Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

August 17, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of	)	Docket Nos.	50-259
Tennessee Valley Authority	)		50-260
			50-296

BROWNS FERRY NUCLEAR PLANT (BFN) -JUSTIFICATION FOR THE NOT USING LEAD TEST ASSEMBLIES (LTA) IN THE BFN BLENDED LOW ENRICHED URANIUM (BLEU) PROGRAM.

On January 11, 2001, NRC and TVA met and discussed the status of the use of excess DOE reactor fuel in TVA owned plants. This letter addresses comments made by NRC during this meeting that were summarized in the NRC meeting summary report dated January 29, 2001.

The staff requested that a case be presented to demonstrate that sufficient information was available for use in evaluating BLEU material without using LTAs. TVA tasked the vender for the BLEU project, Framatome ANP Richland, Inc., to evaluate the need for LTAs. The Enclosure provides the technical justification for concluding that sufficient lead test data has been collected to allow the safe use of BLEU fuel in the BFN reactors without the use of LTAs.

If you have any questions concerning this report, please call me at (256) 729-2636.

Sincerely,

T. E. Abney
Manager of Licensing
and Industry Affairs

Enclosure
cc: See page 2

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ENCLOSURE

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT (BFN)

BLENDED LOW ENRICHED URANIUM (BLEU) LEAD TEST DATA

ENCLOSURE

Blended Low Enriched Uranium Program – Lead Test Assemblies

TVA plans to use Blended Low Enriched Uranium (BLEU) fuel in its Browns Ferry BWR reactors beginning in 2005. A question that has been raised is whether the use of the fuel in reload quantities should be proceeded by lead test assemblies (LTAs). This question has been evaluated and the conclusion reached is that LTAs in the Browns Ferry reactors are not necessary. This conclusion is based on a review of the BLEU LTA experience in the Sequoyah 2 reactor (a PWR) and scoping sensitivity studies for the use of the fuel in the Browns Ferry reactors.

The potential impact of the use of BLEU material on reactor operation and safety is through the impact of the high concentration of U-236 on the power distributions. There are no other significant impacts on fuel behavior. U-236 has a high absorption cross section and affects the reactivity of the assembly. The U-235 enrichment must be increased about .34 w/o in order to achieve the same energy as a virgin uranium assembly of equivalent reactivity. The lattice depletion codes and 3D core simulator codes used by Framatome include the isotope U-236 and are able to adequately model the impact of the high U-236 concentration. The LTA program at Sequoyah and the BWR sensitivity studies described below confirm this expectation.

Lead Test Assembly Experience at Sequoyah Unit 2

TVA loaded four LTAs in the Sequoyah (SQN) Unit 2 Cycle 10 (U2C10) core. The LTAs' physical characteristics were identical to the rest of the reload batch with the exception of the uranium isotopes and use of uniform enrichment fuel stack (no axial blanket). The uranium contained in the LTAs was obtained by blending down recycled, highly enriched uranium from the USA nuclear weapons production program with natural uranium. The resulting blended uranium met the commercial grade uranium specification of ASTM C996-96 with the exception of the concentrations of the uranium isotopes U-232, U-234, and U-236. The use of this material in the SQN U2C10 core was approved by the NRC on April 12, 1999 as Technical Specification Change 98-04 (Amendment Number 234).

The four blended low enriched uranium (BLEU) LTAs were loaded in Sequoyah 2 Cycle 10 in 1999. The LTAs were loaded near the center of the core as shown in Figure 1. Startup physics testing was performed at hot-zero-power conditions to verify that the reload core behaved as expected and complied with technical specification requirements. These tests included initial critical conditions evaluation, boron endpoint determination, isothermal temperature coefficient measurement, boron worth measurement, and control rod bank worth measurements. The results of these tests were as predicted and in compliance with the test requirements.

Power distribution flux maps were taken and analyzed during power ascension to further insure that the core was behaving as expected and that technical specification requirements are met. These flux maps were taken at approximately 4%, 30%, 75%, and 100% power. The flux map for 100% power is provided in Figure 2. A plot of full power hot channel factors F-delta H and F-Q predictions and measurements for the LTAs is shown in Figures 3 and 4, respectively. The differences between measured and calculated power distributions for the LTAs are within the normal expected range of differences. This supports the conclusion that the lattice depletion code and 3D core simulator code can adequately model high concentrations of U-236.

