.044
M	4	9.001	8.998	-0.033	0.044
M	5	9.003	9.000	-0.033	0.044
M	6	9.002	9.001	-0.004	0.044
M	BLK AV	9.002	8.999	-0.033	0.044
N	1	9.001	8.998	-0.029	0.044
N	2	9.000	9.000	0.009	0.044
N	3	9.000	8.999	-0.005	0.044
N	4	9.000	9.002	0.021	0.044
N	5	8.999	9.000	0.016	0.044
N	6	9.000	9.003	0.043	0.044
N	BLK AV	9.000	9.001	0.009	0.044
R	1	27.005	26.991	-0.050	0.015
R	2	27.004	26.990	-0.053	0.015
R	3	27.006	26.998	-0.029	0.015
R	4	27.005	27.001	-0.016	0.015
R	5	27.006	27.001	-0.017	0.015
R	6	27.004	27.001	-0.012	0.013
R	BLK AV	27.005	26.997	-0.030	0.015

(a) Dimension L - distance between the top 2 fiducial holes
Dimension M - distance between the middle 2 fiducial holes
Dimension N - distance between the bottom 2 fiducial holes
Dimension R - distance between the top and bottom fiducial holes

The 4 fiducial holes were equally spaced over the elements's length.

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Table 4

MEASURED RADIAL STRAIN (from distance-across-flats measurements) FOR FSV FUEL TEST ELEMENT FTE-1

Distance from Bottom of Block (in.)	Radial Strain (ΔX/X)%			
	Faces A-D	Faces B-E	Faces C-F	Average
1.25	−0.003	0.023	−0.009	0.004
4.25	0.008	0.017	−0.009	0.006
7.25	−0−	0.020	−0.012	0.003
10.25	−0.007	0.014	−0.006	−0−
13.25	−0.011	0.007	−0.022	−0.009
16.25	−0.013	0.003	−0.022	−0.010
19.25	−0.016	−0.001	−0.022	−0.013
22.25	−0.027	−0.010	−0.033	−0.023
25.25	−0.044	−0.015	−0.040	−0.033
28.25	−0.030	−0.025	−0.027	−0.027
~30.35	−0.034	−0.025	−0.036	−0.032
Block Average				−0.012

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Table 5

COOLANT HOLE DIAMETER MEASUREMENTS FOR FSV FUEL TEST ELEMENT FTE-1

HOLE	PRE-IRRADIATION Diameter (in.)	POST-IRRADIATION Diameter (in.)	RADIAL STRAIN	
			ΔX/X %	± 1σ %
312	0.622	0.620	-0.359	0.186
259	0.622	0.620	-0.423	0.186
222	0.622	0.621	-0.262	0.186
181	0.498	0.497	-0.370	0.232
219	0.622	0.621	-0.149	0.186
270	0.622	0.620	-0.310	0.186
319	0.622	0.620	-0.343	0.186
295	0.622	0.620	-0.391	0.186
267	0.622	0.620	-0.310	0.186
235	0.622	0.621	-0.181	0.186
199	0.498	0.497	-0.270	0.232
216	0.622	0.620	-0.310	0.186
264	0.622	0.620	-0.407	0.186
303	0.622	0.621	-0.230	0.186
244	0.622	0.621	-0.262	0.186
213	0.625	0.624	-0.181	0.185
180	0.498	0.497	-0.269	0.232
161	0.623	0.622	-0.165	0.186
158	0.622	0.621	-0.214	0.186
155	0.623	0.621	-0.326	0.186
170	0.623	0.621	-0.310	0.186
167	0.622	0.621	-0.214	0.186
164	0.623	0.622	-0.149	0.186
145	0.496	0.497	0.032	0.233
112	0.623	0.623	-0.005	0.186
81	0.623	0.622	-0.085	0.186
22	0.623	0.623	-0.069	0.186
61	0.623	0.621	-0.326	0.186
109	0.622	0.622	-0.053	0.186
126	0.496	0.495	-0.230	0.233
90	0.623	0.623	0.043	0.186
58	0.623	0.621	-0.278	0.186
30	0.623	0.622	-0.229	0.186
6	0.623	0.623	0.011	0.186
55	0.623	0.620	-0.406	0.186
106	0.623	0.622	-0.165	0.186
144	0.496	0.496	-0.170	0.233
103	0.626	0.625	-0.196	0.185
66	0.623	0.623	-0.005	0.186
13	0.623	0.622	-0.261	0.186
			-0.220 ±	[illegible]

