APPENDIX A

WESTINGHOUSE EXPERIENCE WITH HIGH POWER LEVEL FUEL RODS

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1.0 INTRODUCTION

Indian Point Unit No. 2 is designed to operate at a maximum power rating of 18.5 kw/ft. This slight increase over the value selected for previously designed Westinghouse plants (CVTR is presently licensed to operate at 17 kw/ft) is possible because of improved knowledge of the performance of fuel rods at high power ratings.

Based upon Westinghouse in-pile experiments and tests at other facilities, this reactor can safely operate at 18.5 kw/ft. This test work is summarized in Section 2 of this appendix.

The safety of the operation at 18.5 kw/ft is emphasized by two experiments. The first, reported in Section 2.3.2 of this appendix, was run with intentionally defected Zircaloy-clad rods. No fretting, no crud deposition, no indication of attack of the cladding or fuel was observed in the vicinity of the defect.

The second, reported in Section 2.5, was run at extremely high power ratings (up to 60 kw/ft) resulting in extensive melting (up to 75 per cent of the pellet cross sectional area). Although one of the four capsules experienced localized clad melting, no catastrophic failure occured even after many days of operation with the partially molten clad.

As fuel rod power levels are increased, two areas of behavior of interest related to long term cladding integrity are the thermal expansion (radial and axial) of the UO_2, and the fractional fission gas release from the UO_2.

Radial thermal expansion of the UO_2 fuel relative to the cladding decreases, the pellet-to-clad gap and, in those areas where contact is achieved increases the interfacial pressure between fuel and clad. In both cases the gap and contact conductance are increased and the UO_2 surface temperature is lowered. Radial thermal expansion of the UO_2 is deleterious only if excessive cladding deformation occurs. The clad-pellet gap thickness selected for the present design (0.0065 inches) is large enough to avoid excessive stress of the cladding, and small enough to avoid excessive temperature drop and consequently excessive center temperature.

Axial thermal expansion of UO_2 relative to the cladding results in a movement of the UO_2 fuel column into a plenum designed to accommodate this expansion and the released fission gases.

Gross axial expansion of the UO_2 pellet column is limited by fabricating each UO_2 pellet with a depression in both ends.

The plenum length required to accommodate fission gases is many orders of magnitude longer than that required for the axial expansion alone.

Another area investigated at high power rating is the fuel pellet temperature.

Current practice in the design and operation of ceramic fueled cores generally limits the maximum power density to prevent central melting(1) within fuel pellets. Results from recent in-pile experiments(2) have demonstrated that it may be acceptable to operate with some portion of the fuel molten, but until further evidence is available, the current design practice will be applied to this reactor.

If the UO_2 center melting is the design limitation, the reactor could be designed to operate at steady state maximum fuel rod power levels up to at least 20 kw/ft (656 w/cm) and still allow a 112% overpower condition to occur without obtaining any UO_2 center melting even after 50,000 MWD/MTU as shown in Section 5 of this Appendix. Such a combination of conditions will not occur as the fuel rods which will experience the maximum power ratings will not experience the maximum burnup.

A further area investigated for operation of UO_2 fuel rods at high power levels for long periods of time is the amount of fission gases which will be released from the UO_2, affecting internal pressures within the fuel rod. In order to prevent excessive internal pressures, current fuel rod designs incorporate a void space at one end of the fuel rod as previously mentioned.

Westinghouse experiments with fuel at high power rating are summarized in Section 2.

Section 3 summarizes the Westinghouse experience with large Zircaloy cores, i.e. CVTR and Shippingport which gives a good indication of the general performance of large Zircaloy cores.

Section 4 summarizes the results of the experiments performed to measure UO_2 thermal conductivity.

Section 5 contains general conclusions.

2.0 WESTINGHOUSE EXPERIENCE WITH HIGH POWER LEVEL FUEL RODS

Westinghouse experience with high power level fuel rods is illustrated in this section. Emphasis is given to the experiments developed with Zircaloy clad fuel rods. The main experiments with stainless steel clad fuel rods are only briefly summarized.

2.1 IRRADIATION OF SIX CAPSULES CONTAINING SAMPLES FROM THE CVTR CORE IN THE WESTINGHOUSE TEST REACTOR (MARCH AND NOVEMBER 1960)

2.1.1 GENERAL

As part of the CVTR Research and Development Program, a series of capsule irradiation experiments[3] was devised to define more clearly the thermal performance capabilities of sintered UO_2 pellets contained in Zircaloy-2 cladding.

The philosophy of the program was to carefully control irradiation test in order to obtain unambiguous thermal performance data for use by reactor designers. The parameters to be evaluated at various fuel rod power levels included the effect of the initial cold diametral fuel-to-clad gap on UO_2 surface and center temperatures and on the cladding stress, due to the fuel-clad differential expansion.

The capsules were designed to minimize errors and potential problems which would lead to difficulty in interpreting the experimental results, and to minimize variation in both radial and axial thermal neutron flux.

2.1.2 DESCRIPTION OF EXPERIMENT

Six capsules denoted R-1, R-2, R-4, R-5, R-6, R-8 and R-11, each containing three fuel rods, were irradiated at fuel rod power levels of from 11 to 24 kw/ft (360 to 785 w/cm).

The fuel rod configuration used was a Zircaloy-2 tube containing a column of UO_2 pellets, having a fuel length of about 5 inches. All UO_2 pellets used were right circular cylinders 0.430 inches in diameter with a nominal density of 10.3 g/cm^3 (94% of theoretical). The inside and outside diameters of the Zircaloy-2 cladding were varied to obtain the required cold diametral gaps; however, the cladding wall thickness was maintained at 0.032 inches in all cases.

A 0.080 inch axial plenum was provided in all fuel rods to accommodate axial thermal expension of the UO_2 pellet column relative to the Zircaloy-2 cladding.

Cold diametral gaps of 0.006, 0.012 and 0.025 inches were selected. Three fuel rods, each with a different cold diametral gap, were irradiated simultaneously to eliminate possible variations in fuel rod power level. Different fuel rod power levels were obtained by using UO_2 pellets of different U-235 content.

An irradiation time of 40 hours was chosen to allow fuel redistribution due to either sintering of the UO_2 or other time and temperature dependent phenomena. Gross thermal cycling of the UO_2 fuel was not desired. The 40 hours irradiation time was short enough to preclude the possibility of a major change in thermal neutron flux because of a reactor trip or significant changes in control rod positions.

Table 1 summarizes the basic fuel rod parameters used for the six rabbit capsule irradiations.

