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SAFETY EVALUATION BY THE RESEARCH AND POWER REACTOR SAFETY BRANCH

DIVISION OF REACTOR LICENSING

CONSUMERS POWER COMPANY

PROPOSED CHANGE NO. 10

DOCKET NO. 50-155

Introduction

The Consumers Power Company of Michigan has proposed changes, by letter dated July 29, 1966, and supplemented by TWX on August 16, September 9, and September 26, 1966, 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. These changes would permit Reload "C" fuel bundles to be substituted for the originally installed stainless steel clad fuel. Two special pilot bundles with thinner fuel cladding are to be included as reload bundles. The request for these changes has been designated Proposed Change No. 10.

Discussion

The Reload "C" fuel as described in the proposal by Consumers is similar 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 powder fuel rods are designed to the same cladding stress criteria as those of the pellet fuel. The proposed reload fuel is vibratory compacted to about 85% theoretical density, enriched up to 5.2 weight percent U-235, and has the same clad thickness and outside diameter as the previously installed zirconium clad fuel.

Reduced UO_2 fuel density in the powdered fuel is required to allow for fuel growth and fission gas release over the 15,000 MWD/TU average fuel bundle exposure. Based on the examination of a limited number of in-core irradiation tests and also the continued trouble-free irradiation in the Big Rock Point core of three zircaloy clad 85% theoretical density powder fuel bundles with depletions of 6200 to 7680 MWD/TU and peak heat fluxes of 340,000 to 384,000 BTU/hr. ft^2, fuel integrity is expected to be at least as good as the original stainless steel clad fuel. Because the UO_2 fuel is vibratory compacted to a lower theoretical density than the pellet fuel, the thermal conductivity of the "C" fuel is reduced. Consumers presented curves to show that for the same UO_2 surface temperature, the peak heat generation rate would have to be limited to 19 Kw/ft to avoid center fuel melting, compared to 23 Kw/ft for pelleted 95% dense fuel. However, the improvement in the fuel boundary-clad gap conductance for the powdered fuel (3000 BTU/hr ft^2°F in contrast to 1000 BTU/hr ft^2°F) just compensated for its poorer thermal conductivity and center melting for either fuel would be expected at about 19 Kw/ft. On this basis, the peak heat generation rate for powdered and pelleted fuel will be limited to less than 19 Kw/ft to conservatively prevent center fuel melting.

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The Reload "C" increase in U-235 enrichment over that of Reload "B" results in a total fuel bundle U-235 enrichment approximately equal to the original stainless steel clad fuel bundles. Zircaloy fuel cladding instead of stainless steel fuel cladding also results in a net gain in core excess reactivity. This gain will be cancelled by the more massive cobalt targets in each of the four fuel bundle corners and selective re-use of the stainless steel fuel bundle shrouds used to control the original core reactivity during early core life. This method of reactivity control has previously been successfully employed at Big Rock.

Consumers presented a comparison of nuclear characteristics for "B" and "C" type fuel bundles which showed that moderator temperature and void coefficients are not significantly affected. The Doppler Coefficient is less negative and will be approximately equal in magnitude to that of the original stainless steel clad pelleted UO_2 fuel core loading. Control rod worth for "C" loadings is slightly less than that for the "B" fuel bundles.

Consumers' favorable experience to date with powder fuel lends support to partial refueling of the Big Rock Point core with "C" type fuel. Specifically, operation with intentionally defected fuel revealed no significant differences in performance characteristics of the two fuel types with respect to fuel washout, UO_2 coolant reaction, or susceptibility to waterlogging. In addition, extensive in-pile tests as listed by Consumers have demonstrated the dimensional and chemical stability of the powdered UO_2 fuel. It was noted that diametral strain of the clad of powdered compacted fuel is noticeably less than the strain at midpellet and pellet interfaces of pellet fuel. There have been no confirmed cladding failures where impurities, notably fluorides, in either pellet or powder fuel have been held within allowable limits. During irradiation, powder fuel behaves essentially like pellet fuel when operated at temperatures above 1600-1800°C. At temperatures lower than 1600°C, the compressed particles are bound together by "fission sintering." Grain growth and void migration occurring above 1600°C result in a central fuel void surrounded by a densified annular region while the outer rim of fuel remains at temperatures below which structural changes occur.

Included with the "C" fuel bundles are two pilot bundles which have the same physical dimensions as the "C" bundles except that cladding thickness has been reduced to .025 inch from .034 inch, and fuel diameter has been correspondingly increased resulting in a slight increase in fuel weight and a slight decrease in water-to-fuel volume ratio compared to other "C" fuel bundles. Only the large diameter rods in the two pilot fuel bundles have been changed; i.e., a cobalt rod will be installed at each of the four corners of the fuel bundle. The 0.025 inch thick cladding of the two-pilot fuel bundles is designed to be self-supporting. The stress intensity limits, however, have been increased to a larger fraction of the ultimate strength based in part on recent examinations which show appreciable ductility for Zr-2 cladding irradiated up to fast flux exposures of 1.5×10^{21} nvt greater than 1 Mev. Therefore, if the clad strength is exceeded by external compressive forces, the clad will collapse onto the fuel rather than rupture. (Irradiation effects on ductility appear to saturate at exposures in the vicinity of 0.5×10^{21} nvt.)

