SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

DOCKET NO. 50-155

CONSUMERS POWER COMPANY

PROPOSED AMENDMENT NO. 1

INTRODUCTION

By letter dated May 26, 1967, the Consumers Power Company of Michigan has proposed Change No. 13 to the Technical Specifications which we have redesignated Amendment No. 1 of License DPR-6 for the Big Rock Point Power Plant. Supplemental information was submitted on August 15, 1967, November 10, 1967, and December 14, 1967.

Amendment No. 1 would permit insertion of six high performance developmental fuel bundles into the Big Rock Point core as part of the normal core complement of 84 fuel bundles. The developmental fuel will be irradiated until the most depleted fuel rods acquire 21,000 MWD/T U average exposure. This irradiation program, designed to investigate the performance characteristics of fuel rods with center melting, is an extension of the high performance UO_2 program sponsored jointly by the U. S. Atomic Energy Commission and Euratom. The proposed change was reviewed by the Advisory Committee on Reactor Safeguards (ACRS) which concluded in its report dated December 20, 1967, that "the reactor can be operated with the high performance test assemblies without undue risk to the health and safety of the public". A copy of the ACRS report is attached.

DESCRIPTION

The Table, which follows on pages 3 and 4 sets forth the important characteristics of the high performance fuel bundles and presents a comparison thereof with the Type "C" fuel bundles most recently used to refuel the Big Rock Point reactor. The salient aspects of that table are further discussed below.

The six high performance developmental fuel bundles, containing 324 fuel rods, will operate with peak heat fluxes of 450,000 $\pm$50,000 BTU/hr ft^2. To achieve the design objectives of performance and power output, it will be necessary that the high performance fuel bundles be repositioned into higher neutron fluxes during their approximately 2 - 2 1/2 year in-core irradiation. The use of three

8111190310

different U_{235} enrichments, i.e., 4.3, 5.0, and 5.6%, to control individual fuel rod power generation within a fuel bundle permits most of the fuel bundle power to be generated in about 60% of the fuel rods. The remaining fuel rods contain depleted fuel, i.e., only 0.22% wt percent U_{235}, and therefore generate only a small fraction, about 3%, of the fuel bundle power. Two of the fuel bundles, designated as intermediate performance fuel, contain 0.570 inch diameter fuel rods and are designed to produce incipient center melting. One of the intermediate performance bundles contains pellet type fuel and the other vibratory compacted powder fuel. The other four bundles, designated as advanced performance fuel, contain 0.70 inch diameter rods and are designed to produce substantial center melting. Two of the advanced performance assemblies will contain UO_2 in the form of sintered cored pellets and the other two will contain UO_2 in the form of powder. Selected rods will contain tungsten wafers to minimize axial movement of the molten UO_2 during operation. All six high performance fuel bundles will be distributed within the core so that the minimum center-to-center distance of these fuel bundles is 42 cm (16.5 in.). None of these bundles will be positioned in the outer row of the core. The handles of the high performance fuel bundles will be notched to permit visual verification of their core positions. Correct placement of the rods within the fuel bundle geometry is assured by 3/8 inch identification letters stamped on the fuel rods.

The developmental fuel bundles are designed to permit disassembly when removed from the core during refueling. This feature will facilitate examination and replacement of all individual rods. Selected irradiated rods will be shipped to the Vallecitos Nuclear Center for destructive examination after a suitable period of radioactive decay in the spent fuel storage pit at Big Rock Point. Rods selected for destructive examination will be replaced with new rods.

Lines 21 and 22 of the table show that the average power generation in the "hot" rods (high power producing rods) of both intermediate and advanced performance fuel is significantly larger than the peak power density in the Type "C" Big Rock Point fuel. Lines 24, 25, and 26 show that the total UO_2 in each advanced performance fuel bundle is significantly greater than in either Type C or the intermediate performance fuel, but the total amount of U_{235} in either an advanced or intermediate performance fuel bundle is less than for a Type C fuel bundle. Further, the amount of UO_2 fuel in the power generating rods of each proposed fuel bundle is noticeably less than the amount in the power producing rods of a Type C fuel bundle. The cross-sectional area of the developmental fuel rods, however, is increased in accordance with lines 2 and 6 of Table I by a factor of 1.7 for the 0.570 inch diameter rods and 2.65 for the 0.700 inch diameter rods.