2.1.3 POST IRRADIATION EXAMINATION

After irradiation, the rabbit capsules were examined in the WTR hot cells. The capsules were disassembled and the fuel rod samples removed. Length and diameter measurements were taken on all the fuel rod samples to detect any deformation of the Zircaloy-2 cladding which may have occurred because of interaction with the UO_2 or because of internal gas pressure buildup as the result of excessive fission gas release.

The fuel rods were then punctured and the contained gases collected and analyzed. All the fuel rods were sectioned at the UO_2 pellet interfaces and photomacrographs taken on the interfaces. Selected pellets from each fuel rod were dissolved and the resultant solution diluted for subsequent fission product nuclide analysis to determine fission rates and fuel burnup. Samples of irradiated UO_2 pellets were pressure mounted in an epoxy resin for metallographic examination of the transverse cross sections. These samples were subsequently ground, polished and etched for metallographic examination. The radii at which various microstructural changes occurred such as UO_2 grain growth and columnar grain formation were measured.

2.1.3.1 Fuel Rod Dimensional Changes

The diameter and length of all the fuel rods were measured after irradiation to extablish whether swelling or other deformation of the cladding had occurred during irradiation.

Table 1

RABBIT CAPSULE NUMBER	FUEL ROD NUMBER	FUEL ENRICHMENT	NOMINAL COLD DIAMETRAL CLEARANCE	POWER LEVEL	LINEAR POWER OUTPUT	SURFACE HEAT FLUX AT FUEL ROD O.D.
		% U-235	Inches	Kw/Ft	W/Cm	$\frac{Btu}{Hr, Ft^2} \times 10^3$
R-1	1-1	2.6	0.006			286
	1-2	2.6	0.012	11.0 ± 0.5	360	283
	1-3	2.6	0.025			277
R-2	2-1	2.6	0.006			286
	2-2	2.6	0.006	11.0 ± 0.5	360	286
	2-3	2.6	0.025			277
R-4	4-1	3.8	0.006			417
	4-2	3.8	0.012	16.0 ± 0.8	525	412
	4-3	3.8	0.025			402
R-6	6-1	5.2	0.006			470
	6-2	5.2	0.012	18.0 ± 1.0	590	464
	6-3	5.2	0.025			454
R-8	8-1	5.2	0.012			464
	8-2	5.2	0.012	18.0 ± 1.0	590	464
	8-3	5.2	0.012			464
R-11	11-1	7.6	0.006			624
	11-2	7.6	0.012	24.0 ± 1.2	785	618
	11-3	7.6	0.025			602

Note: UO_2 pellets 0.430 inch diameter, nominal 94% dense, O/U ratio 2.00-2.01
Zircaloy cladding dimensions varied to give required cold diametral clearances.
Irradiated 40 hours in the WTR rabbit tube facilities.

The following measurements were made:

-Overall length: two readings for each measurement;

-Diameters at the following positions:

one-inch from top: two measurements, 90^{o} apart,

Center: as above,

One-inch from bottom: as above.

The diameter measurements taken 90^{o} apart were in agreement with each other within the recognized precision limits ($\pm$ 0.0005 inches).

Table 2 summarizes the pre- and post-irradiation measurements for the fuel rod samples from the rabbit capsules.

No significant dimensional changes occurred during irradiation of the fuel rod samples. In some cases; i.e. fuel rods with initial dimetral gaps of 0.006 inches operated at the higher fuel rod power levels (fuel rods No. 4-1, 6-1 and 11-1, as listed in Table 1), the radial thermal expansion of the UO_2 fuel relative to the cladding should have resulted in a zero diametral gap during irradiation. No significant diametral changes were noted on these fuel rods indicating that the interfacial pressures between the UO_2 fuel and the Zircaloy-2 cladding were not sufficient to plastically deform the clad.

2.1.3.2 Metallography of Fuel

Selected cross sections of UO_2 samples taken from all fuel rods were pressure mounted in an epoxy resin for subsequent metallographic preparation. The radii corresponding to various microstructural features such as equiaxed grain growth, columnar grain formation, and changes associated with the solidification of molten UO_2 were measured to establish radial temperature profiles. No fuel rod (0.006, 0.012, and 0.025 initial diametral gap fuel rods) experienced center melting while operating at 11, 16, and 18 kw/ft. Only the fuel rod with 0.025 inches initial diametral gap experienced melting in the central region. But no significant diametral change was measured, as shown in Table 2. The fuel rods with 0.006 and 0.012 inches initial diametral gap and which operated at 24 kw/ft. experienced no center melting. It can be concluded that the surface and center temperatures of sintered UO_2 fuel can be lowered by decreasing the initial diametral gap, and thus allowing reactor operation at 24 kw/ft. without central melting.

Table 2

DIMENSIONAL MEASUREMENTS ON FUEL ROD SAMPLES FROM THE RABBIT CAPSULES

(Precision ± 0.0005 inches)

Rabbit Capsule No.		R-1			R-2			R-4		
		Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches
Fuel Rod Number		1-1			2-1			4-1		
*Length	l	6.412	6.413	+0.001	6.412	6.411	-0.001	6.419	***6.410	***-0.009
**Diameter	d_1	0.500	0.502	+0.002	0.501	0.501	0	0.501	0.501	0
	d_2	0.502	0.502	0	0.502	0.502	0	0.502	0.502	0
	d_3	0.501	0.502	+0.001	0.501	0.501	0	0.501	0.502	+0.001
Fuel Rod Number		1-2			2-2			4-2		
Length	l	6.416	6.418	+0.002	6.408	6.409	-0.001	6.417	***6.409	***-0.008
Diameter	d_1	0.505	0.505	0	0.501	0.501	0	0.505	0.505	0
	d_2	0.506	0.506	0	0.501	0.502	+0.001	0.506	0.506	0
	d_3	0.506	0.506	0	0.501	0.501	0	0.506	0.506	0
Fuel Rod Number		1-3			2-3			4-3		
Length	l	6.414	6.414	0	6.411	6.409	-0.002	6.410	6.409	-0.001
Diameter	d_1	0.518	0.519	+0.001	0.519	0.519	0	0.518	0.519	+0.001
	d_2	0.518	0.519	+0.001	0.518	0.519	+0.001	0.519	0.519	0
	d_3	0.518	0.518	0	0.518	0.519	+0.001	0.518	0.518	0
Rabbit Capsule No.		R-6			R-8			R-11		
		Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches	Pre Inches	Post Inches	Change Inches
Fuel Rod Number		6-1			8-1			11-1		
Length	l	6.4240	6.4210	-0.003	6.4140	6.4120	-0.002	6.4110	6.4110	0
Diameter	d_1	0.4985	0.4995	+0.0010	0.5035	0.5036	+0.0001	0.4984	0.4986	+0.0002
	d_2	0.4994	0.5004	+0.0010	0.5039	0.5046	+0.0007	0.4993	0.5002	+0.0009
	d_3	0.4990	0.5006	+0.0016	0.5036	0.5043	+0.0007	0.4992	0.5001	+0.0009
Fuel Rod Number		6-2			8-2			11-2		
Length	l	6.4160	6.4130	-0.003	6.4100	6.4094	-0.0006	6.4140	6.4150	+0.001
Diameter	d_1	0.5034	0.5033	-0.0001	0.5033	0.5034	+0.0001	0.5032	0.5038	+0.0006
	d_2	0.5039	0.5051	+0.0012	0.5046	0.5048	+0.0002	0.5041	0.5048	+0.0006
	d_3	0.5035	0.5046	+0.0011	0.5041	0.5043	+0.0002	0.5037	0.5047	+0.0010
Fuel Rod Number		6-3			8-3			11-3		
Length	l	6.4090	6.4070	-0.002	6.4060	6.4050	-0.001	6.4070	6.4040	-0.003
Diameter	d_1	0.5157	0.5167	+0.0010	0.5030	0.5036	+0.0006	0.5159	0.5165	+0.0006
	d_2	0.5164	0.5173	+0.0009	0.5041	0.5045	+0.0004	0.5161	0.5160	-0.0001
	d_3	0.5161	0.5171	+0.0010	0.5035	0.5045	+0.0010	0.5159	0.5168	+0.0009

