Consumers Power Company

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

August 15, 1967

Dr. Peter A. Morris, Director
Division of Reactor Licensing
United States Atomic Energy Commission
Washington, D. C. 20545

Re: Docket 50-155

Regulatory Suppl File Cy.

Dear Dr. Morris:

Attention: Mr. D. J. Skovholt

Transmitted herewith are three (3) executed and thirty-seven (37) conformed copies of Additional Information in Support of our Proposed Change No. 13 dated May 26, 1967.

This Additional Information is being transmitted to you in conformance with your letter request dated July 21, 1967.

Yours very truly,

Robert L. Haueter

Robert L. Haueter
Assistant Electric Production
Superintendent-Nuclear

RLH/wf
Attach.

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CONSUMERS POWER COMPANY

Docket No. 50-155

License No. DPR-6

Answers to Additional Information Requested

Regarding Proposed Change No. 13

Question: 1. Reactivity Insertion Accidents

1.1 It was previously reported (supplement to Big Rock Point Technical Specification Change No. 10) that a gravity drop of a rod worth .04 delta k/k could cause a maximum vertical displacement of the vessel of 0.17 ft and 0% maximum vessel strain. The corresponding limiting rod worth value for the rod ejection accident was reported to be .02 delta k/k.

How have these limiting rod worth values changed for the "C" core with and without the proposed special "center-melt" fuel bundles as a result of the more realistic analysis which prevents the second power burst due to the steam explosion in the vicinity of the dispersed high enthalpy fuel rods?

Answer: The following table lists the limiting rod worths for the .17 ft vessel displacement and zero vessel strain. Also listed are the values for the .5 ft displacement criterion. The heat transfer rates used are those calculated from the TREAT-ANL powder tests. The flow restraint used is thought to be a median value. However, as indicated in the Change 13 submittal, other values can be calculated with reasonable logic so that no one set of calculated results should be accepted without the knowledge that others are feasible.

LIMITING REACTIVITY INSERTIONS

Accident	Rod Drop		Rod Ejection	
Criterion*	A	B	A	B
"C" Core (Chg. 10)	.04 Δk	.045 Δk	.02 Δk	not calculated
"C" Core (chg. 13)	>.045	-	.028	.031 Δk
"C" Core w/ centermelt	.045	>.045	.026	.028

* Criterion A: Vessel displacement of .17 ft., zero strain

Criterion B: Vessel displacement of .5 ft., about 1% strain

Question 1.2: A possibility exists that the very high enthalpy states in the "centermelt fuel" bundles adjacent to the dropped control rod could cause extensive core damage impairing the control rod scram and core cooling capability. Please discuss the analytical methods and test results which assure that core geometry is preserved following control rod drop accidents with rod worths as high as .025 delta k/k.

Answer: The postulated drop of a rod worth .025 Δk will yield about 20 kg of UO_2 promptly failed. This material will be essentially all in a centermelt bundle that is adjacent to the dropped rod.

Conservatively assuming an instantaneous pressure rise, an acoustic pressure wave will be sent out in all directions. That portion of the wave in the horizontal direction will encounter the core material, fuel rods, channels, and inserted control rods, before reaching the vessel wall. Each of these materials will absorb some of the wave energy. However, due to the relative incompressibility of the moderator gross movements are possible only if the restraining pressure vessel were to expand. It has been calculated for this postulated accident that the vessel is not strained. In addition the wave encompasses a bundle in such a short time that a differential force on the bundle exists for less than a millisecond. The pressure wave in the vertical direction will transit the distance to the free surface in a 1.6 milliseconds. The reflected rarefaction or unloading wave will reach the failure zone in another 1.6 ms at which time Newtonian mass movement will commence in the vertical direction. It is this movement that gives the pressure relief in the failure zone and prevents vessel expansion and gross core displacement.