TABLE 1
COMPARISON OF PROPOSED FUEL
TO
BIG ROCK "TYPE C" FUEL

			Big Rock Type "C" Fuel	Proposed Fuel: Intermediate Performance		Proposed Fuel: Advanced Performance	
1	Geometry, Fuel Rod Array		11 x 11(1)	8 x 8		7 x 7	
2	Standard Rod Diameter	inches	0.449	0.570		0.700	
3	Number Standard Rods per bundle		121	36		29	
4	Number Special Rods with depleted uranium		0	28		20	
5	Special Rod Diameter	inches	0.344(2)	0.570		0.700	
6	Standard Rod Tube Wall	inches	0.034	0.035		0.040	
7	Special Rod Tube Wall	inches	0.031	0.035		0.040	
8	Rod Pitch	inches	0.577	0.807		0.921	
9	Active Fuel Length	inches	70	66 - 67.3		65 - 66.3	
10	UO_2 Density, Percent Theoretical Pellet		85 powder	Pellet 94	Powder 85	Pellet 94	Powder 85
11	Fill gas		He	He		He	
12	Spacers per bundle		5	5		5	
13	Clad material		Zr-2	Zr-2		Zr-2	
14	Wt Zircaloy clad per bundle	Pounds	90	62.7		67.6	
15	D_H channels inside fuel bundle	inches	0.497	0.88		0.85	
16	Rod to Rod clearance	inches	0.160	0.237		0.221	
17	Rod to channel clearance	inches	0.128	0.160		0.155	
18	Number of bundles	UO_2 Pellet	N.A.	1		2	
		UO_2 Powder	40	1		2	
19	Steady state heat flux limit	Tech Specs BTU/hr ft^2	500,000 550.000(3)	500,000		500,000	
20	Power generation at limit	KW/ft(6)	19(3)	Pellet 21.8	Powder 21.8	Pellet 26.9	Powder 26.9
21	Peak power at rated conditions	KW/ft	core 12.3	21.8	21.8	25.4	25.4
22	Average power	KW/ft	core 4.1	Hot Rods 16.4	Hot Rods 16.4	Hot Rods 19.4	Hot Rods 19.4
23	Average Heat Flux	BTU/hr ft^2	core 124,000	Hot Rods 375,000	Hot Rods 375,000	Hot Rods 360,000	Hot Rods 360,000
24	Wt UO_2/bundle	kg	133	136	129	159	152
25	UO_2/bundle (U_{235})	kg	4.83	Hot Rods 3.76	Hot Rods 3.7	Hot Rods 4.6	Hot Rods 4.35
26	UO_2 in power producing rods/bundle	kg	133	Hot Rods 76.5	Hot Rods 72.5	Hot Rods 95	Hot Rods 89.5

Table 1 Cont'd

COMPARISON OF PROPOSED FUEL TO BIG ROCK "TYPE C" FUEL

		Big Rock Type "C" Fuel	Proposed Fuel: Intermediate Performance		Proposed Fuel: Advanced Performance	
27	Heat transfer surface bundle ft^2	79	Hot Rods 30.2	Hot Rods 30.2	Hot Rods 29.4	Hot Rods 29.4
28	Average bundle power MWt	2.86	3.25	3.25	3.25	3.25
29	Radial power factor in core	1.30	1.21	1.21	1.14	1.14
30	MCHFR at 122% Power	>1.5	1.53	1.53	1.54	1.54
31	Doppler coef. for bundle at 1000°K delta k/k°F	(4) -1×10^{-5}	-0.86×10^{-5}		-0.95×10^{-5}	
32	Temp coef. 25°C at BOL delta k/k°C	(4) $+3.2 \times 10^{-6}$	$+1.04 \times 10^{-4}$		$+5.6 \times 10^{-5}$	
33	Void coef. 20°C at BOL delta k/k° unit void	-0.28	-.24		-.27	
34	Cold pellet fuel to clad gap inches	Powder	0.012 for pellets		0.013 for pellets	
35	Central hole, pellet fuel inches	(5) N.A.	0		Pellet 0.100	
36	$\int^{2800°C} K\,dt$ Pellet fuel W/cm	N.A. (5)	59 - 62		59 - 62	
	$\int_{500°C} K\,dt$ Powder fuel W/cm	49	49		49	
37	Fuel enrichment wt% U-235	2.9&5.2	4.3;5.0; 5.6;0.22		4.3;5.0; 5.6;0.22;	
38	Fuel Rod lifetime MWD/T U	15,000 ave. rod	Ave. hot rods 21,000		Ave. hot rod 21,000	

Symbols

MCHFR - Minimum critical heat flux ratio.
D_H - Hydraulic diameter of coolant channels.