* Length: Overall length, two readings made for each measurement. Number listed is the average of the two readings.

** Diameter: For each diameter measurement, two readings were made each 90° apart. Number listed is the average of both measurements.

d_1 - measurements made 1 inch from fuel rod top

d_2 - measurements made at center

d_3 - measurements made 1 inch from fuel rod bottom

*** Burrs formed during disassembly filed off prior to making length measurements.

2.2 IRRADIATION OF TWO CAPSULES CONTAINING SAMPLES FROM THE CVTR CORE IN THE WESTINGHOUSE TEST REACTOR (MARCH-JULY 1962)

2.2.1 GENERAL

Capsule irradiations[4] of UO_2 fuel rods were performed to evaluate the effect of fuel rod power level on cladding dimensional changes and fission gas release.

Two capsules, designated A-2 and A-4, were irradiated in the Westinghouse Test Reactor. One capsule contained three fuel rods with a 38 inch fuel length and was irradiated at peak fuel rod power levels of 19 kw/ft to a maximum fuel burnup of 3,450 $\frac{MWD}{MTU}$. The other capsule contained four fuel rods with 6 inch fuel lengths. Peak fuel rod power levels of 22.2 kw/ft were measured with irradiation to 6,250 $\frac{MWD}{MTU}$.

2.2.2 DESCRIPTION OF EXPERIMENT

To evaluate the effect of the initial cold dimetral fuel-to-clad gap on the radial and axial thermal expansion of the UO_2 fuel relative to the cladding, a range of gaps was selected so that high interfacial pressure would exist during operation in some of the fuel rods while a finite hot gap should exist at all times in others when operated at the same power level.

Cold diametral gaps of 0.002, 0.005 and 0.012 inches were used. The Zircaloy-2 cladding dimensions were varied to obtain the desired diametral clearances while keeping the wall thickness constant at 0.032 inches and the UO_2 pellet diameter constant at 0.430 inches.

The various initial cold diametral gaps used in the fuel rods also resulted in different UO_2 fuel surfaces and center temperatures allowing the fractional fission gas release from UO_2 to be measured at different average UO_2 fuel temperatures.

Two capsules, A-2 and A-4, were irradiated in the WTR. Table 3 summarizes the parameters used for the design of the fuel rods for the two capsule irradiations.

A-2 Capsule Design

Capsule A-2 was designed primarily to evaluate UO_2 thermal expansion relative to the Zircaloy-2 cladding with different initial pellet-to-clad gaps. The three fuel rods contained a 38 inch long column of UO_2 pellets. The fuel enrichment was varied along the length of the fuel rods to maximize the length of fuel operating at high temperatures.

Table 3

SUMMARY OF FUEL CAPSULE IRRADIATION EXPERIMENT PARAMETERS

Capsule Number	Fuel Rod Number	Fuel Column Length	Fuel Enrichment	Nominal Initial Diametral Clearance	Peak Fuel Rod Power Level (Actual)		Fuel Rod Nominal O.D.	Maximum Surface Heat Flux at Fuel Rod O.D.
		inches	% U-235	inches	Kw/ft	W/cm	Inches	$\frac{\text{Btu}}{\text{hr ft}^2} \times 10^3$
A-2	2-1	38	Variable	0.002	19.0	624	0.496	498
	2-2	38	Along Length	0.006	19.0	624	0.500	495
	2-3	38	5.7 to 8.5%	0.012	19.0	624	0.506	488
A-4	4-1	6	4.5	0.002	22.2	727	0.496	580
	4-2	6	4.5	0.006	22.2	727	0.500	576
	4-3	6	4.5	0.012	22.2	727	0.506	568
	4-4	6	4.5	0.012	22.2	727	0.506	568

Note: UO_2 pellets 0.430 inch diameter, nominal 94% dense, O/U ratio 2.00 - 2.01
Zr-2 cladding dimensions varied to give initial diametral clearances

All UO_2 pellets used were right circular cylinders 0.430 inches in diameter with a nominal density of 10.3 g/cm^3 (94% of theoretical). The inside and outside diameters of the Zircaloy-2 cladding were varied to obtain the various cold diametral gaps; however, the cladding wall thickness was maintained at 0.032 inches in all cases.

A-4 Capsule Design

Capsule A-4 was designed primarily to evaluate the fission gas release from sintered UO_2 in fuel rods operating with high UO_2 center temperatures.

The various initial cold diametral gaps used would result in different UO_2 fuel surface and center temperatures. The fuel rod configuration used was a Zircaloy-2 tube containing a column of seven UO_2 pellets, 0.860 inches long, giving a fuel length of about 6 inches. A 0.100 inch axial gap was provided in the fuel rods to accommodate any axial expansion of the UO_2 pellet column relative to the cladding.

Capsule A-4 contained four fuel rods with fuel of the same enrichment. The fuel rods had the same initial pellet-to-clad gaps as the A-2 capsule fuel rods.

2.2.3 POST IRRADIATION EXAMINATION

After irradiation the A-2 and A-4 capsule fuel rods were examined in the WTR hot cells.

The capsules were disassembled and the fuel rod samples removed. Diameter and overall length measurements were taken on all the fuel rod samples to detect any deformation of the Zircaloy-2 cladding which may have occurred because of interaction with the UO_2 or because of internal gas pressure as the result of high fission gas releases.

