TABLE 1

COMPARISON OF MINIMUM CRITICAL HEAT FLUX RATIO (MCHFR) WITH NEW AND OLD CORRELATION

	Power Shape		C Type Fuel at 122% Power			
	Radial Factor	Axial Factor	Old Critical Heat Flux Correlation		New CHF Correlation	Ratio New/Old
Beginning of Life Many Control Rods Inserted	1.4	1.45(6)	MCHFR	1.99(8)	2.23(7)	1.12
			Steam Quality Xc	0.151	0.1265	
			Q/Ac BTU/hr ft^2	273,000	325,000	
			Q/Ap BTU/hr ft^2	342,000	342,000	
Middle of Life	1.32	1.652(5)	MCHFR	1.92(5)	2.32(7)	1.21
			Xc	0.0415	0.116	
			Q/Ac	374,000	326,000	
			Q/Ap	374,000	374,000	
End of Life All Control Rods Removed	1.623	1.335(2)	MCHFR	1.42(8)	2.11(6)	1.48
			Xc	0.2205	0.167	
			Q/Ac	184,000	292,000	
			Q/Ap	392,000	392,000	

Xc = Steam quality at the location of MCHFR
Q/Ac = Heat Flux at the location of MCHFR
Q/Ap = Peak Heat Flux

Axial length of fuel divided into nine nodes. Numbers in parentheses indicate node number where node number 1 is at the bottom of the fuel.

Local power factor of 1.3 used for all cases presented (TWX - Feb. 6, 1967).

122% overpower - basis 120% scram set point -- reference p. 26, Amendment 14, Nov. 15, 1963.

Vibratory compacted zircaloy clad fuel -- 109-0.449 inches diameter rods -- 12-0.34 inch diameter corner rods, 11 x 11 fuel rod array -- 0.577 inch rod pitch.

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From the above table, it can be seen that the peak local heat flux, 392,000 BTU/hr ft^2 at 122% power level, occurs at the end of core life and is 5% greater than the middle of core life peak heat flux. Also, it is evident, since the minimum critical heat flux ratio at the end of core life is 1.42 for 122% power, using the old correlation, and below the tech spec MCHFR of 1.5, that rated power at the end of life would not be permitted. Toward the end of an operating cycle when the central control rods are finally withdrawn from the bottom of the core, the flux peak moves toward the bottom as the axial factor column above shows, e.g., at the beginning of life, the axial peak occurred at node 6 in contrast to node 2 at the end of life. The greatest gain in MCHFR according to the table above is in the high quality region where the ratio of the new-to-old MCHFR is 1.48.

Although the peak heat flux is 5% greater than the mid-life value, the heat flux at the MCHFR location is 10% less at the end of life than at the beginning or middle of core life as the table shows when the new CHF correlation is used.

EVALUATION

Consumers Power Company has proposed that the new General Electric CHF correlation presented in APED 5286 be used to update the Big Rock reactor thermal hydraulic parameters to the most recent and representative numbers. The present CHF correlation was based on approximately 1000 data points, essentially all of which were from tests of single, internally heated rods with annular coolant flow. Since the corner rod was clearly design limiting before it was economically feasible to incorporate enrichment variation within each fuel element, the geometry was selected to be representative of this rod. The few multi-rod points available at the time fell significantly and consistently above the single rod design line.

The new CHF correlation is based on approximately 700 multi-rod data points taken on four-rod and nine-rod test sections. Since the corner rod geometry, rod spacing, pressure and flow correspond to actual boiling water core conditions according to General Electric, the correlation (as can be observed in the comparison below) is valid for General Electric boiling water reactors including Big Rock Point.

TABLE 2

		Multi Rod Geometries to Develope The CHF Correlation	*Big Rock Point Fuel Bundles	
Heated Length	inches	36, 45, 48 and 60	70.	
Hydraulic Diameter	inches	0.324 - 0.485	0.508	(1)
Rod to Rod Spacing	inches	0.095 - 0.187	0.129	
Rod to Channel Spacing	inches	0.060 - 0.135	0.157	(2)
Pressure	psia	600 - 1450	1350	
Flow Rate	lbs/hr ft^2	0.2 - 1.5 x 10^6	.46 - .7 x 10^6	
Steam Quality		0 - 0.6	.23 maximum	
Heating Distribution		Uniform and Increased Heat in Corner Rod or Central Rod	non-uniform	

*Information received by telephone from Consumer's Power Company March 3, 1967.

(1) The effects of hydraulic diameter appear to be negligible over the range from 0.3 to 0.5 inches. (Ref APED-5286)
(2) The rod-to-channel spacing is important in determining the critical heat flux. Its effects become noticeable for very small rod-to-channel distances, i.e., lower than the range examined. (Ref APED-5286)

Great care was exercised, according to G.E. to make sure that the test geometries were representative of boiling water reactor fuel arrangements in order to match the anticipated local flow and steam quality conditions, since the application of four and nine rod data points is justified only if the flow and enthalpy redistribution takes place in the same manner in the test sections and reactor fuel assemblies. The test rods consisted of Inconel-X tubes through which electrical current passed to simulate the nuclear heat generation.