(1) 4 corner rods may be cobalt targets
(2) 8 special corner rods and 4 Cobalt target rods for radial supports
(3) Incipient melting. Tech spec limit is 500,000
(4) Proposed Change No. 13 dated May 16, 1967
(5) NA - not applicable
(6) Melting starts in powder fuel at 19.4 KW/ft
Melting starts in pellet fuel at 24.2 KW/ft

The applicant has indicated that power in each 7 x 7 fuel bundle is generated in essentially 29 rods, because the remaining 20 fuel rods contain depleted fuel with little U_{235}. The arrangement of the hot (high power producing) rods and "cold" (depleted fuel) rods in an annular geometry provides the maximum cold surfaces adjacent to the power producing rods and optimizes power distribution and coolant utilization within the four 7 x 7 advanced performance fuel bundles. The enrichments and placement of the 36 high power rods and 28 depleted rods within the two 8 x 8 array intermediate performance fuel bundles as with the advanced performance bundles were specified to provide the maximum number of fuel rods operating at or near the desired heat flux, i.e., 450,000 $\pm$ 50,000 BTU/hr ft^2, at the axial peak.

It should be noted that there are only 29 high power rods in the 7 x 7 matrix in contrast to 36 in the 8 x 8 matrix; therefore, the average power generation in each of the 0.700 inch diameter rods in the 7 x 7 array is greater by the ratio of 36/29 than the power generated in the 0.570 inch diameter rods of the 8 x 8 fuel bundles.

The power of the three types of enriched U rods (high power rods) decreases with exposure as the U_{235} inventory decreases. The power in the depleted rods (cold rods) increases with exposure because of plutonium production. To compensate for this reduction of power in the high power rods, the fuel bundles will be moved to positions of higher radial power factors as irradiation proceeds so that the design heat flux of the high power rods can be maintained throughout exposure lifetime. It has been estimated that the relative power generated by the depleted fuel rods increases over the fuel bundle lifetime by a factor of less than 3, and therefore, would not significantly affect the lifetime thermal performance of the high power rods.

Since a linear power density of 19.4 KW/ft results in incipient center melting in the vibratory compacted powder fuel rods and 24.2 KW/ft correspondingly causes centerline melting in pellet fuel rods, it can be seen from lines 21 and 22 that most of the rods will be near centerline melting at the hottest locations when the reactor is at rated power. In the case of the hottest, large diameter, powder fuel rods, there will be molten fuel over a significant rod length, i.e., from approximately 1 to 1.5 feet on each side of the axial power peak, with the maximum molten fuel cross-sectional area approximately 36% of the cross-sectional area of the fuel rod.

EVALUATION

In consideration of the fuel characteristics identified above, our safety evaluation is concerned with:

1. Normal plant operations, including consideration of the significantly higher average heat fluxes from the hot centermelt fuel rods than the peak heat flux for the normal Type "C" Big Rock Point Core, solid to liquid fuel phase changes, and average hot rod fuel exposure of 21,000 MWD/T U, which is significantly greater than present fuel lifetime exposures;

2. Loss of coolant conditions, including consideration of the greater fuel mass, higher temperatures (fuel enthalpy or stored energy), and higher rod power levels in the proposed fuel rods in relation to the ability to dissipate the residual and decay heat;

3. Reactivity excursions associated with the hypothetical control rod drop accident.

NORMAL PLANT OPERATION

Fuel Rod Power: The fuel rods have been sized and arranged within the bundle so as to achieve the maximum cooling. Test results with hot rods adjacent to cold rods similar to the arrangement in the six proposed developmental fuel bundles confirm that the multi-channel model for calculating the Minimum Critical Heat Flux Ratio (MCHFR), currently used by General Electric to determine fuel rod heat-transfer limits in boiling water reactors, is conservative for the proposed rod arrangement. Using this calculational model, it was determined that the MCHFR at 122% of rated power is greater than the minimum value of 1.5 permitted by the existing technical specifications. It has also been determined that the peak clad temperature at rated power will be approximately 900°F for those hot fuel rods which remain in the core for the full irradiation lifetime of 21,000 MWD/T U. The capability to operate at 122% of rated power without exceeding heat transfer limits provides a sufficient margin, in our opinion, to ensure fuel cladding integrity during normal operation.