The fuel rods were then punctured and the contained gases collected and analyzed. The fuel rods were sectioned at UO_2 pellet interfaces and photomacrographs taken of the cross sections. Selected entire pellets from each fuel rod were dissolved and the resultant solution diluted for subsequent fission product nuclide analysis to determine fission rates and fuel burnup. Samples of irradiated UO_2 pellets were pressure mounted in an epoxy resin for metallographic examination of the transverse cross sections. These samples were subsequently ground, polished and etched for metallographic examination. The radius at which various microstructural changes such as UO_2 grain growth and columnar grain formation occurred was measured. Photomacrography of the irradiated UO_2 fuel were taken across entire pellet diameters.

Capsule A-2 Fuel Rod Dimensional Measurements

The post irradiation diameters of the three A-2 capsule fuel rods were measured at various positions along the lengths of the fuel rods. The overall fuel rod lengths were also measured. Table 4 summarizes the diameter and length measurements made on the A-2 capsule fuel rods.

Within the accuracy of the measurements no significant diameter or length changes occurred during the irradiation of fuel rods A-2-2 and A-2-3 which had initial cold diametral clearances between the UO_2 pellets and the Zircaloy-2 cladding of 0.006 and 0.012 inches respectively.

In the case of fuel rod A-2-1, which had an initial cold diametral gap of 0.002 inches, fuel rod diameter increases were found near the lower end of the fuel rod and some elongation of the fuel rod had occurred.

The Zircaloy-2 cladding used to fabricate this fuel rod had an inside diameter of 0.432 inches and 0.032 inch nominal wall. The UO_2 pellets used in all fuel rods were 0.430 inches in diameter.

These diameter changes are attributed to deformation of the Zr-2 cladding by the UO_2 pellets as they thermally expand radially against the cladding.

Fuel rods with an initial 0.006 inch cold diametral gap or greater showed no measurable diameter change when irradiated at comparable power levels.

Capsule A-4 Fuel Rod Dimensional Measurements

Table 5 summarized the diameter and length measurements made on the A-4 capsule fuel rods.

Within the accuracy of the measurements no significant dimensional changes were noted except in the case of fuel rod A-4-1 which had an initial 0.002 inch cold diametral pellet-to-clad clearance. The diameter of this fuel rod increased slightly due to the radial thermal expansion of the UO_2 pellets against the cladding.

Collection and Analysis of the Gases Within the Fuel Rods

The gases contained in fuel rods from capsules A-2 and A-4 were collected and analyzed. To determine the volume of the fission gases released from the UO_2 the total volume of contained gases was measured. The distribution of the fission gas isotopes was also determined.

From the data obtained and the total number of atoms of U-235 fissioned, the fractional release of fission gas from the UO_2 fuel was calculated. The range of values obtained for the modified diffusion constant for the A-4 capsule

Table 4

DIAMETER AND LENGTH MEASUREMENTS ON FUEL ROD SAMPLES FROM CVTR CAPSULE A-2

Fuel Rod Number	Initial Cold Diametral Gap (inches)	Diameter Measured (1)	Pre (2) (in.)	Post (2) Precision ± 0.0005 (in.)	Change in Diameter mils	Change in Diameter inches	Fuel Rod Lengths + 0.002: Pre-Irradiation (inches)	Fuel Rod Lengths + 0.002: Post-Irradiation (inches)	Overall Length Change (in.)	Uniform Change: Over Entire Length $\frac{In. \times 10^{-3}}{In. of Length}$	Uniform Change: (3) Over 20-In. of Length $\frac{In. \times 10^{-3}}{In. of Length}$
A-2-1	0.002	d_1	0.4963	0.4958	-0.5	-0.0005	40.943	40.960	+0.017	0.43	0.87
		d_2	0.4960	0.4958	-0.2	-0.0002					
		d_3	0.4958	0.4980	+2.2	+0.0022					
		d_4	0.4943	0.4973	+3.0	+0.0030					
A-2-2	0.006	d_1	0.5019	0.5014	-0.5	-0.0005	40.955	40.959	+0.004	0.09	----
		d_2	0.5020	0.5012	-0.8	-0.0008					
		d_3	0.5021	0.5019	-0.2	-0.0002					
		d_4	0.5014	0.5020	+0.6	+0.0006					
A-2-3	0.012	d_1	0.5066	0.5052	-1.4	-0.0014	40.965	40.971	+0.006	0.15	----
		d_2	0.5066	0.5064	-0.2	-0.0002					
		d_3	0.5061	0.5065	+0.4	+0.0004					
		d_4	0.5051	0.5051	0	0					

(1) d_1 - Measurement made 14 inches from top of fuel rod
d_2 - Measurement made 26 inches from top of fuel rod
d_3 - Measurement made 32 inches from top of fuel rod
d_4 - Measurement made 38 inches from top of fuel rod

(2) For each diameter measurement both pre- and post-irradiation, two readings were taken 90° apart. Number listed is the average of both measurements.

(3) Interaction between UO_2 pellets and Zircaloy-2 cladding in fuel rod A-2-1 was great enough to cause radial cladding deformation over about 20 inches of length.

Table 5

DIAMETER MEASUREMENTS ON FUEL ROD SAMPLES FROM CVTR CAPSULE A-4

Fuel Rod Number	Initial Cold Diametral Gap (inches)	Diameter Measured (1)	Pre (2) (inches)	Post (3) (inches)	Change in Diameter (mils)	Change in Diameter (inches)
A-4-1	0.002	d_1	0.4940	0.4956	+ 1.6	+ 0.0016
		d_2	0.4940	0.4955	+ 1.5	+ 0.0015
		d_3	0.4940	0.4956	+ 1.6	+ 0.0016
A-4-2	0.006	d_1	0.5017	0.5020	+ 0.3	+ 0.0003
		d_2	0.5017	0.5012	- 0.5	- 0.0005
		d_3	0.5017	0.5016	- 0.1	- 0.0001
A-4-3	0.012	d_1	0.5064	0.5064	0	0
		d_2	0.5064	0.5059	- 0.5	- 0.0005
		d_3	0.5064	0.5063	- 0.1	- 0.0001
A-4-4	0.012	d_1	0.5064	0.5067	+ 0.3	+ 0.0003
		d_2	0.5064	0.5054	- 1.0	- 0.0010
		d_3	0.5065	0.5065	+ 0.1	+ 0.0001

(1) d_1 = Measurements made 1 inch from top of fuel rod.
d_2 = Measurements made at center of fuel rod.
d_3 = Measurements made 1 inch from bottom of fuel rod.

(2) Average of all measurements made prior to irradiation.

