C O P Y

CONSUMERS POWER COMPANY
General Offices - Jackson, Michigan

July 29, 1966

(suppl)
File Copy

Dr. R. L. Doan, Director
Division of Reactor Licensing
United States Atomic Energy
Commission
Washington, D. C. 20545

Re: Docket 50-155

Dear Dr. Doan:

Attention: Mr. Roger S. Boyd

Transmitted herewith are three (3) executed and nineteen (19) conformed copies of a request for a change to the Technical Specifications of License DPR-6, Docket No. 50-155, issued to Consumers Power Company on May 1, 1964, for the Big Rock Point Nuclear Plant.

This proposed change (No. 10) will enable Consumers Power Company to insert Reload "C" fuel into the Big Rock Point reactor. This fuel will incorporate vibratory compacted $UO_2$ powder but is otherwise physically identical to the previous reload fuel.

Shutdown of the plant for refueling is scheduled presently for around September 5, 1966.

Yours very truly,

Robert L. Haueter (Signed)

RLH/wf/mp
Attach.

Robert L. Haueter
Assistant Electric Production
Superintendent - Nuclear

ACKNOWLEDGED





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CONSUMERS POWER COMPANY Suppl

Docket No. 50-155

Request for Change to the Technical Specifications

License No. DPR-6

For the reasons hereinafter set forth, it is requested that the Technical Specifications of License DPR-6 issued to Consumers Power Company on May 1, 1964, for the Big Rock Point Nuclear Plant, be changed as follows:

I. Section 5

A. In Section 5.1.5, change "(a)" to read as follows:

"(a) Enrichment of Fuel, approximate weight percent U-235 from 2.6 to 5.2, inclusive."

Under "(c) Fuel Bundles," change "$UO_2$ Density, Percent Theoretical" for the reload fuel to read as follows:

"Pellet Fuel - 94 ± 1

Powder Fuel - Approx 85"

B. Add a new section - 5.1.8:

"5.1.8 Thin Clad Powder Fuel Bundle

Two of the Reload "C" fuel bundles may contain standard rods with Zr-2 cladding of 0.025" thickness; otherwise, they will be the same as the remaining Reload "C" fuel bundles."

II. Discussion - Reload "C" Fuel

The proposed changes in Section 5.1 will enable Consumers Power Company to refuel the Big Rock Point reactor with vibratory compacted $UO_2$ powder fuel. Experience has shown powder fuel performance to be at least equal to pellet fuel performance, but with significant potential economic gain.

At the upcoming refueling, it is planned to insert the remaining Reload "B" fuel and a portion of the Reload "C" fuel

A. Fuel Description

The Reload "C" fuel is similar physically to the first reload fuel (Reload "B") except that vibratory compacted $UO_2$ powder will be used in place of the $UO_2$ pellets. The basic fuel bundle design remains the same - same cage, same fuel cladding material and dimensions, same spring clip spacers, etc. The Reload "C" fuel bundle is shown on Figure 1 and described in Table 1. The powder fuel was designed to the same cladding stress criteria as the pellet fuel. All of this fuel will have a cobalt rod at each of the four corners of the fuel bundle.

Included with the "C" fuel bundles are two pilot bundles with thinner cladding. They are the same physically as the "C" bundles except for this thinner cladding and associated increased fuel diameter. This will result in a slight increase in fuel weight and a slight decrease in water-to-fuel volume ratio. Central fuel temperature at the maximum licensed heat flux will remain the same. The cladding thickness has been reduced from 0.034" to 0.025". (See Table 1.) Only the large rods in the bundles have been changed; the small corner rods will remain unchanged.

The clad was designed as a self-supporting tube using standard design criteria but with the stress intensity limits raised to a higher fraction of the ultimate strength. Continuing study of the stress system on fuel cladding indicates that all stresses are being taken into account so that lower design margins are permissible. Use of higher stress intensity limits results in a reduction in clad thickness for prescribed operating conditions and associated savings in the cost of Zircaloy material for a fuel rod.

The current stress intensity limits were established on the assumption that failure of the clad would occur if the ultimate strength of the clad were exceeded. Recent examinations show appreciable ductility for Zr-2 cladding irradiated up to fast flux exposures of about $1.5 \times 10^{21}$ nvt greater than 1 Mev. Although greater fast flux exposures will be experienced in typical fuel clad applications, irradiation effects on ductility appear to saturate at lower exposures of about $0.5 \times 10^{21}$ nvt greater than 1 Mev. The ductility of the clad means that clad collapse onto the fuel, rather than rupture, will occur should the strength be exceeded by the external compressive forces which are limiting. Clad collapse onto the fuel is not expected to result in clad perforation. Internal pressure is maintained below limits by providing plenum space for fission gases.

It is expected that further reduction in design margin and clad thickness will be possible. These two pilot bundles are an intermediate step in conservatively evolving realistic design bases.