U-236 Sensitivity Studies for Browns Ferry

Neutronics calculations for Browns Ferry BLEU fuel will be performed using Framatome's approved CASMO-4 / MICROBURN-B2 methodology. The CASMO-4 code has a significant amount of experience for a wide variety of fuel designs including PWRs and BWRs with either UO2 or MOX fuel. The results of CASMO-4 have been benchmarked against MCNP calculations, measured critical experiments, measured isotopic data and measured doppler experimental data. In addition, the CASMO-4 / MICROBURN-B2 methodology has been benchmarked extensively against commercial reactor operating data.

The primary neutronic concern for BLEU fuel is the higher concentration of U-236. The CASMO-4 methodology explicitly models all aspects of U-236 with a full compliment of cross-sections for a range of temperatures and U-236 is included in the depletion chains for burnup calculations. The cross sections are based upon the JEF-2.1 data base . Comparisons of individual cross sections for U-236 between JEF-2.1, JEF-2.2 and ENDF/B-6 show very close agreement (< 1%). Comparisons of isotopic data from the Yankee Rowe measurements indicate very good agreement for the concentrations of U-236 for all burnups measured (up to about 35 GWd/MTU). The concentrations observed at higher burnups (about 25 GWd/MTU) are a significant fraction of the initial concentrations expected in the BLEU fuel (about 40%).

A scoping sensitivity study (not formally documented) was performed to evaluate the ability of the lattice depletion code and 3D core simulator code to model high concentrations of U-236. A potential uncertainty in the cross section is the most likely issue with respect to the ability of the codes to model high concentrations of U-236. A conservative maximum uncertainty of ±30% on the U-236 capture cross section was assumed in the studies. The cross section uncertainty was modeled by varying the U-236 concentration in the BLEU fuel ±30%. Calculations were performed using the CASMO3/MICROBURN-B methodology, which explicitly models the U-236 isotope. Comparisons were made between a base case equilibrium cycle design for Browns Ferry loaded with BLEU ATRIUM™-10[1] fuel and corresponding designs with the fresh batch U-236 concentration initialized 30% higher than the base case and 30% lower than the base case.

Results of the sensitivity study show negligible effects on core power distribution, cold shutdown margin, and core reactivity. Differences in the core maximum nodal peaking factor for the three cases were limited to less than 3%, and differences in core maximum radial peaking factor were just over 1%. Both of these are well within the power distribution uncertainty of the methodology. Core k-effective differences among the three cases were about 2 mk, which is much less than the Technical Specification ±1.0% Δk reactivity anomaly criterion. Finally, differences in cold shutdown margin between the three cases were limited to less than 0.1% Δk.

The sensitivity studies show that the uncertainty in the U-236 cross section uncertainty has negligible effect on core power distribution and core reactivity. This confirms the ability of the lattice depletion code and 3D core simulator to adequately model high U-236 concentrations.

[1] ATRIUM is a trademark of Framatome ANP.

Operational Experience with Elevated U-236 Concentrations

In both PWRs and BWRs operating with standard uranium fuel, U-236 levels increase as a function of burnup to a significant fraction of the levels initially present in the BLEU. Framatome's design methods have an excellent record of accuracy over the full cycle length of plants operating with 12-, 18-, and 24-month cycles. This is particularly evident in the comparisons of measured-vs.-calculated Beginning-of-Cycle shutdown margins showing a less than 0.5% $\Delta k/k$ variation for a wide range of BWR cores including variations in core power density, cycle lengths, and numerous combinations of fuel assembly types and vendor products.

In addition to addressing the levels of U-236 encountered during operations of standard fuel, Framatome has had a range of experiences with enriched uranium initially containing elevated concentrations of U-236. These experiences from operations both in the US and Europe have shown no operational problems. Framatome has delivered fuel for six BWRs and four PWRs in the US and one BWR and nine PWRs in Europe with initial U-236 concentrations up to 18 times the ASTM limit.