(3) For each diameter measurement after irradiation, two readings were taken 90° apart. Number listed is average of both measurements.

Precision ± 0.0005 inches

UO_2 pellets is in reasonable agreement with values previously obtained for 94% dense UO_2 from other sources.

Metallography of Fuel

Selected cross sections of UO_2 samples taken from the A-2 and A-4 capsule fuel rods were pressure mounted in an epoxy resin for subsequent metallographic preparation. The radius corresponding to various microstructural features of the UO_2 such as equiaxed grain growth and columnar grain formation was measured to establish radial temperature profile.

No evidence of melting in the UO_2 can be seen in any of the samples irradiated in the A-2 and A-4 capsules.

2.2.4 CONCLUSIONS

The results from the A-2 and A-4 capsule irradiation experiments enable some general conclusions pertaining to the design of sintered UO_2 fuel rods to be made.

1) The results obtained from the examination of the Zircaloy-2 clad UO_2 fuel rods indicate that extended operation at fuel rod power levels of 18-22 kw/ft. can be achieved without failure or fuel rod dimensional changes if the initial fuel-to-clad gap is large enough to accommodate the relative radial expansion of the UO_2 fuel against the cladding. The initial diametral gap between the UO_2 and the cladding of 0.0065 inches selected for Indian Point Unit No. 2 will not result in cladding diameter increases due to thermal expansion of the UO_2.

2) The steady state peak fuel rod power level now considered for Indian Point Unit No. 2, 18.5 kw/ft., is well below those necessary to cause center melting of the UO_2 even with a 112% overpower condition.

2.3 LRD IN-PILE TESTS PROGRAM IN THE SAXTON REACTOR

2.3.1 GENERAL

The purpose of this program has been:

a) To perform in-pile proof tests to verify technical feasibility of prototype designs, materials, and fabrication variables proposed for use in a large plant chemical shim environment, and

b) to perform fuel and cladding experiments aimed at reducing overall fuel cycle and plant costs.

A series of subassemblies, which in most cases represents a combination of these objectives, has been irradiated in the Saxton Reactor. The present status of each experiment and significant results to date are detailed in Table 6.

In the subsequent section the performance of Zircaloy clad fuel rods is examined.[5][6]

2.3.2 DESCRIPTION OF EXPERIMENTS AND POST IRRADIATION EXAMINATION OF HIGH POWER LEVEL ZIRCALOY CLAD FUEL RODS

Table 7 summarizes the Zircaloy portion of the LRD irradiations program. The type of Zircaloy used as cladding material, the peak power level and the peak burnup are reported for each experimental fuel rod.

Zircaloy Clad Fuel Rods From Saxton Modified 3x3 Subassembly No. 503-4-23

Evaluation of the in-pile performance of the Zircaloy clad fuel rods irradiated as part of 3x3 subassembly No. 503-4-23 was completed in the period April-June 1965. Two as-pickled and three autoclaved pre-oxidized Zircaloy clad fuel rods designed to operate at 16 kw/ft. (530,000 BTu/hr-ft^9) were irradiated. The rods operated as part of the Saxton core for a total of approximately 58 effective full power days at a maximum clad surface temperature of 640°F. During this time the rods achieved a burnup of approximately 3000 MWD/MTU.

The main purpose of the experiment was to determine the effect of pre-irradiation surface treatment on Zircaloy-clad fuel rods when exposed to nucleate boiling heat transfer conditions in a chemical shim PWR environment.

The post irradiation examination indicated satisfactory in-pile performance of both the pre-oxidized and as-pickled fuel rods irrespective of surface treatment prior to irradiation. No dimensional changes or abnormalities were observed on the fuel rod surfaces.

Saxton In-Pile Defect Test 3x3 Subassembly No. 503-4-24

Visual examination of the intentionally defected Zircaloy-clad rods was completed at the Post Irradiation Facility hot cells. The examination yielded the following observations:

a) No fretting was observed.
b) Crud deposition was very light.
c) No indication of attack of the cladding or fuel was observed in the vicinity of the defect.

Table 6 - LDR In-Pile Assemblies in Saxton

Ass'y Ident.	Assembly Description	Objective(s) of Test	Date In-Core	Date Out-of-Core	In-Core Location	Estimated Effective Full Power Days Exposure	Est. Peak Burnup (MWD/MTU)	Significant Results	Present Status and Location
503-4-21	Modified Saxton 3x3 Assembly	1. Performance of thin-walled SS clad 2. Performance of furnace pre-oxidized Zr clad in chem shim environment 3. Comparison of crud buildup between Zr and SS	2-63	12-63	Center	190	9500	1. Satisfactory performance of SS clad although subjected to severe local plastic deformation. Fretting due to rod design was observed. 2. Satisfactory performance of Zr clad in chem shim environment. Corrosion and hydrogen pickup equivalent to out-of-pile data.	Stored at WTR until disposal at completion of program
503-4-22	Modified Saxton 3x3 Assembly	Same as above	2-63	12-63	Periphery	190	3400	Same as above	Same as above
503-4-23	Modified Saxton 3x3 Assembly (RCC Test)	1. Proof test static simulated RCC elements in chem shim environment 2. Compare performance of as-pickled and autoclave tested Zr clad rods 3. Performance of Zr with nucleate boiling	1-64	5-64	Center	60	3000	1. RCC elements performance satisfactory; no crud build-up or change in force req'd to withdraw rods. 2. Zr clad rods performance satisfactory. No difference between as pickled and auto-clave tested rods. 3. No fretting 4. Light crud deposition.	Same as above
503-4-24	Modified Saxton 3x3 Assembly (Defect Test)	1. Determine if boric acid in coolant enhances a fuel/water reaction or hydriding on the inside surface of Zr clad as a function of burnup and exposure 2. Compare performance of Zr and SS clad loose-oxide fuel elements 3. Extend exposure on Zr clad in chem shim environment 4. Performance of sensitized 304 SS clad	6-64	10-64	Center	56	2800	1. In-pile performance satisfactory. No indication of waterlogging, etc. 2. No indication of attack or reaction in vicinity of defects 3. No fretting observed 4. Light crud deposition	Stored in Saxton canal. Destructive examination initiated on defected rods.
503-4-25	Modified Saxton 3x3 Assembly (Improved Grid Test)	1. Proof test improved grid design 2. Expose new grid & braze mat'ls to chem shim environment 3. Performance of burnished Zr clad rods 4. Extend exposure on Zr clad rods	10-64 5-65	1-65 7-65	Center Periphery	48 51	3300	1. Satisfactory in-pile performance to date 2. No fretting observed 3. No cracking or attack observed on grid braze or cladding materials 4. Heavy crud deposition following crud test	Operating in peripheral core location
503-9-1	Special 2x2 Assembly	1. Determine fuel and fuel element behavior at high specific ratings	1-64 1-65	1-65 12-65	Periphery Center	166 51	6500	1. Satisfactory in-pile performance to date	Operating in central core location
503-10-1	Special 9x9 Assembly	1. Obtain statistically significant irradiation data to verify prototype designs, materials, and fabrication variables considered for use in large plant chem shim environment (see attachments)	6-64	7-65	Center	155	8000	1. Satisfactory in-pile performance to date	Operating in central core location

