SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

DOCKET NO. 50-155

CONSUMERS POWER COMPANY

PROPOSED CHANGE NO. 14

INTRODUCTION

By letter dated February 6, 1968, Consumers Power Company of Michigan proposed Change No. 14 to the Technical Specifications of License No. DPR-6 to permit insertion of vibratory compacted fuel rods, designated Reload "E" fuel, into the Big Rock core. On March 5, 1968, Consumers informed us that the "E" fuel loading was being changed from vibratory compacted fuel to pellet type fuel. By an addendum to Proposed Change No. 14, transmitted by letter dated May 10, 1968, Consumers requested that Proposed Change No. 14 be amended to reflect the design change from powder to pellet fuel. Supplemental information relating to the change request was submitted by a telegram received on June 17, 1968.

DISCUSSION

Proposed Change No. 14 as modified would permit Consumers Power Company to refuel the Big Rock Point reactor with fuel bundles, designated Reload "E" fuel, which incorporate the General Electric Company's current commercial reactor fuel design features. The licensee anticipates that all of the fuel designs now in use in the Big Rock Point reactor will eventually be replaced with the Reload "E" design.

The Reload "E" fuel design is similar to the original Big Rock Point "A" fuel design in that machined castings are used for the upper and lower tie plates. The fuel bundle is held together by eight tie rods which are fuel rods with special threaded end plugs which mate with the upper and lower tie plates. Fuel rod clearances are maintained by 3 spring clip spacers positioned along the length of the fuel bundle and held in place by a central capture rod. Six rods, one in each corner and two adjacent to diagonally opposite corner rods, are removable to accommodate cobalt target material. The outside diameter of the fuel rods in the 9 x 9 Reload "E" fuel rod array is 0.5625 inch which is about the same as the diameter of the intermediate centermelt fuel rods presently being irradiated in the Big Rock reactor. However, the zircaloy-2 clad thickness of the Reload "E" fuel rods is nearly 15% greater. Since there are 81 rods per bundle, in contrast to 64 rods in each intermediate centermelt fuel bundle, the clearance between rods is reduced.

A comparison of the significant data for Reload "B", "E" and Intermediate centermelt fuels is provided in Table I.

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TABLE I

		Reload "B"[5]	Type "E" Fuel Pellet		Intermediate Centermelt Pellet Fuel
Geometry, Fuel Rod Array		11 x 11[1]	9 x 9[1]		8 x 8
Fuel Rod - O.D. of clad	inches	0.449	0.5625		0.570
Rod pitch	inches	0.577	0.707		0.807
Fuel pellet diameter	inches	0.373	0.471		0.488
Pellet-clad diametral gap	inches	0.008	0.0115		0.012
Clad Thickness	inches	0.031	0.040		0.035
Active length of fuel	inches	70.0	69.75		66 - 67.3
D_H channels inside fuel bundles	inches	0.497	0.57		0.38
			High Density[3]	Low Density[3]	
$\int_{0^\circ C}^{2800^\circ C} K dt$ [2]	w/cm	93	93	85.5	93
Fuel rod power to reach melting - operating conditions	kw/ft	>18.3	21.8	20.6	21.4[4]
Heat flux for incipient melting	BTU/hr-ft^2	530,000[6]	504,000	477,000	490,000
Average bundle power @ 235 Mwt	Mwt/bundle	2.80	2.80		3.25
Average fuel rod power generation at rated power	kw/ft	4.1	6.26		16.4 (hot rods
Average heat flux at rated power	BTU/hr-ft^2	124,000	144,000		375,000 (hot rods)
Peak heat flux at rated power	BTU/hr-ft^2	321,000[7]	382,000[7]		490,000
Peak heat flux at 122% overpower	BTU/hr-ft^2	392,000[7]	465,000[7]		600,000
Peak fuel rod power at 122% overpower	kw/ft	13	20.2		26.2
Average U-235 enrichment	%	2.98	2.98		2.86
Wt. UO_2/bundle	kg	149 pellet[8] 139 powder	155		136 Total 76.5 (hot rods)
Wt. enriched UO_2/bundle	kg	4.45	4.62		3.76 (hot rods)
Wt. zircaloy clad/bundle	kg	41	40.5		28.6
Heat transfer area/bundle	ft^2	79	66		30.2 (hot rods)
Pellet density	% theo	94 ± 1	2.35%E/90-92 2.93%E/95 3.55%E/95 with 3½% dished pellet volume		94