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Table 6

DISTANCE-BETWEEN-COOLANT-HOLE MEASUREMENTS FOR FSV FUEL TEST ELEMENT FTE-1

HOLES	PRE-IRRADIATION Distance (in.)	POST-IRRADIATION Distance (in.)	STRAIN	
			ΔX/X %	± 1σ %
312 TO 270	1.598	1.598	-0.019	0.317
270 TO 219	1.598	1.599	0.062	0.317
219 TO 106	3.818	3.813	-0.112	0.133
106 TO 55	1.597	1.597	0.034	0.318
55 TO 13	1.597	1.598	0.059	0.317
312 TO 13	12.697	12.688	-0.068	0.040
319 TO 295	0.659	0.665	0.954	0.770
295 TO 267	0.660	0.657	-0.557	0.768
267 TO 235	0.660	0.661	0.132	0.769
235 TO 90	4.502	4.499	-0.071	0.113
90 TO 58	0.659	0.659	-0.048	0.769
58 TO 30	0.659	0.659	0.066	0.770
30 TO 6	0.658	0.658	0.067	0.771
319 TO 6	12.193	12.185	-0.070	0.042
303 TO 264	1.598	1.598	0.028	0.317
264 TO 216	1.598	1.598	0.023	0.317
216 TO 109	3.817	3.818	0.031	0.133
109 TO 61	1.598	1.597	-0.046	0.317
61 TO 22	1.596	1.597	0.010	0.318
303 TO 22	12.696	12.691	-0.044	0.040
170 TO 167	1.598	1.599	0.046	0.317
167 TO 164	1.598	1.598	-0.005	0.317
164 TO 161	3.818	3.826	0.224	0.133
161 TO 158	1.598	1.599	0.083	0.317
158 TO 155	1.596	1.598	0.108	0.318
170 TO 155	12.699	12.706	0.053	0.040
13 TO 22	6.037	6.038	0.015	0.084
22 TO 170	6.037	6.035	-0.025	0.084
170 TO 312	6.038	6.036	-0.039	0.084
312 TO 303	6.038	6.038	0.009	0.084
303 TO 155	6.038	6.036	-0.036	0.084
155 TO 13	6.036	6.039	0.043	0.084
AVERAGE			0.029	[illegible]

Table 7

CALCULATED AND MEASURED STRAINS AND BOW FOR FSV FUEL TEST ELEMENT FTE-1

Parameter	Time-averaged[(a)] Temperature (°C)	Fast Fluence[(b)] (10^{25} n/m^2 E>29fJ$_{HTGR}$)	Calculated Value	Measured Value
Element average[(c)] axial strain (%)	615	0.64	-0.135	-0.039±0.009
Axial strain distribution (%)				
Local point 1	619	0.64	-0.132	-0.038±0.009
2	597	0.74	-0.159	-0.046±0.009
3	603	0.70	-0.145	-0.044±0.009
4	621	0.58	-0.115	-0.025±0.009
5	622	0.56	-0.113	-0.027±0.009
6	615	0.62	-0.138	-0.042±0.009
7	612	0.69	-0.155	-0.050±0.009
Element average[(d)] radial strain (%)	615	0.64	-0.096	-0.012±0.016
Radial strain[(d)] distribution (%)				
Top of block	573	0.63	-0.118	-0.032±0.011
Middle of block	615	0.64	-0.097	-0.010±0.016
Bottom of block	659	0.61	-0.071	0.004±0.016
Bow (mm)[(e)]	—	--	0.10	0.08

[(a)] Temperature obtained from SURVEY code calculations based on the GAUGE code depletion analysis of FSV cycle 2 and the FTE-1 power history calculated by the GATT code. The temperature uncertainty (1σ) is approximately 10% of the difference between the temperature and the gas-inlet temperature (~340°C, time averaged).

[(b)] Fast neutron fluences obtained from SURVEY code calculations based on the GAUGE code depletion analysis of FSV cycle 2 and the axial flux profiles

Notations in this column indicate where changes have been made

Notations in this column indicate where changes have been made

Table 7. (Cont.)

for the FSV core, (it was assumed that FTE-1 did not significantly affect the fast neutron flux distribution). The uncertainty in the fast fluence is approximately ±10% (1σ).

(c)The axial strains were obtained by subtracting the thermal strain for H-451 graphite in the axial orientation at 177°C (0.062×10^{-2} mm/mm) from the end-of-life shutdown strains calculated by SURVEY/STRESS.

(d)The radial strains were obtained by subtracting the thermal strain for H-451 graphite in the radial orientation at 177°C (0.071×10^{-2} mm/mm) from the end-of-life shutdown strains calculated by SURVEY/STRESS.

(e)The bow was calculated at the element midplane by SURVEY/STRESS.