B. Fuel Thermal Data

The thermal conductivity data and the heat transfer coefficient between fuel and clad for the "C" powder fuel are the same as previously described for ten pilot bundles included in the Type III-f reload fuel for Cycle 4 in the Dresden Nuclear Power Station. The previous thermal data were obtained from irradiations performed under the High Performance $UO_2$ Program sponsored by the Joint U.S.-Euratom Research and Development Program. The $UO_2$ powder conductivity data, as submitted for the Dresden reload pilot bundles, yield an integral KdT from the fuel surface temperature (500° C by definition) to the melting temperature (2800° C) of 49 w/cm. (This was reported as preliminary data in the Dresden submittal.) In the summary report of the High Performance Program (GEAP-5100-1, "$UO_2$ Powder and Pellet Thermal Conductivity During Irradiation," by M. F. Lyons, et al, March 1966), the value of the integral as used for the Dresden powder bundles was confirmed. Based on post-irradiation analysis, Lyons, et al, recommended an integral value from surface to melt (as defined above) $\cong$ 49 w/cm. (See Figure 2.)

The establishment of an appropriate thermal conductivity curve for powder $UO_2$ is a deduction from this. The approach used to establish the working curve for the Dresden and Big Rock Point powder fuel is based on the following line of reasoning. During the first few minutes of operation, the powder conductivity will be uncertain and probably change rapidly due to sintering of powder above the $UO_2$ sintering temperature ($\sim$1600° C) and densification. However, as these processes continue, the conductivity of the fuel above the temperature for the onset of grain growth will become very close to that determined earlier for pellets. The remaining differences in conductivity and in the conductivity integral values from surface to melting between powder and pellets are attributable to the poorer conductivity of the unsintered powder rim operating below the grain growth temperatures. Working backwards from the previously established pellet $UO_2$ conductivity curve, the powder conductivity was assumed to be identical, at the melting point temperature, to that for pellets. Below this temperature, the powder curve was assumed to gradually and smoothly fall below that for pellets. The deviation between the two curves was adjusted so that, upon reaching reasonable value for the surface temperature ($\sim 500^\circ$ C), the difference in area under the two curves equals the difference in the integral values from surface to melting. Below this temperature, the curve was simply extended smoothly to provide a reasonable mean through the out-of-pile data. (See Figure 3.)

The powder fuel has a higher $UO_2$ to clad gap conductance than pellet fuel. This tends to counteract the lowered $UO_2$ conductivity. In this particular instance, the two effects cancel. (See Table 2.)

C. Fuel Physics Data

The characteristics of the "C" fuel with annular cobalt target rods in each of the four corners of the bundle have been calculated and compared to the "B" or pellet fuel.

1. Reactivity ($k_\infty$)

| Temperature | "B" Fuel Without Cobalt | "B" Fuel With Cobalt | "C" Fuel With Cobalt |
|---|---|---|---|
| 68° F, Zr Channel | 1.275 | 1.215 | 1.244 |
| 572° F, Zr Channel | 1.303 | 1.241 | 1.272 |
| 572° F, Zr Channel 20% Steam Void | 1.296 | 1.231 | 1.256 |

2. Moderator Temperature Coefficient ($\Delta k_{eff}/k_{eff}$ per °F in Zr Channel at 77° F)

| | "B" Fuel Without Cobalt | "B" Fuel With Cobalt | "C" Fuel With Cobalt |
|---|---|---|---|
| Start of Cycle | $+3.2 \times 10^{-6}$ | $+8.3 \times 10^{-6}$ | $+2.6 \times 10^{-6}$ |
| End of Cycle | $+5.5 \times 10^{-5}$ | $+5.1 \times 10^{-5}$ | $+4.9 \times 10^{-6}$ |

3. Void Coefficient ($\Delta k_{eff}/k_{eff}$ per Unit Void Within the Channel)

| | "B" Fuel Without Cobalt | "B" Fuel With Cobalt | "C" Fuel With Cobalt |
|---|---|---|---|
| Cold | -0.04 | -0.06 | -0.06 |
| Hot | -0.09 | -0.09 | -0.08 |

4. Doppler Coefficient ($\Delta k_{eff}/k_{eff}$ per °F)

| Fuel Temperature | Moderator | "A" Fuel (SS Clad) | "B" Fuel With or Without Cobalt | "C" Fuel With Cobalt |
|---|---|---|---|---|
| 68° F | 68° F, 0 Voids | $-1.47 \times 10^{-5}$ | $-1.4 \times 10^{-5}$ | $-1.2 \times 10^{-5}$ |
| 1323° F | 550° F, 0 Voids | $-1.03 \times 10^{-5}$ | $-1.1 \times 10^{-5}$ | $-1.0 \times 10^{-5}$ |
| 1323° F | 550° F, 20% Voids | $-1.15 \times 10^{-5}$ | $-1.4 \times 10^{-5}$ | $-1.1 \times 10^{-5}$ |

It can be seen from the above comparison that the moderator temperature and void coefficients for the "C" fuel are not significantly different from those for the "B" fuel. Although the "C" fuel has a slightly lower Doppler Coefficient than the "B" fuel, it is very similar to the initial ("A") fuel load of stainless steel clad fuel at all conditions except the cold, zero-power case.