The licensee has reported that failure of the main coolant circulation pumps or the main power generator with the high power density fuel in the core will not result in violation of the minimum critical heat flux limits. Although a nuclear power surge would occur due to pressurization and void collapse, the heat transfer limits would not be exceeded and fuel temperatures would not be excessive. The MCHFR during the power and flow transient would be greater than the minimum value of 1.5 permitted by the technical specifications. We are in agreement with the calculational methods used in developing their conclusion. Therefore, it is our opinion that such unanticipated transients would not cause a significant release of radioactivity from the fuel rods into the primary coolant system or diminish the integrity of the primary system.

The high linear power density of the developmental fuel rods causes significant increases in the fuel temperature compared to Type "C" Big Rock fuel, and results in possible melting of the UO_2 at the center of the hot fuel rods whenever the power density is above 19.4 KW/ft for powder fuel or 24.2 KW/ft for pellet fuel. These values correspond to $\int Kd\theta$ values of 50.5 W/cm and 63 W/cm. Only the 58 large diameter, vibratory compacted, hot fuel rods will have significant molten fuel. The remaining 130 high power density fuel rods will have incipient fuel melting conditions at the peak axial power locations. The effects of increased fuel temperature, as well as center melting, necessitated new design provisions which have been developed and tested during the AEC-EURATOM sponsored "UO_2 High Performance Program".

Clad-Fuel Interaction: Mechanical interaction between the fuel and the clad resulting from differential expansion was considered as one of the largest potential contributors to high cladding strain. To minimize this potential cause of cladding failure, a large fuel-to-clad diametral gap, i.e., 0.012 inches for 0.570 inch diameter fuel rods and 0.013 inches for the larger rods, has been provided. At rated power, the diametral clearance will not be less than 1 mil over the lifetime of the fuel.

Rod Fabrication: A depleted UO_2 pellet is placed at the end of each active fuel column to minimize the effects of hot fuel at the end plug and plenum spring. Dished and cored fuel pellets have been provided to allow sufficient volume to accommodate the phase change volume expansion of UO_2 on melting, based on 9.6% measured UO_2 volume changes during destructive examination of the irradiated fuel. A 15% void space has been provided in each of the 94 hot fuel rods using vibratory compacted powder by compacting to only 85% of the theoretical UO_2 density. In consideration of the test irradiation experience and the observed resultant sintering densification and accompanying center voids, we believe this void is sufficient to accommodate the volume expansion caused by melting of powder fuel. The applicant has calculated that power levels of 136% and 200% of rated power would be required to cause cladding failure of the small and large diameter pellet fuel rods, respectively. The maximum melt fraction for pellet fuel at 122% of rated power is only 22% compared to the design capability of 71%. The maximum melt fraction of powder fuel at 122% of rated power is 45% compared with a theoretical limit of 100% melting which represents a substantial margin above the MCHFR power limit.

Fuel Migration: While operating with molten fuel conditions, the possibility of fuel migration within the fuel rod exists. Studies performed by General Electric indicate that increased or decreased fuel concentration along the axial direction of the hot rods causes insignificant reactivity effects. Moreover, as mentioned earlier, only the 58 hot, large diameter powder fuel rods have molten center fuel over a sufficient length to permit fuel movement by this means. A proposed modification to the technical specifications, which restricts the rate of initial power increase between 170 MWt and 240 MWt to 1/2 MWt per minute, allows time for fuel migration. This is particularly significant for the powder fuel rods, when the fuel is sintered and densifies the first time melting occurs. This restriction also permits gradual axial expansion of the molten fuel when the peak power has moved downward. We believe that this is a prudent restriction to avoid the remote possibility of fuel rod damage from these factors. Also, tungsten wafers are placed at 18 inch intervals in 4 hot, large diameter, pellet rods and 4 hot, large diameter, powder rods to minimize the axial movement of molten fuel for comparative performance evaluation.

Fuel Rod Lifetime: The capability of the hot fuel rods to achieve 21,000 MWD/T U average accumulated exposure without failure will be confirmed by removing 4 representative fuel rods during each core refueling for destructive examination at the Vallecitos Nuclear Center. These examinations will reveal the amount of fuel burnup, extent of melting, fission product distribution, gas release, fuel movement and dimensional changes. Deviations from predicted conditions will be carefully evaluated to determine the effect on predicted fuel lifetime. The times required for fuel cooling, shipment, and examination result in about a five month delay after rod removal before results of the examinations are obtained. We believe that the confirmation of design performance by the destructive examinations is valuable in ensuring safety of continued irradiation for these bundles. Therefore, the modified technical specifications require that after the first rod removals, the four advanced performance bundles should not be reinserted until the initial destructive examination results are obtained and evaluated.