After the initial spike of several thousand psi and a few ms in duration a pressure several hundred psi above the initial system pressure is calculated to exist for about 100 ms. This pressure could collapse the fuel channels. Stress considerations indicate that the mode of collapse would normally be concave onto the fuel rods. However, it is possible to get some convex deformation on one or two sides of the channel. Control blades adjacent to channels that deformed normally, or did not deform at all, could scram. Those few control blades that encountered deformed channels which completely closed off the blade path might have trouble completing the entire stroke. However each control drive has a force of about 700 lb. available to overcome restrictions. All the control rods need not scram to effectively bring the core to a subcritical situation. The scrammable rods at the hot standby condition are worth a negative .25 Δk so that only a fraction of the bank need be inserted.

It is not expected that the channel containing the ruptured fuel will completely maintain its integrity; however due to above circumstances it is believed that the control rod system will be capable of shutting down the reactor after the postulated accident.

In addition to the scram system there are other delayed negative reactivity sources that are available. These are not to be confused with the mechanisms which eliminate the second power burst, i.e., fuel dispersal, steam exposion and flux spatial shift. There is considerable energy to be transferred from the fuel rods as they begin to cool. This energy will cause boiling and negative reactivity. The reactor operator can use the high pressure liquid poison system if there is not enough shutdown capability. This system can deposit a negative .16 Δk available.

Any channel collapse will reduce the coolant flow area and consequently the effective cooling. However the recirculation system is pumping full flow throughout the above events and should provide adequate cooling in all regions with the possible exception of the fuel adjacent to the destroyed centermelt fuel bundle. The MCA as analyzed in the FHSR and amended in Change 8 of the Technical Specifications demonstrates containment integrity for at least 26.8% of the core zirconium reacting with the water. It is not expected that the results of this postulated accident will be nearly as severe.

The above discussion is not meant to be a definitive answer; we do not believe such an answer is available until some destructive tests are performed on an oxide core. The only tests that have been performed which in any way closely resemble the postulated accident are the SPERT 1 oxide core tests.* The primary power bursts in both these tests were prematurely terminated by fuel rod failure and dispersal.

Question 1.3: What is the flux suppression factor at the center of the "centermelt fuel" rods, particularly the 0.700 inch diameter fuel rods, and how is the fuel enthalpy gradient across the fuel rod cross section considered in assessing the extent of fuel and core damage?

Answer: The flux suppression factor of the various centermelt fuel rod types is listed below. The factor (FSF) is defined as the ratio of the power on the fuel surface to the radial average.

Fuel O.D.	Enrichment	FSP	Number of rods per bundle
.7	5.6	1.1	5
.7	5.0	1.08	12
.7	4.3	1.07	12
.57	5.6	1.07	8
.57	5.0	1.06	16
.57	4.3	1.05	12

It can be seen that the highest suppression value is a 10 percent increase in five of the "high performance" fuel rods. However, the specific power, kW/kg, of the "intermediate performance" fuel elements is higher than that in the "high performance" bundles. Thus, for conservatism in the postulated nuclear excursion, the dropped control rod should occur adjacent to an 8 x 8 fuel bundle. As noted in the above table this gives a maximum higher surface energy density by 7 percent in 8 rods.

* GRUND, J.E., Editor, "Experimental Results of Potentially Destructive Reactivity Additions to an Oxide Core", IDO 17028.

The calculations do not take this anomaly into account because of its small effect. Theoretically it would change the energy density threshold of prompt rupture by 5 to 7 percent in the 8 x 8 bundle. Because of the lack of experimental data on the threshold and the steep slope of UO_2 vapor pressure versus temperature this change of threshold is thought to fall within the uncertainty band of these calculations.

Although there may be an effect on the fuel rod breach threshold there should be little effect on the thermo-hydrodynamic calculations, and thus little effect on the primary system breach threshold. This is because the thermodynamic calculations now assume that the promptly dispersed material is in small particles that would not be made smaller by a radial temperature gradient. In fact, some knowledgeable people have suggested that such a gradient would have an imploding rather than exploding effect on the fuel rod and thus tend to retard heat transfer.