The enrichment of the "C" fuel has been increased to partially offset the loss of reactivity due to the cobalt targets. As seen above, the reactivity lies between "B" fuel without cobalt and "B" fuel with cobalt.

5. Control Rod Worth

Control rod worth for the "C" fuel is slightly less than that for the "B" fuel containing cobalt. This is due primarily to the increased enrichment of the "C" fuel which is designed to attain a discharge exposure of 15,000 Mwd/T of uranium with cobalt targets in place for three quarters of the design life. (The cobalt will be replaced by standard $UO_2$ fuel rods of suitable enrichment to control power peaking.)

D. Operational Safety for Powder Fuel

Factors relating to the operational safety of $UO_2$ powder fuel in power reactors include those which can cause cladding failures, propagate existing failures or limit the performance of the plant itself.

Mechanical and chemical interaction between $UO_2$ fuel and clad, if not properly controlled, can result in failure of the cladding.

Propagation of a failure by such phenomena as waterlogging, gross oxidation of the $UO_2$, or internal hydriding of Zircaloy cladding, are no more severe for $UO_2$ powder fuel than for $UO_2$ pellet fuel.

1. $UO_2$-Clad Mechanical Interaction

The temperature rise and the thermal expansion of the $UO_2$ are much greater than those of the cladding when a $UO_2$ fuel rod is heated from ambient temperature to steady-state full power operating conditions. The degree of mechanical contact between the fuel and the cladding depends upon the type of fuel (pellet or compacted powder), the initial fuel-cladding gap, the fuel rod thermal rating, the fuel exposure and the type of cladding material. The mechanical contact between the fuel and the cladding can vary from zero to a point where both diametral and axial plastic strain of the cladding occur.

The premise that fuel-cladding mechanical interactions can be eliminated completely by the use of freestanding cladding with a nominal clearance between the pellets and the cladding is not valid.

A significant aspect of recent measurements is that circumferential ridging occurs in all pellet fuel rods of current designs, even though the cladding is freestanding with diametral gaps of up to 0.013" between the pellets and the cladding. Measurements made on various types of freestanding cladding, including Zircaloy-2, stainless steel, Incoloy and Inconel, show ridging at the pellet interfaces when operated at nominal power levels. Ridging of freestanding, Zircaloy-2-clad fuel rods operated for a short time in the Consumers' Big Rock Point reactor has been measured. The ridges are visually accentuated by selective crud deposition on the cladding surface. The magnitude of the ridging appears to be a function of initial cold gap and fuel rod specific power. Ridges of up to 0.001" in height have been measured in 0.425" OD, Zircaloy-2-clad fuel rods having a wall thickness of 0.030", an as-fabricated pellet-cladding gap of 0.007", and operated at a peak surface heat flux of up to 350,000 Btu/hr-ft$^2$.

No comparable ridging has been observed on fuel rods containing compacted $UO_2$ powder fuel.

A comparison of the thermal expansion of compacted powder fuel with the thermal expansion of small diametral gap pellet fuel indicates that the diametral strain from compacted powder fuel is about one third less than the strain at midpellet and about 40% less than the maximum strain at the pellet interfaces.

2. $UO_2$-Clad Chemical Interaction

There have been no confirmed cladding failures related to impurities in either pellet or powder fuel in which the fuel material has been within the allowable impurity levels. However, clad failures have occurred in both pellet and compacted powder fuel rods in which excessive amounts of impurities have been present. During early tests of Zircaloy-2-clad pellet fuel rods in the VBWR, presence of up to 1,000 ppm of fluoride in the $UO_2$ pellets resulted in severe cracking of the cladding when water entered the fuel rod through a cladding defect. In the PRTR program at Battelle Northwest Laboratories, failures in the Zircaloy-clad compacted powder fuel rods have been attributed to the presence of fluoride and other contaminants in the powder.

3. Structural Changes During Irradiation

Although there are differences in the physical characteristics of as-fabricated sintered pellets and compacted powder, they tend to be eliminated or diminished in magnitude as irradiation proceeds. The higher the thermal conditions or fuel operating temperatures, the more alike the two fuel types become as exposure proceeds. Powder fuel becomes essentially identical with pellet fuel when operated above about 1600° to 1800° C (the temperature range above which sintering, grain growth, void formation and void migration begin to occur). There is experimental evidence that powder fuel irradiated at bulk temperatures below sintering temperatures (1600° C) undergoes a phenomenon in which the mechanically pressed particles are more firmly bound together. This is attributed to "fission-sintering" in which very high local temperatures are achieved as a result of fission spikes.

As pellet fuel is subjected to high thermal stresses during irradiation, the pellets tend to crack into smaller pieces in the direction of compacted powder fuel. The powder fuel which starts out as a mechanically bonded compact of small particles tends to become a more cohesive body in the direction of pellet fuel. Thus, the morphologies of the two fuel types tend to become more alike as irradiation proceeds.