Fuel Rod Destructive and Non-Destructive Tests: A number of non-destructive tests will be performed on each of the developmental fuel bundles during each core refueling period. Each bundle will be leak tested by the "dry sipping method". The bundles will be examined visually using television and an underwater periscope. The diameter of each of the 188 hot rods will be measured with a go-no-go gauge to detect dimensional changes resulting from irradiation. In addition, several rods will be measured with the profilometer for comparison with the pre-irradiation traces.

An unexpected diameter increase will be cause to discontinue irradiation of the affected rod. These examination procedures, first performed when the hot rods reach less than 15% of the expected lifetime, will provide added assurance that the fuel is performing in accordance with predictions.

Continuous monitoring of the gases from the air ejector monitor will detect fuel element rupture. To provide early warning of a fuel failure, the monitor alarm set points will be lowered to about twice the background level for the duration of these developmental fuel irradiations. The few fuel rod failures experienced during the Euratom high power density fuel development program caused the release of only a small fraction of UO_2 fuel into the coolant system. Based on examination of the Euratom fuel rods and shrouds adjacent to the failed element, it has been concluded that a similar single rod clad failure would not cause adjacent rods in the six high performance fuel bundles to fail. The 188 high power fuel rods employ the same basic mechanical design and fabrication methods as the Type C Big Rock Point reload fuel which has performed with no known failures to date. The developmental fuel is expected to be equally effective in maintaining fuel rod geometry and containing fission products. Nevertheless, if a fuel rod failure should occur, the release of radioactivity to the coolant will be detected in the manner described and any fission products released to the atmosphere via the air ejector and stack will remain within the technical specification limits for continuous operation and will not create a hazard to the health and safety of the public.

LOSS OF COOLANT

As shown in Table I, lines 26 and 29, the two intermediate performance fuel bundles will generate the most power in the least amount of fuel. Decay heat in these small diameter rods, following the design basis accident (double ended rupture of a recirculating pipe), will cause the fastest rise in average enthalpy at the various fuel rod cross sections, if film blanketing and insulated fuel rod conditions are assumed. However, the residual or stored energy in the large diameter fuel rods, which normally operate with peak heat generation rates of 25.4 KW/ft, results in average fuel enthalpies which are initially the highest in the core, but which would rise more slowly than those of the smaller diameter rods following the design basis accident. Redistribution of energy within the rods (after heat transfer from the rod ceases) also influences the rate of clad temperature increase.

For loss-of-coolant caused by primary system breaks which are an order of magnitude smaller than those involved in the design basis accident (see Figure 7-1 of Applicant's supplement dated November 10, 1967 for break analysis summary), most of the residual heat would be removed by the coolant before the coolant is expelled from the primary system to the containment. For such breaks, the present emergency core cooling system will provide adequate cooling for the core including the 6 high performance fuel bundles.

For the very small breaks, where the system would not depressurize to permit the low-pressure core spray system to function, the core mid-plane could be exposed and fuel damage could occur. The slow loss of water, which keeps the system pressurized after the water level has decreased below the core mid-plane, prevents fuel cooling by the 400 gpm core spray header until the pressure is reduced below 140 psig. Because of this behavior, much of the normal fuel rod clad will be overheated and perforated before the spray system can deliver water to cool the fuel. The presence of the six high performance bundles does not significantly alter the consequences resulting from breaks of this size.

In the design basis accident (DBA) rapid depressurization of the system occurs allowing the spray system to perform its cooling function. If the spray system cools the core as effectively as expected, peak clad temperatures in the normal Big Rock Point fuel will not exceed 1500°F. However, the peak clad temperatures for some of the 188 high power density rods will reach temperatures as high as 3500°F. The capability to cool the fuel by spray water without excessive damage and fuel configuration changes after initial clad temperatures exceed 2500°F has not been demonstrated. Therefore, we believe that the presence of the high performance fuel in the Big Rock Point core makes the consequences of the design basis accident somewhat greater than with the normal core, even assuming that engineered safety features including emergency core spray water and containment isolation perform as designed. Conservatively, we have assumed that the entire fission product inventory is released from the six developmental fuel bundles. With this assumption, the maximum site boundary doses are approximately as previously reported in the Big Rock Point Final Hazards Summary Report for the 10% core melt case. The large break accident is less severe in terms of radioactive release, and is considered less probable than a small break accident.