Symbols

D_H - hydraulic diameter
$\int$ Kdt - thermal conductivity integral

Notes

(1) 4 corner rods may be cobalt targets
(2) Incipient melting
(3) High density; 95% theoretical; low density, 90-92% theoretical
(4) Start of life - Proposed Change No. 13, 5/26/57 (p. 7)
(5) Proposed Change No. 8, 1/24/66
(6) Proposed Change No. 8 - Additional information, 3/17/66
(7) End of Life
(8) Proposed Change No. 8 (p. 5), Proposed Change No. 10 (July 1966) Table I

The licensee has reported that the minimum critical heat flux ratio (MCHFR) with Reload "E" fuel will be above 1.5 for steady state operating conditions at 122% of rated power level and also during the loss of coolant flow transient resulting from pump motor power failure. The MCHFR of 1.5 is above the minimum values permitted by the Big Rock Point Technical Specifications and the calculational method has been previously evaluated and accepted by us; therefore, we conclude that the probability of fuel damage caused by excessive Reload "E" fuel clad temperature is acceptably small.

In contrast to the "B" and "C" reload fuel bundles which have been used for refueling the Big Rock Point reactor over the past two years, the "E" reload fuel bundles have fewer rods. ("E" bundles contain a total of 81 rods made up of 77 fuel rods and 4 cobalt targets; "B" and "C" bundles formerly used for Big Rock reactor refueling contain 121 rods made up of 117 fuel rods and 4 cobalt targets.) Since the average fuel bundle power at rated reactor power, 2.80 Mwt, remains unchanged, the power produced by each fuel rod within the "E" bundle must increase. The net effect is an increase in the average linear fuel rod power generation from 4.1 kw/ft to 6.3 kw/ft with corresponding peak values increased from 10.6 kw/ft to 16.5 kw/ft at rated power conditions. Similarly, the average heat flux increases from 124,000 to 144,000 BTU/hr-ft^2 and the peak flux at rated power increases from 321,000 to 382,000 BTU/hr-ft^2. The calculated heat flux to cause incipient centermelting is 504,000 BTU/hr-ft^2 for the high density fuel pellets and 477,000 BTU/hr-ft^2 for the low density fuel pellets. As shown in Table I, the maximum heat flux at 122% of rated power is 465,000 BTU/hr-ft^2. On the basis of these calculations, we have concluded that fuel melting will not occur during normal operation and the 22% power margin between normal and centermelting conditions is adequate assurance against unintentional melting.

The licensee has also requested that the technical specification limit for heat flux be reduced from 530,000 BTU/hr-ft^2 to 500,000 BTU/hr-ft^2 for the Reload "E" fuel bundles. Operation at this limit, which is significantly in excess of the maximum heat flux of 465,000 BTU/hr-ft^2 expected at 122% of rated power, would result in a molten volume fraction of less than 0.005 in the 90-92% dense fuel. Based on a 10% phase-change volume expansion of the molten fraction, the increase in volume fraction occupied by molten UO_2 could be 0.0005. Since molten volume fractions significantly higher than 0.005 have been accommodated by fuels of similar density without excessive cladding strain and subsequent cladding damage, we conclude that fuel rod integrity will not be affected should this heat flux limit be reached.

During normal operation, Reload "E" center fuel temperatures will be higher than type "B" fuel. Consequently, in the hottest regions of the fuel rods slightly more fission product gases will be released from the fuel matrix in the "E" fuel. However, the net increase in the release of fission product gases in comparison with type "B" fuel is negligibly small and of minor importance in safety considerations.

The design basis accident, which assumed that all of the fission product gases are released from the fuel matrix without time dependence, is not altered. We have concluded, therefore, that operation of the "E" reload fuel with calculated center temperatures greater than the previous reload fuel, a result of operating with fewer rods per bundle, does not significantly affect the fuel rod integrity or accident consequences.

The fuel bundle design uses three different fuel rod enrichments selectively located to achieve the desired heat generation characteristics. We understand that the rods are fabricated in batches and clearly identified to prevent mixup. Further, their assembly into clusters is in accordance with written procedures. Visual inspection after the rods are assembled into bundles provides added assurance that rods have been properly placed within the bundles. We therefore conclude that the probability of loading improperly assembled fuel bundles is acceptably low.