Initially, there had been some concern that, when relatively low-density (80% to 90% TD) compacted powder fuel rods were subjected to high-temperature irradiation, the powder would densify into a smaller diameter mass. However, compacted powder fuel rods behave the same as pellet fuel with respect to densification during irradiation. As grain growth and void migration occur in the central fuel portion operating above about 1600° to 1800° C, the voids migrate toward the hotter central region of the fuel resulting in a central void surrounded by a densified annular region. At typical BWR fuel and coolant conditions, the outer rim of pellet or compacted powder fuel never operates at temperatures high enough for structural changes to occur other than the fission-sintering.

4. Propagation of Failures

In the operation of intentionally defected pellet and compacted powder BWR type rods, there do not seem to be any significant differences in the performance characteristics of the two types of fuel with respect to fuel washout, $UO_2$ coolant reaction, or susceptibility to waterlogging.

E. Background of Experience With Powder Fuel

Process development and irradiation testing have been conducted as part of many programs to reduce fuel fabrication costs and to investigate alternate fabrication methods showing promise of low fabrication costs and potential improvement in fuel performance. Of these, the powder compaction process shows the greatest potential. The relative fabrication cost economics of compacted powder fuel has been the subject of frequent reviews. It appears to be favorable. Fuel rods made by powder compaction techniques were first tested by General Electric Company, APED, in the Vallecitos boiling water reactor (VBWR). They are now under irradiation in the Big Rock Point, Dresden and JAERI reactors. Compacted powder fuel rods comprise a large portion of the plutonium recycle test reactor (PRTR) core at the Battelle Northwest Laboratories. Other tests of this fuel concept are being conducted by Bettis, Westinghouse (Saxton), ORNL, Combustion Engineering and APED (EVESR superheat).

Consumers Power Company has participated in the High Power Density Fuel Development Program which has as one of its primary goals the development and irradiation testing of new and/or improved processes for fabrication of $UO_2$ fuel for power reactors. Compacted powder $UO_2$ fuel has been irradiated in the VBWR and the Big Rock Point reactor as part of the HPD program. In addition, many developmental fuel bundles containing compacted $UO_2$ powder fuel are currently being irradiated in various GE BWRs. Table 3 summarizes some of this experience with compacted powder fuel rods. No verified failures have occurred in any of the developmental bundles being irradiated in power reactors except for failures of HPD fuel rods in the VBWR and some GETR loop tests.

The HPD program 304-SS clad powder $UO_2$ fuel was fabricated by various powder compaction techniques such as one-, two- and three-pass swaging, hot swaging, tandem rolling and vibratory compaction. Approximately 150 powder fuel rods were irradiated in the VBWR until its shutdown in December 1963. Exposure achieved by the powder fuel was approximately:

*9100 Mwd/T peak for cold swaged powder fuel,
7350 Mwd/T peak for hot swaged powder fuel,
9500 Mwd/T peak for tandem rolled powder fuel and
8000 Mwd/T peak for vibratory compacted powder fuel.

Early in the HPD program, high-temperature erosion stability tests were performed in conditions simulating boiling water reactor environment and in the VBWR utilizing fuel rods with intentional defects. The out-of-pile tests resulted in no significant loss of $UO_2$ from cold swaged (~92% TD) or hot swaged (~95% TD) specimens and variable loss (<50 mg to 1-1/2 grams in 60- to 70-gram specimens) of $UO_2$ from tandem rolled (~88% TD) and vibratory compacted samples (65% TD and 92% TD). No unusual swelling of the $UO_2$ was apparent in any of the compacted powder specimens. In-reactor tests in VBWR utilizing compacted powder with densities ranging from 90% TD (cold swaged) to 95% TD (hot swaged) resulted in no loss of $UO_2$ by erosion and no $UO_2$ swelling.

*Average exposure is lower than the peak value by a factor of about 1.55.

Three in-service failures related to the stress-assisted intergranular corrosion of 304-SS occurred in compacted powder fuel rods. Post-irradiation examination of these fuel rods has verified the satisfactory performance of defected $UO_2$ powder fuels in reactor service. A thin clad (10 mil SS) cold swaged powder rod was operated in the VBWR for approximately 300 hours at 1000 psi, 545° F, with an 8" long longitudinal defect. Although the $UO_2$ was completely exposed to the reactor coolant, no significant amount of the 93.1% TD compacted powder was washed out. Operation of a 16 mil clad rod in the same assembly with a less severe defect resulted in no loss of the $UO_2$.

A failed 86.8% TD vibratory compacted powder fuel rod had been exposed to flowing steam and water in VBWR for at least 72 hours. Again, no significant loss of $UO_2$ was observed.

In all cases, there was no evidence of "waterlogging" or swelling of the $UO_2$. Dial gauge and profilometer dimensional measurements confirmed these results.

The good dimensional and chemical stability of the powder $UO_2$ fuel, when exposed to flowing steam and water, is attributed to in-reactor bonding between the powder $UO_2$ particles. This is due to the combined effects of thermal sintering and fission sintering.

## Experience at Other Sites

The most extensive testing and application of compacted powder fuel has been associated with the PRTR program at Battelle Northwest Laboratories.