Our conclusion, based on a review of all relevant information, is that in the more probable range of primary coolant rupture accidents, i.e., the small breaks, the increase in hazard caused by the high power density rods is negligible. For the larger breaks, the magnitude of the radioactive release is larger than it would have been for a normal core, but is still less than the radioactive release previously evaluated as acceptable. Therefore, based on a relative evaluation of the Big Rock Point core with and without the high performance fuel, we believe that the consequences of loss of coolant do not represent an unacceptable risk to the health and safety of the public. In addition, the installation of a new 44 kv power supply enhances the probability that high pressure feedwater pumps will continue to deliver water to the reactor vessel after a primary system rupture. This improvement in power reliability increases the probability of maintaining high pressure feedwater flow to the steam drum following primary system ruptures as large as 11.5 square inches. With high pressure water delivery capability in excess of 16 minutes, the core will not be uncovered. Under these conditions the fuel, including the developmental fuel, would not be damaged.

REACTIVITY EXCURSIONS

We have reviewed the reactivity excursion which could occur if one of the control rod blades immediately adjacent to one of the two small diameter developmental fuel bundles becomes separated from its drive shaft, sticks in the fully inserted core position while the control rod drive shaft is fully withdrawn, and subsequently drops out of the core at a time when the maximum reactivity excursion would occur. For the purpose of our evaluation it is assumed that the center-to-center spacing between the developmental bundle and another small diameter developmental bundle is at the minimum spacing limitation of 16.5 inches, set forth in the technical specifications. The small diameter rods were selected for the evaluation because they contain the smallest mass of active fuel to accumulate the energy released during the transient, and they will, therefore, experience the sharpest fuel enthalpy rise. If the fuel reaches temperatures in excess of boiling temperatures, the internal pressure will increase rapidly. The licensee has assumed that when 425 cal/gram fuel enthalpy is attained, as a result of very rapid power transients, prompt rupture of the clad will occur and the fuel, which has reached 425 cals/gm, will be instantly dispersed in the form of small spherical particles into the coolant. The vaporized portion of the fuel will transfer its energy instantly to the water and the remaining energy will be transferred on a time dependent basis, depending on particle size. The lowest and, therefore, most conservative time constant of 4 milliseconds for particles of 20 mil size was used to calculate the transfer of the remaining heat to the water. The applicant believes that unless the threshold of prompt failure (425 cals/gm) is exceeded, there is no mechanism for prompt fuel dispersal and rapid energy conversion; therefore, primary system integrity would not be affected. The results of reactivity excursions with peak energies in excess of 425 cals/gm, calculated in the manner described, indicate that peak enthalpies of 625 cals/gm would be required to cause the unrestrained vessel to lift 0.5 ft. Similarly, the applicant shows that, if the peak fuel enthalpy reached approximately 800 cals/gm, the energy release associated with this condition would exceed strain limits and cause vessel rupture.

The effect of lowering the prompt threshold failure was evaluated by the applicant for an energy release spectrum associated with a rod drop accident of 0.021 delta k/k. It was shown that excessive vessel strain or vessel movement would not occur for the same power excursion with prompt failure thresholds as low as 230 cals/gm.

All of the calculations assume reactor shutdown by the Doppler effect alone, followed by control rod scram within the normal 290 milliseconds.

To produce the maximum reactivity insertion assumed in the evaluation, the following compounded errors and failures must occur while the reactor is in the hot standby condition (HSB): (1) violation of normal operating procedures by fully withdrawing a single control rod of maximum worth rather than equally withdrawing control rods in banks; (2) this particular rod must stick in the core and simultaneously become separated from the drive, and (3) the stuck control rod then must free itself and drop from the core at a precise time to cause the maximum reactivity excursion. If any of these conditions do not prevail, the excursion either would not occur or would be of a smaller magnitude.

We have concluded that a reactivity excursion resulting from dropping a control rod worth 2.1% delta k/k, as described by the applicant, is extremely improbable but is a reasonable upper limit for the comparative evaluation of reactivity accidents with and without centermelt fuel. This conclusion is based on the following considerations:

1. The irradiation period of the developmental bundles will be limited to approximately 24-30 months.

2. The two small diameter rod fuel bundles with the least fuel and highest potential fuel enthalpies can affect the accidents associated with only 2 of the 32 control rods in the Big Rock Point core.

