DEC 21 1972

Consumers Power Company

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

December 18, 1972

DOCKETED USAEC DEC 21 1972 REGULATORY MAIL SECTION DOCKET CLERK

Regulatory File Cy.

Mr. John F. O'Leary, Director
Directorate of Licensing
US Atomic Energy Commission
Washington, DC 20545

Re: Docket No 50-155
License No DPR-6

Dear Mr. O'Leary:

On May 18, 1972, we informed the Directorate of Licensing of the results of the reanalysis of the Design Basis Loss-of-Coolant Accident for the Big Rock Point Plant. Results were included for the present reload fuel, Type "F"; and four (4) fuel bundles that utilized an unpowered rod concept to provide a heat sink near the center of the fuel assembly to improve the radiation heat transfer characteristics. We also informed the Directorate of Licensing of our intention to utilize the unpowered rod concept in future reload fuels at Big Rock Point.

The Type "F" fuel has been in use at Big Rock Point since early 1971. Sixty-three (63) of the eighty-four (84) assemblies had been delivered to the plant prior to our May 18, 1972 letter. At that time, hardware and uranium were both on order for the remaining twenty-one (21) assemblies. These assemblies are scheduled for delivery to Big Rock Point this month. Based on the improvements in ECCS performance projected for future reload fuels utilizing an unpowered rod at the center of the fuel assemblies, Consumers Power Company requested General Electric Company to proceed through the component fabrication period of the fabrication campaign with two parallel efforts. The first was the fabrication of a fueled spacer capture rod (located at the center of the fuel assembly) similar to those of the three previously fabricated Type "F" fuel batches. The second was the design and fabrication of a perforated hollow spacer capture rod, similar to that used in the three EEI-Pu assemblies, to provide an additional heat sink at the center of the fuel assemblies. While these two parallel paths were in progress, a physics, thermal-hydraulic and loss-of-coolant accident design review would be conducted to insure that the use of an unpowered rod did not violate the design considerations of the Type "F" fuel.

The design review has been completed and the last twenty-one Type "F" fuel assemblies (designated "Modified F") have been fabricated using the unpowered spacer capture rod. Other than the center spacer capture rod, the "Modified F" fuel is identical to the previously licensed Type "F" fuel (Technical Specifications Change No 21 dated February 9, 1971) in hardware, enrichment and performance.

8101100631

6975

At the time this project was undertaken, it was anticipated that ECCS performance would be improved by at least 200°F. However, two considerations not anticipated in the design stage precluded any large gains in ECCS performance at the licensed output condition. First, the fuel enrichments could not be altered and the additional moderation resulted in essentially the same power output for the three-by-three rod array at the center of the "Modified F" assemblies as is experienced in the original Type "F" fuel. Second, the rod is hollow and heats up very quickly after the postulated loss-of-coolant accident. These considerations resulted in the unpowered rod wetting being delayed to 190 seconds and achieving a maximum temperature of about 2700°F prior to wetting. The maximum clad temperature achieved for the Design Basis Accident is 2730°F for the "Modified F" compared to 2740°F (refer to our May 18, 1972 and September 22, 1972 submittals) for the original Type "F" fuel. As a function of break size, the "Modified F" does not exceed 2300°F for break sizes up to 0.12 ft^2, which is an improvement over the original Type "F" fuel (0.04 ft^2, refer to our September 22, 1972 submittal to the Director of Licensing). Even though these numbers do not represent spectacular gains, we have concluded that the "Modified F" fuel is significantly improved over the original Type "F" fuel from an ECCS viewpoint. A reduction of decay power of about 10% from the use of the ANS + 20% decay power specified for use by the Interim Acceptance Criteria results in the maximum clad temperature being less than 2300°F. For the original Type "F" fuel, the decay power would have to be reduced by about 20% to achieve similar results. We are convinced that the use of the full Baker rate constant for metal-water reaction and the use of ANS + 20% decay power represent significant conservatism in the AEC assumptions in the Interim Acceptance Criteria.

The central spacer capture rod is perforated to permit water ingress. The design of this rod is similar to that used in EEI-Pu fuel assemblies, which have been satisfactorily operated in the Big Rock Point reactor since April 1970. During steady state operation of the reactor, a small portion of the assembly flow ($\sim$ 200 lb/h out of $\sim$ 120,000 lb/h) will be diverted through the spacer capture rod. This will reduce the critical heat flux ratio by less than 0.1% as compared with original Type "F." One additional effect of the water rod is to lower the local power peaking from 1.24 for the original Type "F" fuel to 1.22 for the "Modified F" design. K_∞ values are almost identical to the original Type "F" fuel. The design of the water rod included the consideration that it maintains its geometry throughout the assumed worst case (loss-of-coolant accident) pressure transient.

For the "Modified F" fuel, the fuel pellet length has been reduced and the pellet ends chamfered. The length-to-diameter ratio is reduced from approximately 1.8 for the original Type "F" fuel to 1.0 for "Modified F." The reduction in length-to-diameter ratio and chamfering are expected to reduce local cladding strains associated with pellet distortion. In addition, the dishing of the pellets in the original Type "F" fuel has been eliminated. Data from GETR capsule irradiations, including density measurements on irradiated fuel pellets and average dimensional measurements, show no significant fuel irradiation swelling in the range of peak pellet exposures to be experienced in BWRs. All of the General Electric fuel rod data on UO_2 swelling indicate that the internal porosity in the sintered UO_2 pellet will be sufficient without the dish, to accommodate fuel internal irradiation swelling for the range of exposures to be encountered in the BWR.

Based on the analysis performed we have concluded:

1. Mechanical design of the "Modified F" fuel is essentially the same as that of the previous Type "F" fuel. Thermal and mechanical design evaluations have shown that the "Modified F" design satisfies design limits on cladding stress, strain and temperature under the maximum expected performance conditions.

2. The "Modified F" fuel assembly has nuclear characteristics similar to those of previous Type "F" fuel. The K_∞ data and reactivity coefficients indicate that this assembly can be safely controlled and operated.

3. The thermal-hydraulic characteristics of the "Modified F" fuel are essentially the same as those of the Type "F" fuel. The only differences are due to the water rod which causes a reduced local power peaking in the assembly. The effect of active coolant loss to the water rod is that of reducing the critical heat flux ratio by less than 0.1%. The total effect of the water rod will improve the critical heat flux ratio because of the reduced local power peaking.

Further, we have concluded that the operating limits contained in the Technical Specifications for Type "F" fuel are applicable to and appropriate for the "Modified F" fuel. In our opinion, the use of a central water rod or the minor changes in pellet design in the Type "F" fuel do not constitute a sufficient change in this fuel to require a change in the Technical Specifications.

Yours very truly,

Ralph B. Sewell

RBS/map

Ralph B. Sewell
Nuclear Licensing Administrator

CC: BHGrier,
USAEC