Table 7

SUMMARY OF ZIRCALOY CLAD IN SAXTON

Saxton Assembly No.	Assembly Type	No. of Zirc. Rods	Material	Condition	Surface Treatment	Full Power Days Exp.	Peak Power (kw/ft)	Peak Heat Flux (Btu/hr/ft)	Peak Burnup (MWD/MTU)
503-4-21	3x3	4	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	190	13.4	450,000	9,500
503-4-22	3x3	4	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	190	4.5	150,000	3,400
503-4-23	3x3	3	Zr-2 (Ni-free)	Annealed	Autoclave Pre-oxidized	58	16.0	550,000	2,900
"	"	2	Zr-2 (Ni-free)	Annealed	As-pickled	58	16.0	550,000	2,900
503-4-24	3x3	3(a)	Zr-4	10% C.W.	Autoclave Pre-oxidized	56	14.0	465,000	2,900
"	"	1(b)	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	246	13.5	450,000	12,300
"	"	1(c)	Zr-2 (Ni-free)	Annealed	Furnace Pre-oxidized	246	13.5	450,000	12,300
"	"	1(d)	Zr-4	10% C.W.	Furnace Pre-oxidized	56	13.5	450,000	2,800
503-4-25	3x3	1	Zr-4	10% C.W.	Surf. Prep. & Autoclave Preoxidized	99	14.0	465,000	3,300
"	"	1	Zr-4	10% C.W.	As-Surf. Prep.	99	14.0	465,000	3,300
"	"	1(e)	Zr-2 (Ni-free)	Annealed	Furnace	336	13.5	450,000	15,600
"	"	1(f)	Zr-2 (Ni-free)	Annealed	Autoclave	150	16.0	550,000	6,200
503-10-1	9x9	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	12.0	400,000	8,000
"	"	5	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	13.5	450,000	8,000
"	"	3	Zr-2	10% C.W.	Autoclave Pre-oxidized	155	14.0	465,000	8,000
"	"	3	Zr-2 (Ni-free)	10% C.W.	Autoclave Pre-oxidized	155	14.0	465,000	8,000
"	"	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	14.0	465,000	8,000
"	"	3	Zr-4	10% C.W.	As-pickled	155	14.0	465,000	8,000
"	"	3	Zr-4	10% C.W.	Furnace Pre-oxidized	155	14.0	465,000	8,000
"	"	3	Zr-4	10% C.W.	Autoclave Pre-oxidized on O.D. & I.D.	155	14.0	465,000	8,000
"	"	3	Zr-4	10% C.W.	Autoclave Pre-oxidized	155	16.0	550,000	8,000

NOTES:

(a) Contain loose oxide UO_2 fuel.
(b) Previously irradiated in Assembly No. 503-4-21.
(c) Previously irradiated in Assembly No. 503-4-21. (Contains 15 mil dia. defect.)
(d) Contains 15 mil diameter defect.
(e) Previously irradiated in Assembly Nos. 503-4-21 and -24.
(f) Previously irradiated in Assembly No. 503-4-23.

Therefore, even if a defect were to occur no further undesirable problem, such as defect enlargement, clad bursting, etc., is expected.

Saxton Special 9x9 Assembly No. 503-10-1

The operating characteristics of this assembly are shown in Table 7. In-core examination was successfully performed with the aid of a boroscope upon completion of the crud test. The assembly has experienced approximately 8000 MWD/MTU. No indication of failure, cracking, or attack of either the Zircaloy or stainless steel cladding was observed.

Saxton Advanced Fuel Assembly No. 503-4-25

The operating data are shown on Table 7. The present general appearance of the subassembly is satisfactory. No fretting, cracking, or attack of the grid or cladding material was observed.

2.4 SAXTON SPECIAL 2x2 STAINLESS STEEL FUEL RODS SUBASSEMBLY NO. 503-9-1

The special 2x2 subassembly, which has been operating in a peripheral location of the Saxton core since January 1964 was successfully transferred to the central core location. These four fuel rods operated successfully at a peak power rating of 23 kw/ft to a burnup exceeding 6500 MWD/MTU. There was no indication of cracking or swelling of the clad and the overall appearance of the rods was excellent.

2.5 NASA-PLUM BROOK REACTOR-HIGH POWER-HIGH BURNUP-IRRADIATION PROGRAM

UO_2 fuel capsules are being irradiated in the NASA - Plum Brook Reactor as part of the High Power, High-Burnup Irradiation Program.(7) Fuel pins containing 0.300 inch diameter pellets 96% dense with a 6 inch fuel column are clad in Type 304 stainless steel. The capsules are being irradiated at a rating of 20 to 60 kw/ft, to a maximum burnup of 80,000 MWD/MTU. Four capsules have been irradiated to 10,000 MWD/MTU at a peak power rating of 39 kw/ft. Three of these capsules experienced no clad deformation even though they were exposed for a long time at a very high power levels causing large fragmentation in some fuel pellets and center cavities with a diameter as large as 20% of the fuel diameter in others. All of the fuel pellets had experienced center melting.

In the fourth capsule, over 75% of the cross sectional area of the pellets melted due to exposure at extremely high power levels. Part of the clad melted (possibly because molten uranium was momentarily in contact with the cladding). The center of the pellets shifted about 13% of the fuel radius toward the molten clad zone and an internal cavity, whose diameter was about 45% of the fuel diameter was formed. However, no expulsion of uranium into the coolant or excessive clad deformation occurred.

Three capsules were irradiated in the Plum Brook Reactor in a program designed to measure the thermal conductivity of UO_2 at temperatures up to 2300^oC.[8] The UO_2 fuel columns were 4-1/2 inches long and 1-1/4 inches in diameter. They were successfully irradiated at rod powers of 22-25 kw/ft.

2.6 FUEL PIN IRRADIATION IN THE GETR

Four vibratory compacted and two pelleted fuel pins were successfully irradiated in the GETR at peak rod power of 21 kw/ft.[9] The pins were 5.2 inches long, had an active fuel diameter of 0.56 inches and were 304 stainless steel cladded. The pelleted UO_2 was 88.3% dense while the vibratory compacted UO_2 was from 80.4 to 86.7% of theoretical density.