Conclusions

The rigorous benchmarking of the analytical tools to be used in developing fuel assembly and cycle designs for Browns Ferry demonstrate the adequacy of these tools to accurately predict the fuel performance of assemblies utilizing BLEU. This will assure that cycle designs can be developed to provide adequate margins to limits. Additionally, shutdown margin can be established with confidence to assure safe reactor operations. The accuracy of these methods has been validated in their application in reloads utilizing standard uranium with elevated U-236. Further, the experience with the Sequoyah LTA's provides specific validation of CASMO-based methods for material directly representative of the BLEU material. Therefore, the capability to accurately predict the operational performance and safety margin of BLEU fuel has been adequately established and there is no need for LTA's to precede reload fuel supply of BLEU fuel to Browns Ferry.

Figure 1 Summary of Unit 2 Cycle 10 Final Core Loading Pattern

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A	
					S51	X56	K73	K79	K65	X53	F67					1
			X51	X26	Y48	Y61	Y77	Y05	Y46	Y74	Y75	X17	X49			2
		X79	X28	Y76	Y85	X81	Y10	J39	Y09	X77	Y78	Y49	X23	X47		3
		X21	Y73	X15	X09	Y24	X71	X03	X74	Y11	X07	X16	Y71	X24		4
	F62	Y52	Y84	X06	Y25	X57	Y26	X29	Y27	X66	Y28	X10	Y79	Y72	S59	5
	X60	Y55	X44	Y17	X64	Y29	X37	Y30	X36	Y31	X62	Y16	X50	Y56	X57	6
	K70	Y54	Y22	X70	Y32	X34	**Y04**	X80	**Y03**	X33	Y37	X75	Y21	Y47	K69	7
90°	K80	Y08	J58	X04	X30	Y34	X82	Y35	X41	Y33	X31	X02	J55	Y06	K77	8
	K62	Y69	Y13	X73	Y36	X40	**Y01**	X83	**Y02**	X35	Y38	X72	Y14	Y70	K75	9
	X55	Y68	X43	Y20	X61	Y39	X39	Y41	X38	Y40	X63	Y23	X45	Y57	X58	10
	S46	Y67	Y83	X08	Y42	X68	Y43	X32	Y44	X65	Y45	X11	Y80	Y60	F52	11
		X13	Y53	X27	X12	Y15	X76	X01	X69	Y12	X05	X22	Y50	X20		12
		X52	X18	Y65	Y81	X42	Y18	J48	Y19	X78	Y82	Y58	X19	X84		13
			X48	X14	Y64	Y63	Y62	Y07	Y66	Y51	Y59	X25	X46			14
					F40	X59	K74	K78	K63	X54	S60					15
								0°								