Since start-up of the PRTR in July 1961, a total of 66 $UO_2$ and 90 $UO_2$-$PuO_2$ 19-rod fuel bundles have been irradiated in the PRTR. These, plus a small number of other experimental elements, comprise approximately 4,400 individual full-length rods.

The current PRTR fuel exposure status is:

| Type of Fuel | Average Exposure of Leading Bundle (Mwd/T) |
|---|---|
| $UO_2$-$PuO_2$ (Vibratory Compacted) | 8300 |
| $UO_2$ (Vibratory Compacted) | 6100 |

There have been over 35 fuel rod failures in the PRTR. All of the failures have been attributed to deficiencies in the fuel processing such as: fluoride impurities in plutonium, gas phase hydriding in susceptible cladding regions caused by irradiation decomposition of residual water in the fuel and hydrocarbon impurities, or both. No failures can be attributed to inherent problems with the design of fuel rods utilizing $UO_2$ powder fuel.

Oak Ridge has conducted irradiation tests of 26 vibratory compacted $ThO_2$-$UO_2$ capsules containing powder made by arc fusing and by the sol-gel process. The peak burn-up achieved was approximately 81,000 Mwd/T in a 10" long stainless steel clad fuel specimen containing 85% TD $ThO_2$-$UO_2$.

Most of the ORNL irradiations were conducted in the NRX and MTR low-temperature process water with cladding temperatures of about 100° C. Therefore, these are tests of the fuel meats and fuel-cladding interactions rather than cladding-environment tests. No failures and no significant changes in dimension of the irradiated specimens were found.

## III. Hazards Considerations

The Reload "C" fuel bundles described above utilize the same hardware as the Reload "B," the Phase I and the Phase II R&D fuel bundles. This hardware continues to give excellent performance in the Big Rock Point reactor.

A great deal of experience with powder fuel is accumulating as discussed above. Based upon this experience, we believe that powder fuel is now a commercially acceptable fuel design and that its performance should be as good as, if not better than, pellet fuel.

The nuclear characteristics of the Reload "C" fuel are not significantly different from previous fuels, and its performance under normal and transient conditions should be comparable. The thermal-hydraulic performance will be identical to the Reload "B" fuel.

Based upon the above considerations, we have concluded that use of Reload "C" fuel in the Big Rock Point reactor does not present a significant change in the hazards considerations described or implicit in the Final Hazards Summary Report.

CONSUMERS POWER COMPANY

By H R Wall
Vice President

Date: July 29, 1966

Sworn and subscribed to before me this 29th day of July 1966.

Grace R Warner
Notary Public, Jackson County, Michigan
My commission expires February 16, 1968

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TABLE 1
Reload "C" Fuel Data Sheet

| Fuel Rod, Cold | | | | | |
|---|---|---|---|---|---|
| Rod Type (See Figure 1) | 1 | 2 | 3 | C | Thin Clad |
| Fuel Diameter, Inches | 0.381 | 0.381 | 0.282 | - | 0.399 |
| Cladding Thickness, Inches | 0.034 | 0.034 | 0.031 | 0.031 | 0.025 |
| Cladding Outside Diameter, Inches | 0.449 | 0.449 | 0.344 | 0.344 | 0.449 |
| Active Fuel Length, Inches | 70 | 70 | 70 | - | 70 |
| Fuel Material | $UO_2$ | $UO_2$ | $UO_2$ | - | $UO_2$ |
| Cladding Material | Zr-2 | Zr-2 | Zr-2 | Zr-2 | Zr-2 |
| Number of Rods per Bundle | 37 | 72 | 8 | 4 | 109* |
| Enrichment, W/O U-235 | 5.2 | 2.9 | 2.9 | - | * |

| Fuel Bundle | |
|---|---|
| Number of Fuel Bundles | 40 |
| Fuel Rod Array | 11 x 11 |
| Weight $UO_2$ per Bundle, Lb | 305 |
| Water-to-Fuel Volume Ratio | 2.6 |

* 37 at 5.2 W/O U-235
72 at 2.9 W/O U-235
109 Rods Total

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TABLE 2

Thermal Performance Comparison of Pellet and Powder Fuel

| | *Pellet | *Powder |
|---|---|---|
| Fuel Diameter, Inches | 0.373 | 0.381 |
| Cladding Thickness, Inches | 0.034 | 0.034 |
| Cladding Diameter, Inches | 0.449 | 0.449 |
| Fuel Density, % Theoretically | 95 | 85 |
| Nucleate Boiling Heat Transfer Coefficient, Btu/Hr-Ft$^2$-$^\circ$F | 10,000 | 10,000 |
| Heat Transfer Coefficient Between Fuel and Cladding, Btu/Hr-Ft$^2$-$^\circ$F | 1,000 | 3,000 |
| $UO_2$ Conductivity Integral $\int_{T = 500^\circ C}^{T = 2800^\circ C} KdT$, W/Cm (See Figure 2) | 59 | 49 |
| Incipient Melting Temperature of $UO_2$, $^\circ$F | 5,080 | 5,080 |
| Heat Flux for Incipient Melting, Btu/Hr-Ft$^2$ | 550,000 | 550,000 |
| Fuel Linear Heat Generation, Kw/Ft (For Incipient Melting) | 19 | 19 |

*Data are given for large fuel rods only as they are most limiting from fuel temperature considerations.