3.0 SUMMARY OF WESTINGHOUSE EXPERIENCE WITH LARGE ZIRCALOY CORES

Table 8 summarizes the excellent behavior of Zircaloy cladding in pressurized-water reactors.

In the CVTR 1,368 Zircaloy clad fuel rods have been operating at a maximum power rating of 14 kw/ft and a maximum burnup of 15,000 MWD/MTU for 450 days with no failure. In August 1965, the reactor was shutdown and subsequently operated at an increased power level (from 43 to 65 MWt). The new maximum design linear power rating is 17 kw/ft.

In the Shippingport Reactor[10] (Core I blanket) 94,920 Zircaloy clad fuel rods have been operating at a maximum power level of ~ 13 kw/ft and a maximum burnup of 36,600 MWD/MTU for about 2200 days with only three pin hole failures from fabrication defects.

Even through these two reactors have not been operating at high peak power rating, they give good statistical indication (large number of fuel rods and high burnup) of the suitability of Zircaloy clad UO_2 fuel elements in Westinghouse Pressurized-Water Reactors.

Table 8

REACTOR EXPERIENCE WITH ZIRCALOY CLAD UO_2 PELLETS

Reactor and Type	No. of Rods	Rod Diam. (in.)	Clad Thickness (in.)	Max. Clad O.D. Surface Temp. (°F)	Max. Heat Flux Btu/hr-ft^2	Peak Power Level kw/ft	Peak Burnup MWD/MTU	Resistance Time In-Pile (days)	No. of Failures	Remarks
CVTR	1,368	0.482	0.025	589	374,000	14	15,000	450	None	Until Aug. 1965
Shippingport (Core I Blanket)	94,920	0.411	0.022	630	343,000	13	36,600	2,200	3	Pinholes from fabrication defects

4.0 URANIUM DIOXIDE THERMAL CONDUCTIVITY

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The absolute value of the temperature distribution is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

Figure 1 summarizes the valid laboratory and in-pile data and shows the Westinghouse design curve for the UO_2 thermal conductivity versus temperature. The recommended design curve between 0^oF and 3000^oF is based on the data of Godfrey[11], which are the results currently available. Data obtained at Chalk River[12], Hanford[13], and Harwell[14] indicate that the thermal conductivity of UO_2 can be depressed up to 50% by irradiation at low temperatures. This decrease occurs only at temperatures below $900-1100^oF$, however, and is annealed out at higher temperatures. Since the surface temperatures of the peak fuel pellets are approximately 1100^oF, the design curve is valid for both irradiated and unirradiated fuels.

The section of the curve between 3000^oF and melting (5072^oF) is based on two factors:

1. In-pile observations of fuel melting indicate a positive temperature coefficient for conductivity above approximately 3000^oF. The temperature dependence in this range should conform to an exponential curve since this reflects a credible physical interpretation of the high temperature conductivity increase.[15] As the figure shows, the design curve is somewhat conservative in this respect, falling between the extremes reported by other investigations, notably the Chalk River and G. E. curves.
2. The area under the recommended curve corresponds to an $\int kdT$ of approximately 97 w/cm as deduced from in-pile studies of grain growth and center melting by Robertson, et.al.[16] and Duncan.[3] This value is based upon the interpretation of fuel melt radius as determined at Hanford[17] and Chalk River.[16]

Thermal conductivity can best be represented by the following equations:

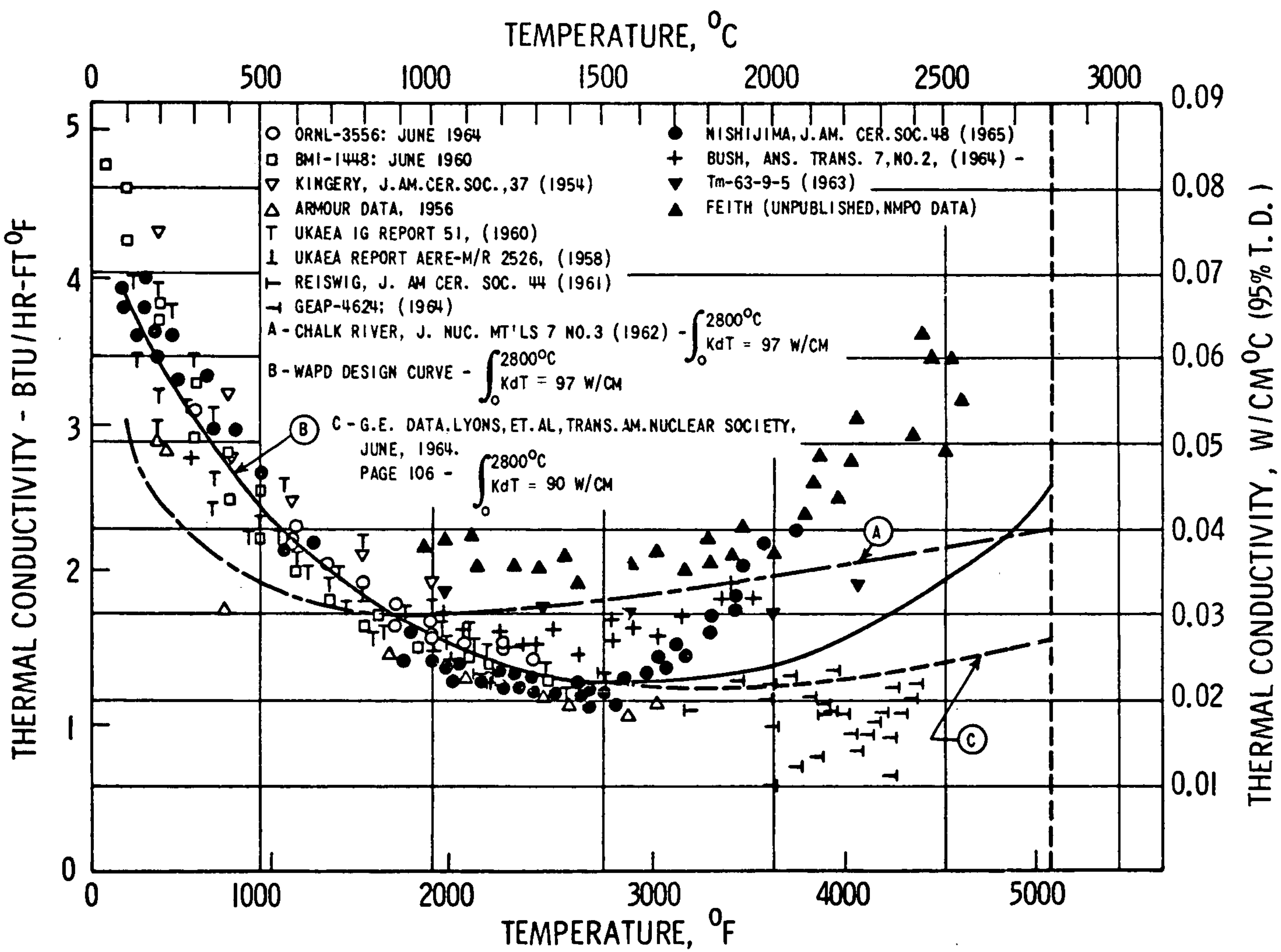
(a) Temperature Range - $0 \leq T \leq 1650^{o}C$