LTAs in **bold**

Figure 2 Summary of 100% Power Flux Map Analysis

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					0.282 0.285 -1.2	0.433 0.435 -0.4	0.368 0.372 -1.1	0.348 0.352 -1.2	0.364 0.370 -1.6	0.425 0.429 -1.0	0.276 0.279 -1.3				
2			0.331 0.334 -0.8	0.636 0.642 -0.8	0.988 1.000 -1.2	1.097 1.106 -0.8	1.083 1.089 -0.6	1.003 1.008 -0.4	1.081 1.084 -0.2	1.099 1.099 0.0	0.995 0.994 0.1	0.643 0.640 0.4	0.333 0.332 0.1		
3		0.329 0.332 -0.9	0.725 0.732 -1.0	1.192 1.207 -1.2	1.305 1.321 -1.2	1.122 1.134 -1.1	1.255 1.254 0.1	0.918 0.914 0.5	1.257 1.248 0.7	1.145 1.130 1.4	1.327 1.318 0.7	1.207 1.205 0.2	0.732 0.732 0.0	0.332 0.334 -0.4	
4		0.634 0.640 -1.0	1.191 1.205 -1.2	1.217 1.234 -1.3	1.209 1.225 -1.4	1.332 1.348 -1.2	1.273 1.248 2.0	1.172 1.150 1.9	1.266 1.245 1.7	1.361 1.345 1.2	1.234 1.223 0.9	1.233 1.234 -0.1	1.203 1.207 -0.3	0.639 0.642 -0.4	
5	0.273 0.279 -2.2	0.982 0.994 -1.1	1.311 1.318 -0.5	1.222 1.223 -0.1	1.287 1.303 -1.2	1.209 1.222 -1.0	1.284 1.273 0.8	1.213 1.191 1.9	1.288 1.268 1.6	1.236 1.219 1.4	1.294 1.303 -0.7	1.219 1.225 -0.5	1.313 1.321 -0.6	0.994 1.000 -0.6	0.283 0.285 -0.6
6	0.416 0.429 -3.2	1.086 1.099 -1.2	1.145 1.130 1.4	1.362 1.345 1.3	1.224 1.219 0.4	1.244 1.258 -1.1	1.150 1.151 0.0	1.257 1.250 0.6	1.173 1.148 2.1	1.285 1.258 2.2	1.262 1.222 3.3	1.336 1.348 -0.9	1.127 1.134 -0.6	1.098 1.106 -0.7	0.433 0.435 -0.4
7	0.360 0.370 -2.9	1.072 1.084 -1.1	1.260 1.248 1.0	1.259 1.245 1.2	1.274 1.268 0.5	1.150 1.148 0.2	1.226 1.226 0.0	1.119 1.110 0.8	1.245 1.226 1.6	1.175 1.151 2.1	1.293 1.273 1.5	1.252 1.248 0.4	1.248 1.254 -0.5	1.080 1.089 -0.8	0.367 0.372 -1.4
8	0.343 0.352 -2.5	0.992 1.008 -1.5	0.911 0.914 -0.3	1.163 1.150 1.1	1.207 1.191 1.4	1.255 1.250 0.4	1.116 1.110 0.5	1.215 1.208 0.6	1.131 1.110 1.8	1.265 1.250 1.2	1.206 1.191 1.3	1.155 1.150 0.4	0.909 0.914 -0.5	0.998 1.008 -0.9	0.344 0.352 -2.2
9	0.363 0.372 -2.5	1.071 1.089 -1.6	1.240 1.254 -1.2	1.254 1.248 0.5	1.278 1.273 0.4	1.167 1.151 1.4	1.250 1.226 2.0	1.133 1.110 2.1	1.237 1.226 0.9	1.158 1.148 0.8	1.270 1.268 0.1	1.249 1.245 0.4	1.244 1.248 -0.4	1.075 1.084 -0.8	0.362 0.370 -2.2
10	0.418 0.435 -3.9	1.062 1.106 -4.0	1.106 1.134 -2.5	1.342 1.348 -0.5	1.218 1.222 -0.3	1.266 1.258 0.6	1.188 1.148 3.4	1.283 1.250 2.7	1.183 1.151 2.8	1.267 1.258 0.7	1.216 1.219 -0.3	1.338 1.345 -0.5	1.122 1.130 -0.7	1.041 1.099 -5.3	0.407 0.429 -5.2
11	0.269 0.285 -5.5	0.946 1.000 -5.4	1.249 1.321 -5.4	1.221 1.225 -0.4	1.295 1.303 -0.6	1.224 1.219 0.4	1.293 1.268 2.0	1.231 1.191 3.4	1.316 1.273 3.3	1.259 1.222 3.1	1.308 1.303 0.5	1.232 1.223 0.7	1.324 1.318 0.5	0.941 0.994 -5.3	0.264 0.279 -5.4
12		0.622 0.642 -3.1	1.194 1.207 -1.1	1.222 1.234 -0.9	1.213 1.223 -0.8	1.331 1.345 -1.0	1.262 1.245 1.4	1.181 1.150 2.6	1.281 1.248 2.7	1.387 1.348 2.9	1.251 1.225 2.1	1.260 1.234 2.1	1.212 1.205 0.5	0.621 0.640 -3.0	
13		0.331 0.334 -0.8	0.725 0.732 -0.9	1.192 1.205 -1.1	1.304 1.318 -1.1	1.118 1.130 -1.0	1.241 1.248 -0.6	0.919 0.914 0.6	1.264 1.254 0.8	1.145 1.134 0.9	1.331 1.321 0.7	1.227 1.207 1.7	0.739 0.732 0.9	0.330 0.332 -0.8	
14			0.330 0.332 -0.8	0.631 0.640 -1.5	0.971 0.994 -2.3	1.077 1.099 -2.0	1.073 1.084 -0.9	0.998 1.008 -0.9	1.080 1.089 -0.8	1.093 1.106 -1.1	1.002 1.000 0.2	0.653 0.642 1.7	0.335 0.334 0.5		
15					0.270 0.279 -3.4	0.420 0.429 -2.2	0.364 0.370 -1.7	0.346 0.352 -1.7	0.367 0.372 -1.5	0.431 0.435 -0.9	0.285 0.285 0.2				