A multichannel model was developed to predict the coolant behavior in complex geometries, including reactor fuel assemblies. The model subdivides the geometry into several parallel and adjoining channels which run over the entire flow length. Each channel has its own characteristic hydraulic diameter and heat input and the effects of boiling in some or all of the channels are included within the accuracy of engineering correlations of two phase flow effects. The mass flow rate into the channels is assumed to be uniform, and local flow in each channel is obtained by allowing the flow to redistribute itself between the various channels until they all sense the same pressure at each axial position. The best fit CHF correlation for an internally heated annulus was employed for the corner rods, and circular pipe correlations were used for the spaces between the rods. The proportionality constant, i.e. the mixing constant was adjusted to obtain acceptable agreement between model and test critical heat values. The uncertainty associated at this time with these mixing constants is one of the reasons for developing multi-rod correlations based on average axial enthalpy or steam quality, and average flow for the fuel bundles rather than local CHF water properties. This analytical model was applied to the four and nine rod test sections for which critical heat flux data are available. In this manner the use of four and nine rod test results to predict the critical heat flux in reactor fuel assemblies containing up to 121 rods has been justified.

The CHF limit lines described by the proposed new multi-rod G.E. correlation were derived with minimal rod spacers in the test assembly. The rod-to-rod and rod-to-channel spacing of the test assemblies was maintained by slender spool type spacers designed to have negligible influence on the flow. Other tests were performed to show that fuel rod spacers used in boiling water reactors could have a beneficial effect by promoting mixing and thereby increasing the critical heat flux, although in this particular Big Rock application the benefits of improved mixing have not been considered in calculating the MCHFR.

G.E. has stated that the new CHF correlation based upon multi-rod test data cannot be applied indiscriminately to any geometry and in particular, the correlation is valid only for lattices typically found in boiling water reactors. We have listed the important physical characteristics of the Big Rock Point fuel geometry in Table 2. Further, we have been assured by G.E. that the new correlation also accounts for local peaking factors of the magnitude to be encountered in Big Rock cores, i.e. 1.24.

For the present core configuration:

1. All of the original stainless steel clad fuel bundles yet remaining in the core are in the outer low power region where the flow is also reduced. (Most of these fuel assemblies will be removed during the June 1967 refueling.)

2. The central, high power region of the core, is occuplied by type B and C zircaloy clad fuel assemblies which utilize two zone enrichment with the outer two rows containing the lower enrichment fuel rods. Three small diameter rods as in the original fuel assemblies (in some cases

cobalt targets replace one of the three small diameter corner rods) are provided at the corners of the fuel bundle to improve heat transfer conditions in the regions which are normally limiting.

3. The corner (or adjacent rod where coablt targets are substituted for the corner rods) rod although smaller in diameter ard of lower enrichment than the inner bundle fuel rod enrichment is the thermally limiting rod, i.e. the location of the minimum critical heat flux ratio (MCHFR). Consumers Power Company reported (telecon April 21, 1967) that although the highest local peaking factors with the improvements described above is 1.2, they will continue to use 1.3 for additional conservatism in their calculations. (Refer to Section 5.1.5 of the Technical Specifications for Big Rock Point Nuclear Power Plant for fuel bundle description.)

Earlier fuel failure following the maximum credible accident of a loss of core coolant, is an undesirable characteristic inherited with the increase in local power peaking resulting from the new General Electric CHF correlation. In response to this concern Consumers Power Company reported by telephone on April 26, 1967, that fuel cladding perforations would occur several seconds earlier than most recently reported in the evaluation which accompanied Proposed Change No. 8 December 23, 1965. Further, the curve representing percentage of fuel rod perforations as a function of time after MCA will cross the old curve (see attached figure) when 20 to 30% of the fuel clad has failed. The volume percent of UO_2 fuel over the 3000°F clad melting temperature is not noticeably affected. The significance of these fuel rod failure changes caused by increased core power peaking in relation to previously reported post accidents conditions summarized in the attached curve is in our opinion negligible.

In summary, our conclusion that the new General Electric critical heat flux (CHF) correlation for the Big Rock Point operational thermal calculations may be safely used is based on the premise that the test assembly heat fluxes and geometries suitably match the Big Rock Point boiling water reactor conditions and is further supported by the following factors:

1. Rated power level of 240 MWt will remain unchanged.
2. Average power density at rated condition 46 MW/liter will remain unchanged.
3. The minimum critical heat flux ratio, MCHFR, of 1.5 at 122% overpower continues in effect.

4. The Technical Specifications limiting heat flux and fuel rod power remain unchanged at 530,000 BTU/hr ft^2 and 17.2 KW/ft respectively. (392,000 BTU/hr ft^2 heat flux which corresponds to 122% overpower at end of core life in the representative results presented in Table 1 is equivalent to 10.3 KW/ft for the small 0.344 inch diameter corner rods.)
5. The average fuel bundle exit steam quality will remain unchanged at 8.1%.
6. Peak power generation in the corner rods prior to MCA is, consistent with the information presented above, 8.45 KW/ft, and although this value is approximately 5% greater than the midlife peak value for the same core configuration we believe this is an insignificant change.