Control rod reactivity worths for "B", "C" and "E" fuel types or combinations thereof are similar for given core loadings and on this basis most of the studies related to Big Rock Point reactivity transients performed in conjunction with the centermelt fuel irradiation program are applicable. For example: rod drop accidents analyzed for the centermelt fuel bundles are unaffected by the change from "C" fuel to "E" fuel. It is also shown that "E" fuel without centermelt fuel bundles in the core results in lower peak fuel enthalpies and a smaller amount of fuel with enthalpy above 220 cal/gram than type "C" fuel bundles in the Big Rock core. This behavior is predominantly a result of the reduced local peaking factors for "E" fuel due to improved distribution of enriched U-235 since the average U-235 enrichment and mass of UO_2 fuel and zircaloy clad are approximately the same for the "B", "C" and "E" type fuel bundles and therefore should otherwise show similar heat generation and energy storage characteristics. Accordingly, we have concluded that the use of Reload "E" fuel, considering reactivity transients and resultant fuel enthalpy peaks associated with rod drop accidents, will not cause damage to the primary system or changes to the core geometry which could interfere with core cooling following the incident.

The licensee has reevaluated the loss of coolant accident over the complete spectrum of primary coolant system break sizes. The results show that for the small breaks, insertion of "E" reload fuel will not significantly change the characteristic mode of fuel failure. For small bottom breaks, the primary system depressurization is too slow after water level drops below core midplane to prevent excessive fuel rod clad temperatures before emergency core spray can be initiated. A report, presently being prepared by Consumers, will address this problem. For small breaks at the top of the reactor vessel (less than 0.5 ft^2 break), adequate core spray is achieved before the water level drops below the core midplane and therefore clad temperatures remain low and fuel clad perforations are near zero. We agree that differences in the pattern of fuel failure under these circumstances between "E" and "B" or "C" type fuels are negligible because the residual or stored heat will be dissipated in the water in all cases before water level reaches core midplane. Further, the mass of fuel and clad available as a heat sink to accumulate the decay heat energy (which is not radiated after water level drops below core midplane and initiation of fuel heat up is assumed) is approximately the same.

For larger primary system breaks, when most of the residual heat is not released to the primary coolant during the blow down period prior to core spray initiation, the "E" fuel clad temperatures are significantly higher than for "C" reload fuel. Additional information supplied by the licensee reveals that the high calculated Reload "E" fuel clad temperatures for the large breaks are due to the low UO_2 thermal conductivity and low wetted channel heat transfer coefficients assumed in the calculations. If these values are adjusted to most accurately reflect the UO_2 temperature prior to and during the accident transient and the heat transfer from the fuel after the fuel bundle channel boxes are wetted by the core spray, the peak clad temperature following large primary system breaks becomes less than 2200°F instead of 2800°F as reported in the original proposal. We agree that for the temperature range of interest and considering pellet densities as low as 91%, the 1.6 thermal conductivity value used in the revised calculations is justified. We also agree that there is an acceptable basis for using channel box heat transfer coefficients of 500 BTU/hr-ft^2 after the channel is wetted, instead of 5. The higher value has been derived from core spray effectiveness tests using electrically heated 36 and 49 rod bundle arrays and is consistent with values currently accepted for larger BWR plants. We have concluded that the reduction in calculated peak clad temperature from 2800°F to 2200°F is justified for the reasons stated and that 2200°F is an acceptable clad temperature limit because it is within the range of core spray test experience. We have also considered that, in the event these less conservative coefficients were not applicable, not more than 4 of the 84 fuel bundles would contain rods reaching 2800°F following a large primary system break. No more than 196 fuel rods (less than 3% of the core fuel rods) would reach temperatures in excess of 2200°F. Consumers Power Company has reported that removal of undue conservatism reduces to 16 the number of fuel rods with calculated peak clad temperatures as high as 2200°F.

The calculated percentage of fuel rod perforations following primary system breaks of less than 0.9 ft^2 is lower than for previously used Reload "C" fuel and not significantly greater for the larger breaks. Considering the random nature of clad perforations and the conservative assumption that fuel rod gas pressure is sufficient in all rods to cause clad perforations at 1500°F, there is reasonable assurance that the core geometry with "E" fuel will not be disturbed following a loss of coolant accident to the extent that adequate core spray cooling would be prevented for those break sizes where the core spray is considered effective.

CONCLUSION

We have concluded that inserting Reload "E" fuel into the Big Rock Point core does not increase the probability of an accident which could release fission products from the primary system nor impair the effectiveness of the installed engineered safety features beyond conditions previously analyzed. On this basis, we conclude that Proposed Change No. 14 does not represent significant hazards considerations not described or implicit in the safety analysis report and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, we believe that the Technical Specifications of License No. DPR-6 may be revised as indicated in Attachment A.

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