Key:

xxx	xxx – Measured
yyy	yyy – Predicted
zzz	zzz – Percent Difference

SUMMARY

RMS ERROR (%):	1.758
MEAS. AFD:	-0.040
PRED. AFD:	-3.256
MEAS. AX. OFFSET:	-0.040
PRED. AX. OFFSET:	-3.256

CORE TILT:

0.9971	1.0047
0.9938	1.0043

TITLE LINE:

1 IN10F204 RTP=100% D@218 STEPS BU=164 MWD/MTU 1 05/17/99 10:35:56 INCORE3D

Figure 3 Hot Channel Factor (F-DH) at Full Power

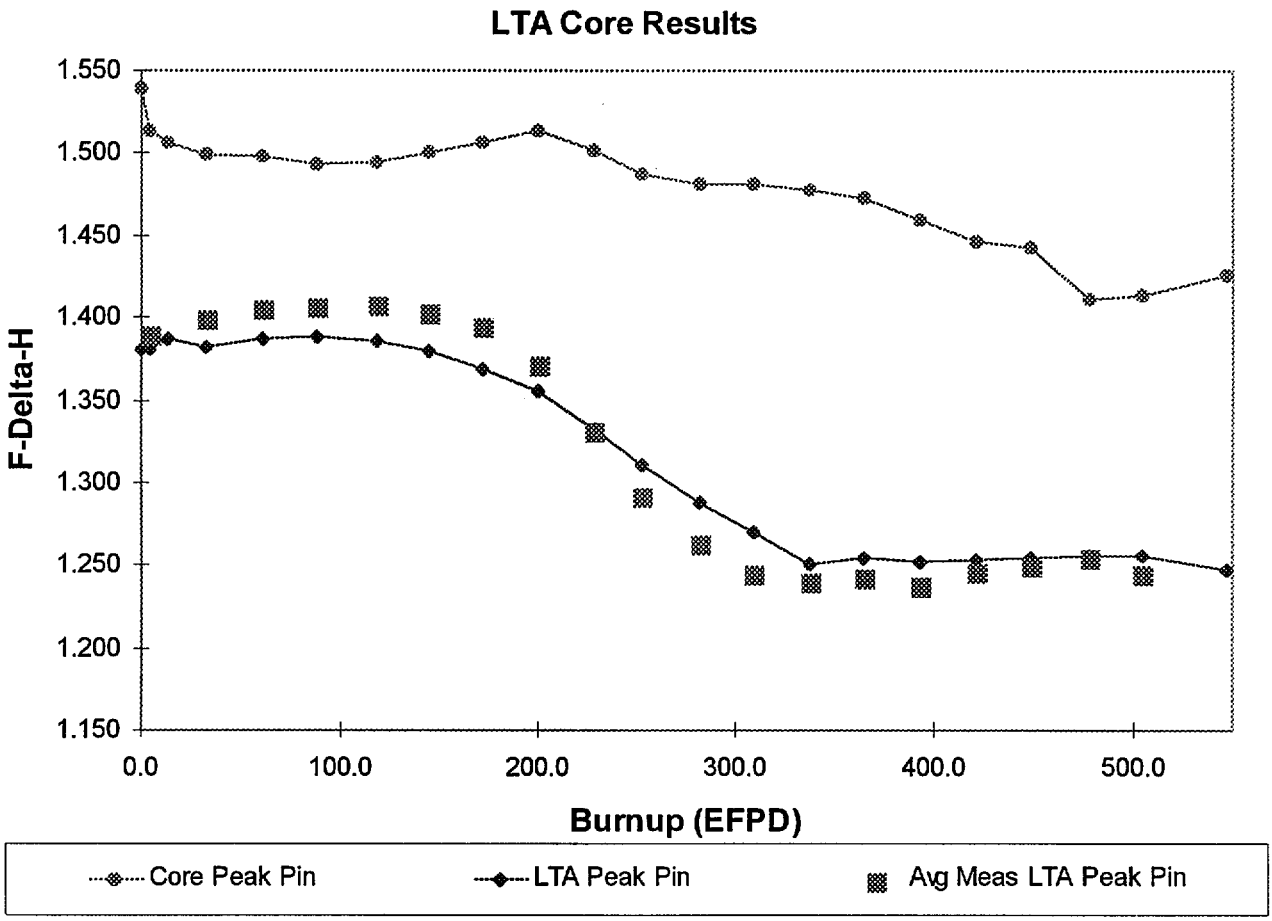
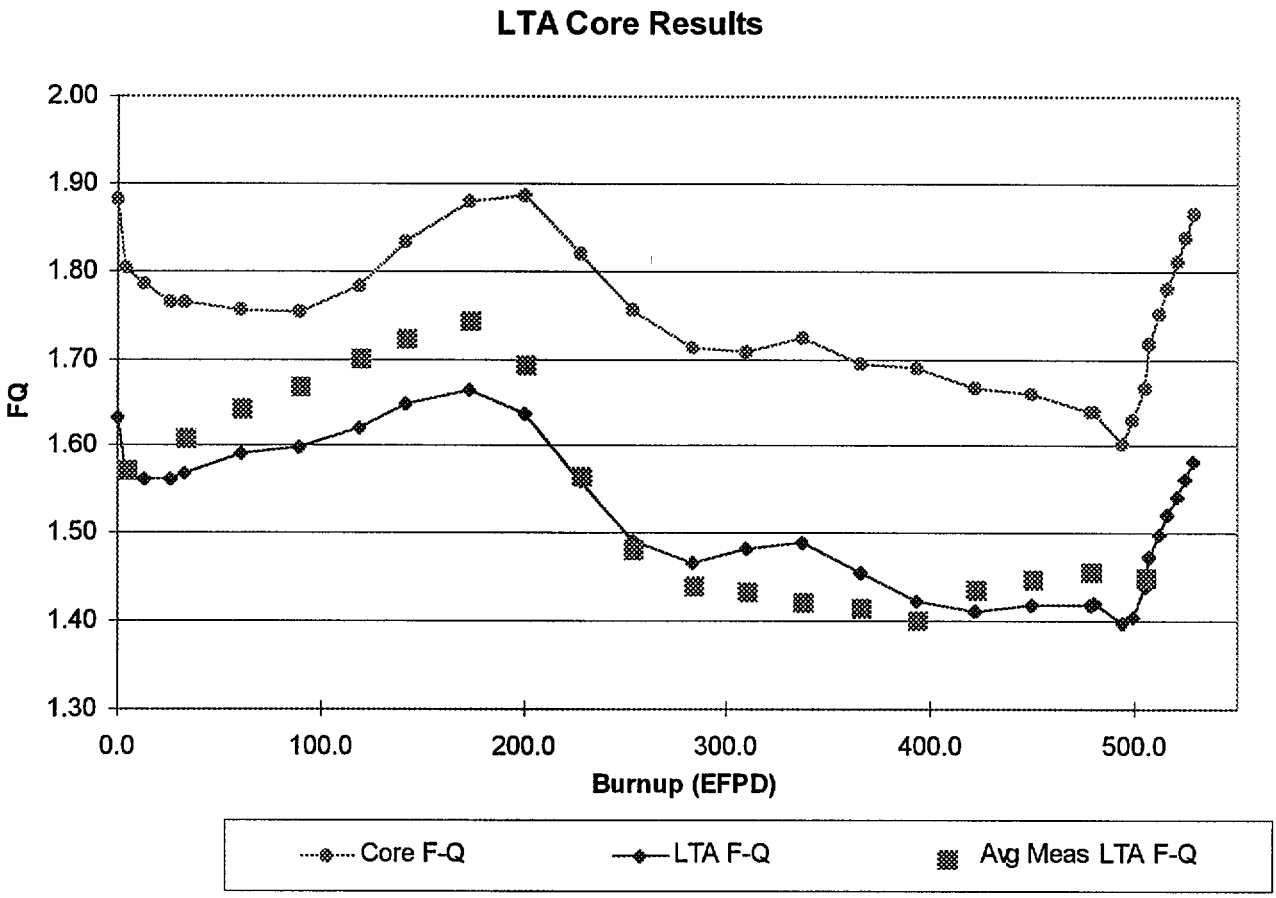
Figure 4 Hot Channel Factor (FQ) at Full Power