CONCLUSION

We have concluded, for the reasons stated above, that Proposed Change No. 12 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.

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

Donald J. Skovholt

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

Date: May 26, 1967

ATTACHMENT A

CONSUMERS POWER COMPANY

CHANGES TO TECHNICAL SPECIFICATIONS

LICENSE NO. DPR-6

CHANGE NO. 12

1. Delete the footnote in section 5.2.1(b) and replace with:

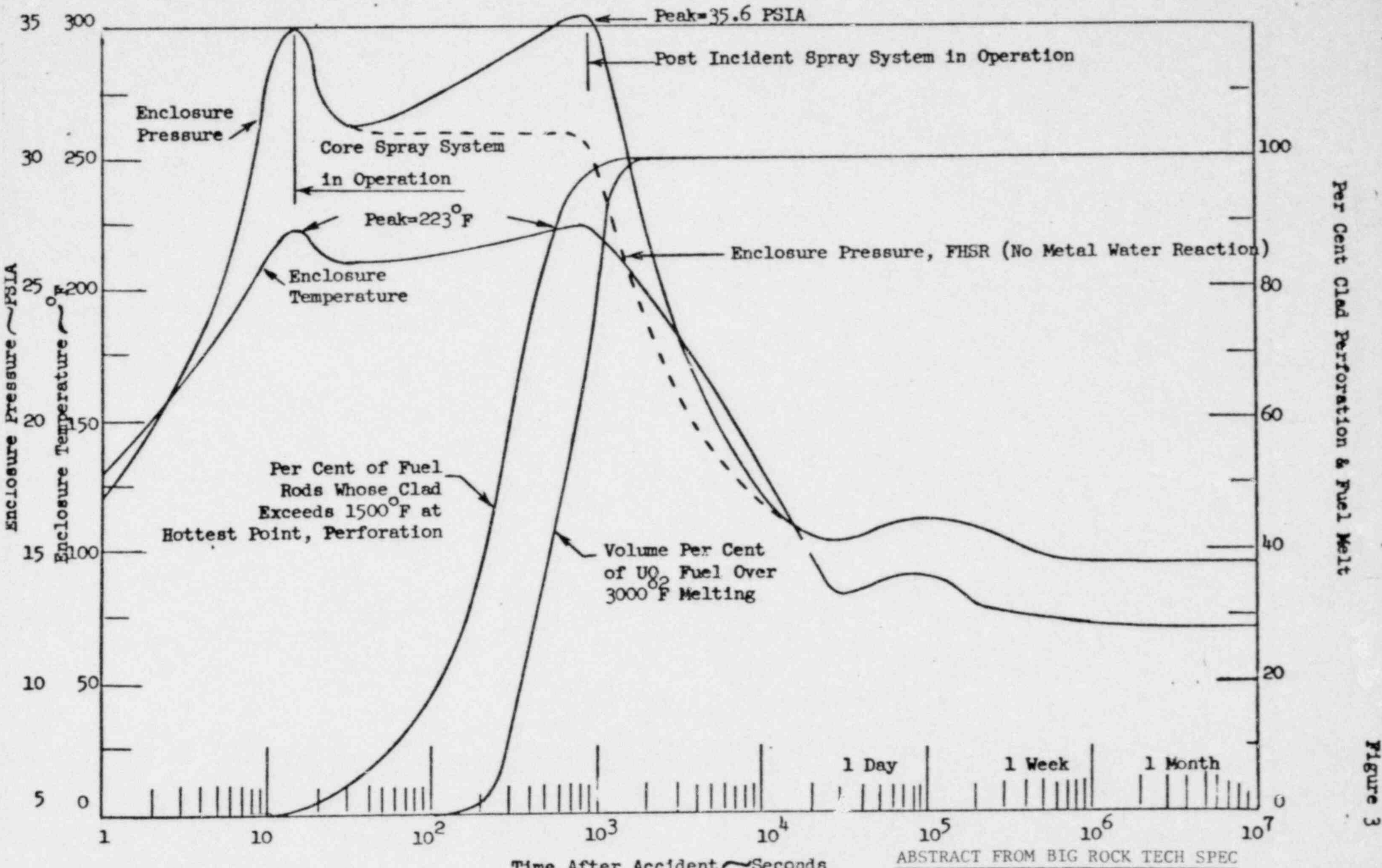
   "*Based on correlation given in "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," by J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, September 1966 (APED 5286 and APED 5286, Part 2)."

2. Delete the footnote in section 8.3(a) and replace with:

   "*Based on correlation given in "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," by J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, September 1966 (APED 5286 and APED 5286, Part 2)."

BIG ROCK POINT

Transients of Reactor System Rupture For
Maximum Credible Accident Evaluation With Metal Water Reaction



ABSTRACT FROM BIG ROCK TECH SPEC
PROPOSAL #8 (DECEMBER 23, 1965)

Figure 3