TABLE 3

GE-APED Compacted Powder Developmental Fuel Irradiation Experience

| Reactor | No. of Rods | Clad Material | Clad OD (Inch) | Clad Thickness (Inch) | $UO_2$ Density (% TD) | Peak Q/A (Rated Power) | Avg Burn-Up of Leading Bundle Mwd/T | Approximate Peak Burn-Up Mwd/T | Remarks |
|---|---|---|---|---|---|---|---|---|---|
| Tests: | | | | | | | | | |
| VBWR | 150 | 304-SS | 0.400 to 0.565 | 0.005 to 0.016 | 83 to 95 | Up to 527,000 | 6,000 | 9,100 | 3 Failed Rods |
| VBWR | ~210 | 304-SS | 0.250 | 0.028 | 53 to 67 | | 20,000 | 30,000 | |
| GETR-PWL | 18 | Zr-2 | 0.560 | 0.030 | | Up to $1.4 \times 10^6$ | 20,000 | 30,000 | 3 Failed Rods |
| Power Reactor Demonstration: | | | | | | | | | |
| Dresden | 350 | Zr-2 | 0.5625 | 0.035 | 84.7 | 330,000 | 5,700 | 8,550 | |
| JPDR | 72* | Zr-2 | 0.557 | 0.030 | ~90.0 | 300,000 | 2,500 | 3,750 | |
| Consumers Big Rock Point | 363 | Zr-2 | 0.425 | 0.030 | 85 | 340,000 | 6,200 | 9,150 | |
| | 484 | 304-SS | 0.425 | 0.010 | 91 | 384,000 | 7,680 | 11,200 | |
| | 726 | Incoloy-800 | 0.425 | 0.011 | 91 | 430,000 | 6,400 | 9,450 | |

*Two segments per rod.

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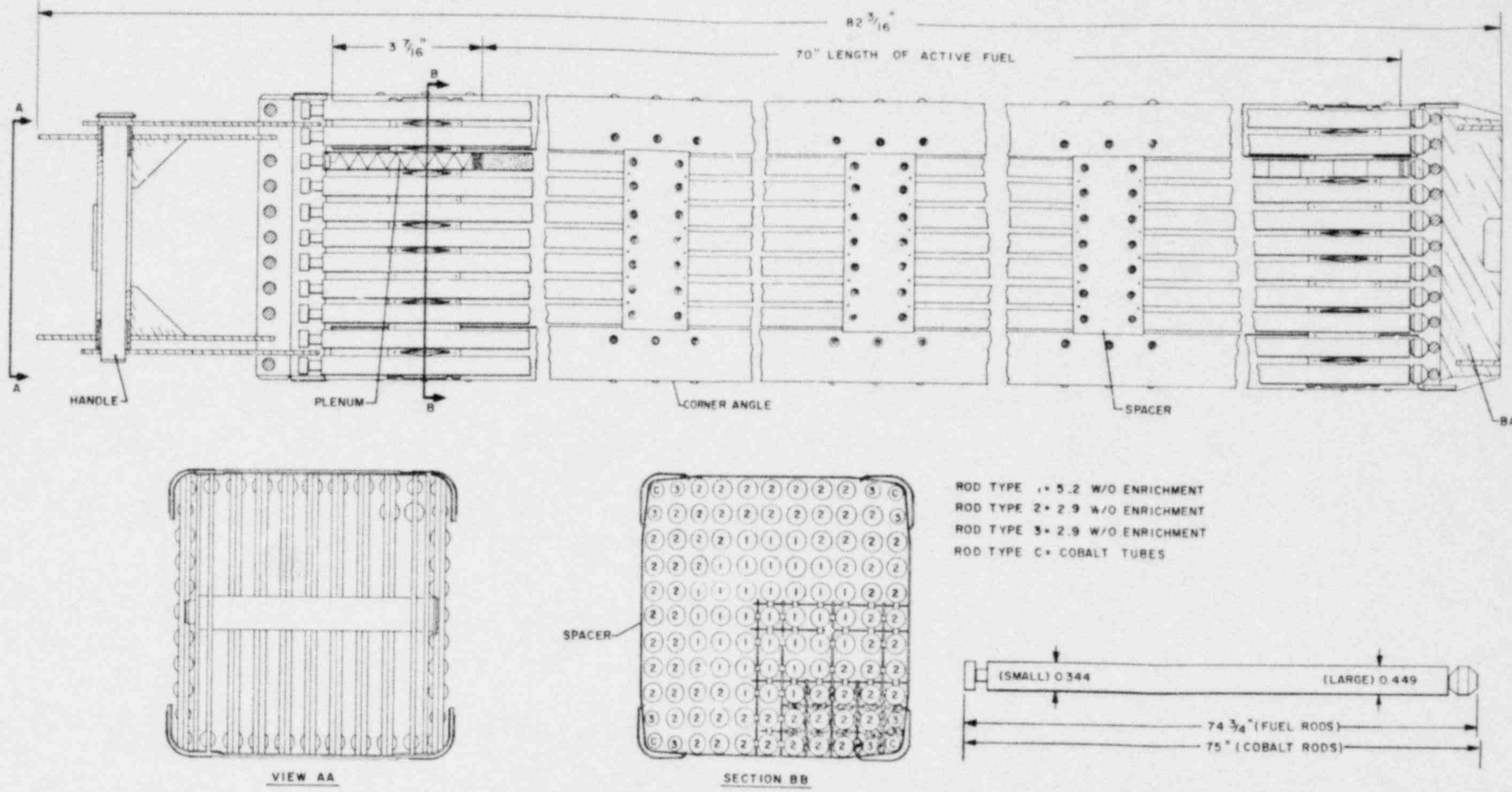