2. Loss of Coolant Accident

It has been stated in Proposed Change No. 13 that, as a consequence of the duration of the postulated primary system blowdown (greater than 4 seconds) and the high heat transfer rates expected in the core during blowdown, the centermelt fuel rod temperatures will be reduced to a level characteristic of the ensuing decay power generation and are thus virtually independent of the initial stored energy content. The fuel, however, heats up again during the time interval of film blanketing following the blowdown and until core cooling is effective.

Question 2.1: Please discuss the analytical methods and input assumptions for determining peak fuel rod temperatures, and present comparative maximum temperatures for "C" and "centermelt" fuel rods after the MCA until the temperature rise has been arrested by fire main water sprayed onto the core via a single spray header inside the reactor vessel.

Answer: The fuel thermal transient during and after blowdown is analyzed using a digital code which treats the core as five radial zones with five axial nodes per zone. Each radial zone is further divided into four zones and four different rod types are considered in each fuel bundle. This allows accurate modeling of power distribution by including the axial, radial and local rod peaking factors. The fuel volume within the clad is also nodalized radially. The code considers decay power, stored energy in the core, energy contributions of any chemical reaction that may occur and the thermal radiation and convection between fuel rods and fuel channels. It calculates the cladding and channel temperatures, degree of metal water reaction, hydrogen and energy release and other parameters of interest on a nodal basis as a function of time. Fuel channels are included and treated on a nodal basis.

The convective heat transfer from the fuel rods to coolant passing through the core is determined by using the correct heat transfer coefficient for each case. The heat transfer coefficient used in the code during the blowdown phase is obtained from correlations determined experimentally by the General Electric Company. These experiments indicate that a high nucleate boiling coefficient is effective for a short time (the "dryout" time) after blowdown is initiated, and then drops monotonically to zero at the conclusion of the blowdown.

These calculations are felt to be conservative in all respects and to give the maximum expected temperatures. The following are important features of the core heat-up calculations:

1. Bundle power is decay power corresponding to the <u>peak</u> bundle (3.25 MW_t) even though all bundles will not operate at peak position.
2. Four rod types are considered in each bundle, each with 5 axial nodes. All high power centermelt rods were grouped at the center of the bundle in the model, while, in reality, they are mixed with "cold" rods which provide excellent heat sinks since liquid films are more easily established. This

grouping at the center is the worst case and gives conservatively high temperatures.

3. The bundle is insulated. That is, the effect of spray water flowing <u>between</u> bundles has been conservatively neglected. Recent spray cooling tests conducted by General Electric with and without outside-of-channel coolant have shown an outer coolant flow of 20% of inner coolant which results in lower peak temperatures of 100 to 200 °F.

4. Blowdown corresponds to the maximum credible accident; i.e. a break in a recirculation line approximately equal to 6.5 ft^2.

5. Using the above conditions, "dryout"times were calculated from GE test data; results for "dryout" times of 1.7 and 3.0 seconds are presented.

6. Spray cooling systems are expected to be initiated in 15 seconds. However, the time for completely effective spray cooling is assumed to be 30 seconds. Calculations are also included for 15 sec. and 45 sec.

7. The effective heat transfer convection coefficient is determined from GE tests and ranges from 3-5 Btu/hr-ft^2-°F.

8. Liquid films are assumed to be established on the channel in about 100.seconds after spray is initiated.

9. An axial peaking factor of 1.31 was assumed.

Comparative <u>maximum</u> clad temperatures calculated with this model for 7 x 7 high-performance, 8 x 8 intermediate, and 11 x 11 "C" -type fuel bundles are shown in Figure 1. The calculations show that the greater energy storage of the centermelt fuel results in higher peak temperatures than the bulk of the BRP bundles. Maximum (centerline) fuel temperatures are compared in Figure 2. Note the short time that centermelting remains after the accident.

Question 2.2: Indicate the sensitivity of the fuel rod temperatures to duration of blowdown and core spray initiation delays.

Answer: Fuel rod temperatures show little sensitivity to dryout time (see Figure 3) or short term core spray initiation delays (see Figure 4).