3. The developmental fuel bundles will be located in the center region of the core, rather than the periphery where maximum control rod worths would occur.

4. The probability of the rod drop accident is minimized by written procedures which govern operator movement of control rods and assure that control rods are latched to their drive shafts prior to initiating control rod withdrawals for approaches to criticality.

5. There has been no evidence of the poison section of control rods sticking or binding within the cores of existing boiling water reactors.

Investigation of the control rod drop accident, while at power condition and with operator errors equivalent to those assumed in the HSB analysis, indicates the maximum control rod worth to be only 0.95% delta k/k compared to 2.1% delta k/k for the HSB rod worth, and the resultant excursion energy would be only 75 cals/gm. This value, when added to the normal energy while at power, results in peak fuel enthalpy of approximately 280 cals/gm compared to 450 cals/gm for the HSB excursion. Since the latter case is more severe, our evaluation is based on the HSB condition.

Because of the time dependence for prompt release of the fuel energy to the coolant during the reactivity excursion, as described by the applicant, and the lack of suitable tests to confirm the overall validity of the calculational model and assumptions, we cannot accept without reservation the conclusions based on these calculational methods. The application indicates that, with a peak energy density of 450 cals/gm for the control rod drop accident, the instantaneous release of the transient energy in fuel above 320 cals/gm would correspond to about 32 MW-sec energy burst. Similarly, 64 MW-sec of energy would be released promptly if all of the energy above 230 cals/gm were considered. Since fuel melting begins at enthalpies of about 220 cals/gm and is complete at about 280 cals/gm, we believe that the prompt energy release values derived in the above manner are reasonable boundaries. Our independent calculations, based on equivalent explosive energy releases, give vessel lift and strain values higher than the applicant's estimation. We have determined that vessel lift could be as much as 0.4 ft and maximum vessel strain will be below 0.7%. We consider strain of 5% and lift below 0.5 ft to be acceptable as this amount of vessel movement would not damage major primary piping. We therefore believe the integrity of the primary system will be maintained throughout a reactivity excursion resulting from dropping a control rod worth as much as 2.1% delta k/k although there is a possibility that the 3-inch spray header pipe connection near the top of the reactor vessel might be damaged if all of the energy in the fuel above 230 cals/gm is released promptly. If the spray header should fail, the core mid-plane would not be uncovered for at least 18 minutes following a double-ended break of this 3-inch pipe. To provide a backup for the core spray system, we are requiring an additional means of introducing water from the fire main as an emergency core cooling system. This system will supply enough water to keep the core covered in event the spray system is damaged so that fuel temperatures will not rise excessively and core geometry will be preserved. The valves to activate this sytem will be located in an accessible location and will be operated manually.

Most of the damage resulting from a control rod drop under the conditions described would be in the adjacent high performance fuel bundle with significantly less damage to other bundles in the immediate vicinity of the dropped control rod. We have concluded that sufficient control rods will scram within 0.290 seconds of scram signal activation to restore subcritical reactor conditions. Soluble poison may be added during the 18 minute cool down and depressurization period prior to addition of core flooding water from the fire system.

TECHNICAL SPECIFICATIONS

We have reviewed the proposed modifications of the technical specifications submitted by the applicant in conjunction with this high performance fuel irradiation program. The requested modifications include descriptions of the developmental fuel assemblies, and special operating and administrative restrictions appropriate for operation with the developmental fuel. We have made some changes in the proposed modifications in the interest of clarification and two new specifications have been added. The new specifications require: (1) removal of the advanced performance developmental bundles until the results of initial destructive examinations have been evaluated and (2) provisions for utilizing water from the fire main for core cooling as a supplementary core cooling system.

The changes made in the proposed modifications have been discussed with the applicant and he is in agreement with the revisions.

CONCLUSION

We have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation of the Big Rock Point Plant with the developmental fuel described in the applicant's Proposed Change No. 13. We believe, therefore, that the Technical Specifications of License No. DPR-6 may be revised as indicated in Attachment A to Amendment No. 1.

Donald J. Skovholt

Donald J. Skovholt
Assistant Director for Reactor Operations
Division of Reactor Licensing

Date: JAN 30 1968