Safety Evaluation

Fuel element replacement during the second partial core reloading of Big Rock Point is dependent on the results of fuel examination after the reactor vessel head is removed. Those irradiated fuel bundles which are to be replaced by Reload "C" bundles, will be identified by visual inspection and other examinations. When the bundles to be replaced have been identified, it will be necessary to then calculate bundle power and core reactivity. Thus, by selective placement of fuel bundles and stainless steel shrouds, power and reactivity limits will be maintained. The safety evaluation in support of Technical Specifications Change No. 8 which related to the thermal-hydraulic performance of Reload "B" fuel is valid for Reload "C" fuel; i.e., neither the licensed heat flux limit, 17.2 Kw/ft, nor the minimun DNBR limit, 1.5, will be reached during normal operation or anticipated transients.

We have considered that a reactivity transient of sufficient magnitude to burst the fuel cladding and release the powdered fuel into the coolant might be an incipient source of primary system failure. The finely divided nature of the powdered fuel compared to pellets would result in better energy transfer to the coolant and potentially more severe consequences. Thus, General Electric Company has re-evaluated the consequences of rod-drop and rod-ejection accidents using time constants derived from TREAT experiments (ANL-7204) with vibratory compacted powdered zircaloy clad fuel elements. The transient energy release and potential consequences are tabulated in Table 1.

According to Consumers Power Company and the General Electric Company calculations, primary system rupture would occur with 0.5 feet of vessel movement or 13% vessel strain. Based on these calculations, the worst credible rod-drop accident (4.5% Δ k/k) would result in a transient reactivity insertion below the vessel damage threshold. This high rod worth could only be achieved through multiple operator errors. Normal configurations have rod worths less than about 1.5%.

Table 1 is a summary of three studies by the applicant: (1) calculations were performed relating control rod worth to peak excursion energies, (2) calculations were performed to relate peak excursion energy to physical damage to the reactor vessel, and (3) peak excursion energies were calculated for instantaneous ejection of rods with worths near the expected maximums in the core.

The vessel vertical movement of 0.5 ft. for the 4.5% transient was calculated assuming that there was no energy absorption by the vessel internals and that only the weight of the vessel and internals resisted movement. Vessel strain calculations also assumed no energy absorption by intervening internals.

The peak transient energy in each excursion was calculated by General Electric. The model combines a point-kinetics code and a few-group diffusion code. A spatial Doppler weighting is fed back after each temperature step.

With respect to the 0.025 inch thick zircaloy clad pilot fuel, in contrast to the standard 0.034 inch zircaloy clad, the reduction in the prompt rupture threshold of the clad is negligible. In either case, failure of the clad would occur due to rapid vaporization of fuel before clad temperature has increased noticeably. The vapor pressure would increase very rapidly when fuel temperature rises above the melt-point.

Table 1

Gravity Drop Rod Worth Δ k/k Percent	Peak Calculated Enthalpy (Cal/gm)	Peak Enthalpy Used in Damage Calculations (Cal/gm)	Pounds Above 425 cals/gm (Vaporization, Prompt Rupture)	Pounds Above 280 cals/gm (Fully Molten)	Pounds Above 220 cals/gm (Start Melting)	Maximum Vessel Vertical Movement Feet	Maximum Vessel Strain Percent
1.5	221.						
	*290						
2.0	305.						
	*506						
2.5	366.						
3.0	420	450	3	460	1080		
3.5	475	490	30	660	1450	0	0
4.0	530	540	100	750	1810	0.17	0
4.5	585	590	210	1130	2480	0.50	1.1

* Energy associated with instantaneous rod ejection.

In summary, the reduced UO_2 density of the vibratory compacted powdered fuel and the increased U-235 enrichment in contrast to Reload "B" fuel bundles do not present a significant change to the reactor safety considerations described or implied in the Final Hazards Summary Report. Further a conservative analysis of pressure transients which could conceivably result from the release of molten-vaporized fuel to the surrounding coolant due to a reactivity excursion shows that the conversion of heat to mechanical energy is not great enough to damage the integrity of the primary system and therefore a major reactivity transient would not be expected to cause a loss-of-coolant accident. Any fission products released in the postulated transients would be confined within the primary system since no break would occur.

Conclusion

Based on the foregoing, we have concluded that Proposed Change No. 10 does not present significant hazards considerations not described or implicit in the hazards summary report and there is reasonable assurance that the health and safety of the public will not be endangered.

We therefore believe the Technical Specifications of License No. DPR-6 may be revised as indicated in Attachment A.

Original signed by:
Roger S. Boyd

Roger S. Boyd, Chief
Research and Power Reactor Safety Branch
Division of Reactor Licensing

Date: OCT 7 1966