FIGURE I BIG ROCK POINT RELOAD (B R P 3 OR TYPE C)

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FIGURE 2

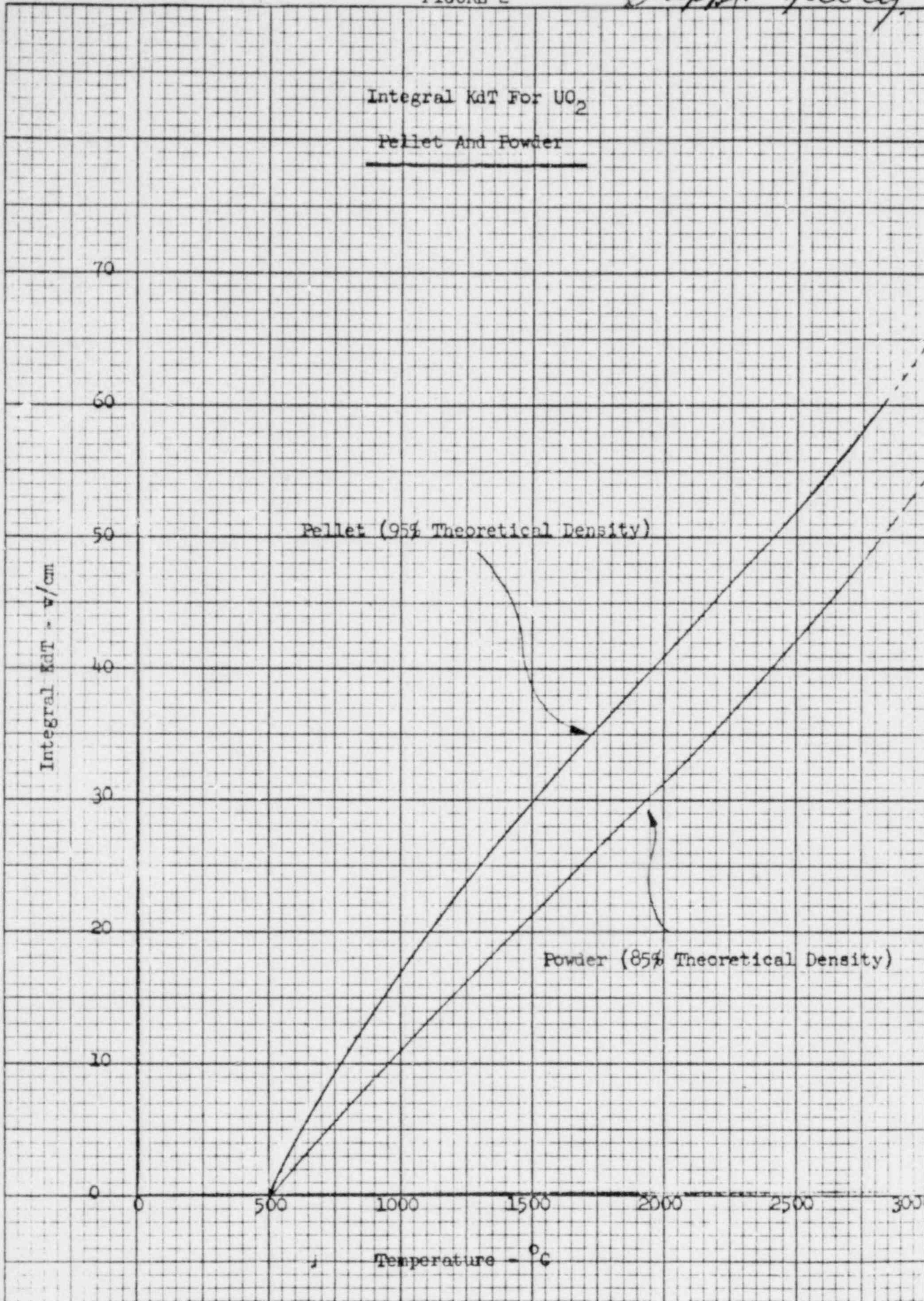

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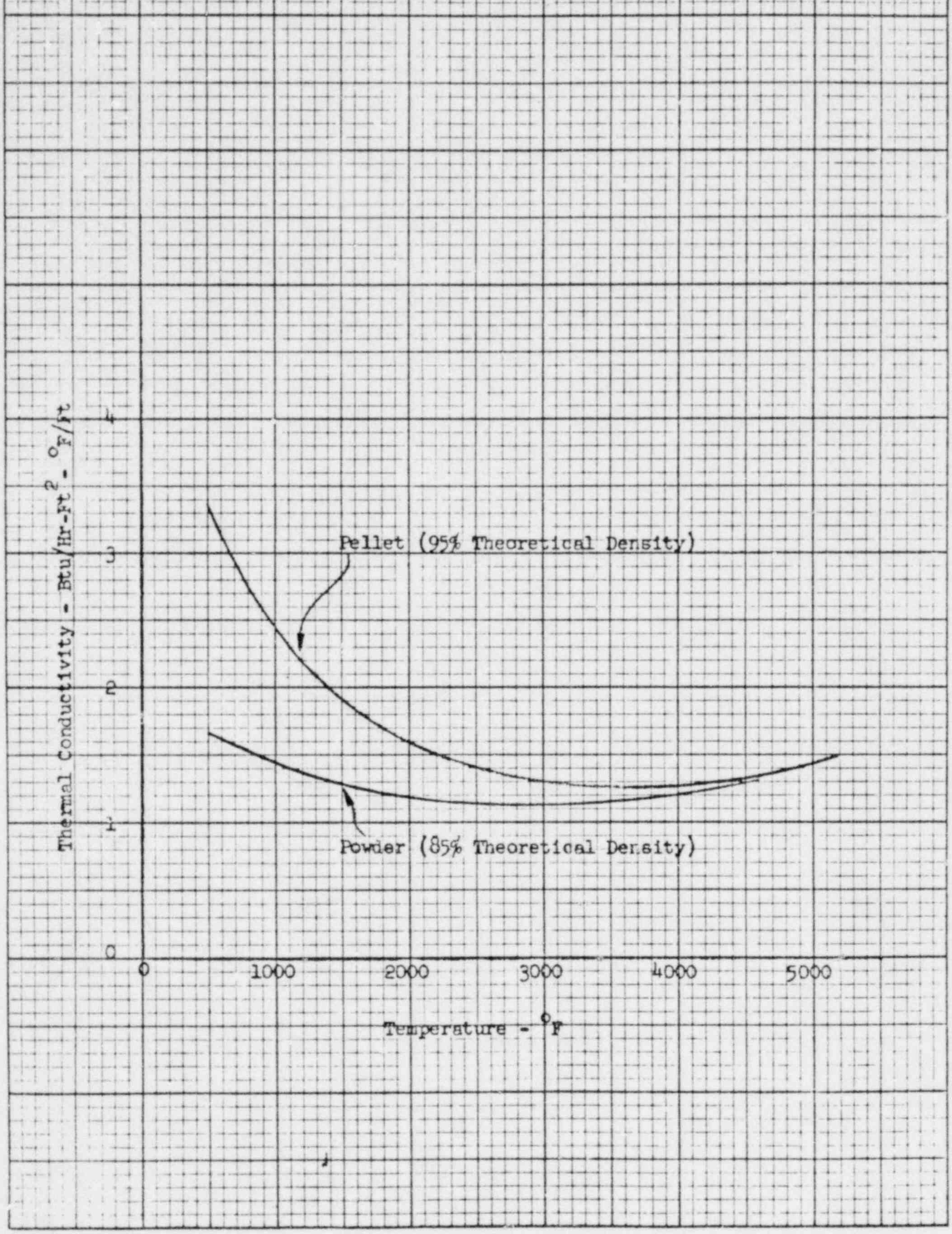
FIGURE 3

Thermal Conductivity For $UO_2$

Pellets And Powder



U. S. Government Printing Office: 1965—769-445

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|---|---|
| FROM: Consumers Power Company<br>Jackson, Michigan<br>Robert L. Haueter | DATE OF DOCUMENT: 7-29-66 / DATE RECEIVED: 8-2-66 / NO.: 2241 |
| | LTR.: X / MEMO: / REPORT: / OTHER: |
| TO: Dr. Richard L. Doan | ORIG.: 1 / CC: / OTHER: (3) executed and (19) conformed c |
| | ACTION NECESSARY ☒ / CONCURRENCE ☐ / DATE ANSWERED: |
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| CLASSIFICATION: U / POST OFFICE REG. NO.: | FILE CODE: Docket No. 50-155 |
| DESCRIPTION: (Must Be Unclassified)<br>Ltr. trans. | |
| ENCLOSURES:<br>Request for Change (No. 10) to the Technical Specifications Lic. No. DPR-6<br>Notarized July 29, 1966.<br>(22 cys. received) | |
| REMARKS: Distribution:<br>1-formal 1-suppl 1-AEC PDR<br>2-Compliance 1-OGC 1-Stoner | |

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| Boyd w/all extra cys...FOR ACTION | 8-2 | | |
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U. S. ATOMIC ENERGY COMMISSION MAIL CONTROL FORM

Form AEC-326S (8-60)