$$K = \frac{40.4}{464 + T} + 1.32 \times 10^{-4} e^{1.88 \times 10^{-3} T}$$

(b) Temperature Range - $1650^{o}C < T \leq 2800^{o}C$

$$k = 0.019 + 1.32 \times 10^{-4} e^{1.88 \times 10^{-3} T}$$

with k in w/cm^{o}C for 95 per cent dense UO_2 and T in ^{o}C.



THERMAL CONDUCTIVITY OF URANIUM DIOXIDE

Figure 1

5.0 APPLICATION OF THE WESTINGHOUSE EXPERIENCE TO THE INDIAN POINT UNIT NO. 2 DESIGN

The experiments reported in Section 2 have confirmed the criteria followed in the design of the various test fuel rods and have supplied further information to permit the design to be improved.

The behavior of Zircaloy clad fuel rods was tested under many operating conditions (different power levels, cladding thicknesses, pellet-clad gaps, fuel enrichments, UO_2 densities, degrees of cold work, water chemistry, etc.) to obtain a high degree of confidence on the operating limits of Zircaloy clad fuel rods.

The operation of the Shippingport reactor and the CVTR give good statistical indication (large number of fuel rods and high burnup) of the suitability of Zircaloy clad UO_2 fuel elements in Westinghouse Pressurized Water Reactors.

As mentioned in Section 1, Introduction, present day practice in the design and operation of ceramic fueled cores generally limits the maximum power density to prevent central melting in fuel pellets.

The power level necessary to reach center melting at various burnups was calculated and is shown in Figure 2. This evaluation is based on the reduction of melting point due to burnup[1], the UO_2 thermal conductivity curve (as shown in Figure 1) corrected for 93% UO_2 theoretical density, flux depression factors for the lowest U-235 enrichment, constant fuel surface temperature of 1100°F, and the preliminary Indian Point Unit No. 2 mechanical design.

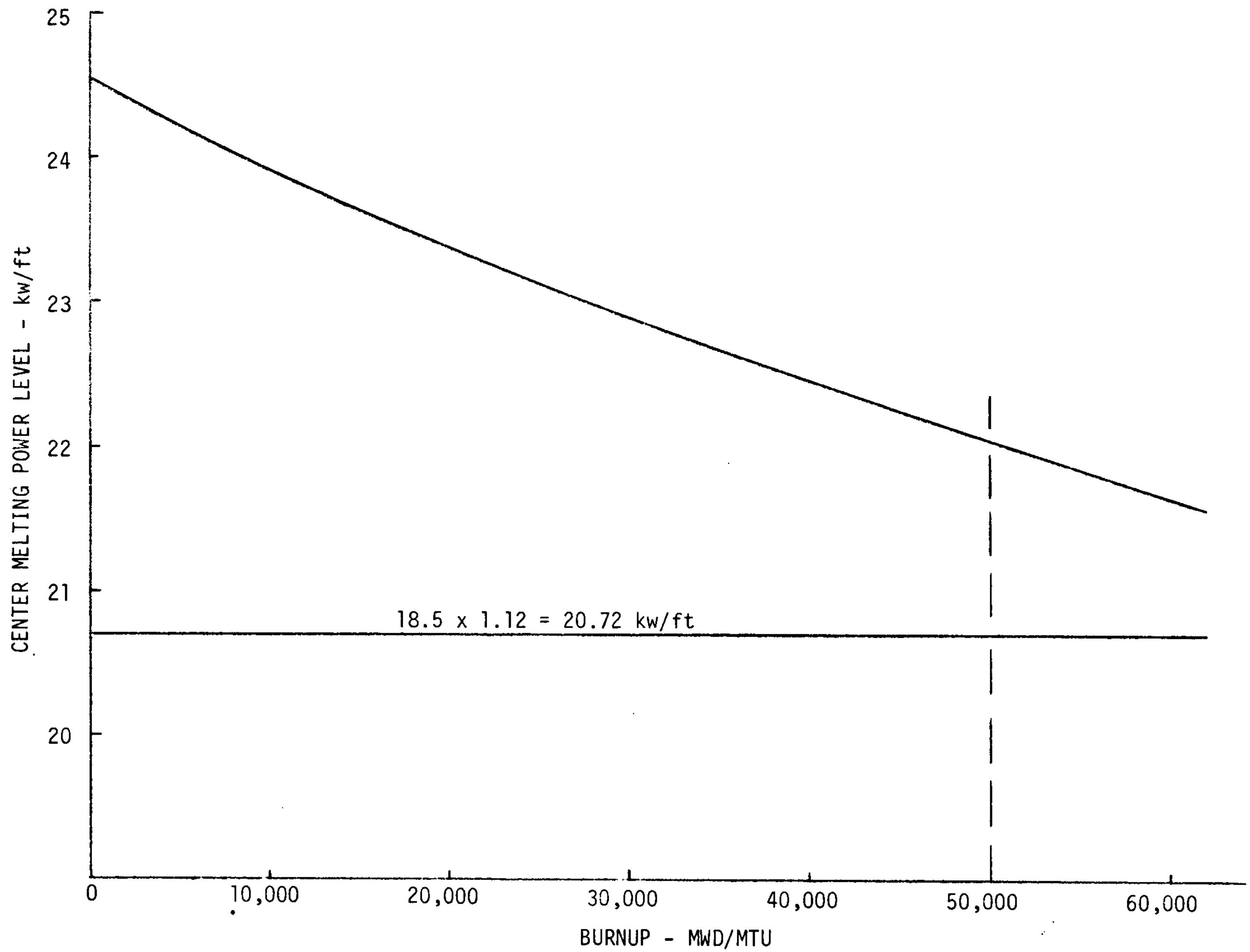

CENTER MELTING POWER LEVEL - kw/ft
25
24
23
22
21
20
18.5 x 1.12 = 20.72 kw/ft
0
10,000
20,000
30,000
40,000
50,000
60,000
BURNUP - MWD/MTU

Figure 2

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