Question 2.3: Discuss the applicability of the test data presented in Figure 25 of the referenced proposal, which is based on initiation of core cooling (core spray) before fuel temperatures reach 1250°F.

Answer: The spray-cooling data of Figure 25 of the proposal for Change 13 were used to evaluate overall convection coefficients by comparing these data to the core heat-up model. The code is then used to predict core thermal behavior over a wide range of bundle powers and initial temperatures-such as the subject centermelt bundles. The effects of initial temperature are encompassed by the heat-up model.

The peak temperature predicted during a loss-of-coolant transient is a function of the bundle temperature at spray initiation. At a given power level, the magnitude of temperature rise (initial to peak) diminishes as the initial temperature is increased. This trend is shown by peak versus initial temperature relationships obtained in recent General Electric spray-cooling tests shown in Figure 5.

Question 2.4: Present clad stresses during the coolant blowdown period when the fuel rods are being rapidly cooled by ejected primary coolant.

Answer: Clad stresses during blowdown will be equal to,or less than,clad stresses during normal operation based on the following observations: Since the surface heat flux is the same as or less than normal operation, the temperature gradient across the clad and associated thermal stresses are reduced. The decreasing surface heat transfer coefficient results in increased clad temperature, coupled with a decreasing fuel bulk temperature. Therefore the clad stress due to possible fuel swelling is diminished. Clad stress due to internal fission gas pressure will not be significant during the blowdown period.

Question 3: Multi-Rod Critical Heat Flux Correlation

Please justify the use of the new GE Multi-Rod Critical Heat Flux Correlation in calculating "centermelt fuel" core performance limits, considering the spiked arrangement of the centermelt fuel rods within the bundles where power ratios of adjacent rods are approximately 18:1 in contrast to the normal situation where adjacent fuel rod power ratios are near to 1.

Answer: Salient features of the GE Multi-Rod Critical Heat Flux Correlation (APED-5286) are:

1. The correlation is based upon extensive critical heat flux data obtained in test geometries representative of boiling water reactor fuel arrangements. That is, the correlation consists of limit-lines drawn conservatively below some 700 CHF points obtained in 4-and 9-rod test sections with typical BWR flow and pressure conditions.
2. Application of these data to predict the critical heat flux in reactor geometries with up to 121 fuel rods is justified by the application of a multi-channel model which predicts coolant behavior and critical heat flux within subchannels of multi-rod geometries. An empirical mixing constant is determined from the 4-and 9-rod test results, and the model is then employed to show that the use of the Multi-Rod Correlation (which is based on average channel quality) is conservative for reactor geometries.

The "centermelt" fuel bundles are developmental fuel with non-standard BWR features. Specifically, the large variation in rod-to-rod power and larger rod diameters (for the high performance bundles) are outside the range of variables of the data upon which the Multi-Rod Correlation is based. A "correction factor" must, therefore, be applied to the correlation for application to these special bundles. This correction factor was determined by applying the Multi-channel Model to the centermelt bundles. The applicability of the Multi-channel Model to test sections with severe rod-to-rod peaking and slightly larger rods was ascertained by conducting special critical heat flux tests in such a nine-rod test section and comparing the model predictions to the test results. The agreement between Multi-channel Model prediction and the test data was excellent as shown on Figure 6.

The "correction factor" determined for the centermelt bundle configurations by applying the Multi-channel Model is unity. That is, the conclusion of the analyses and special tests conducted to determine the correction factor is that no correction is necessary. The GE Multi-Rod Correlation is applicable to the centermelt bundles.

Results of the detailed Multi-channel Model calculations are given in Table I. These calculations were based on 122% overpower conditions using the "worst" axial power distribution with axial peak of 1.4 average at 1350 psia. Note that all calculations show a minimum Critical Heat Flux Ratio greater than 1.5 at 122% overpower condition.

TABLE I

Comparison of MCHFR calculated by the Multi-channel Model based on local property values to MCHFR calculated using Multi-Rod Correlation using bulk properties:

Centermelt Bundle	Flow [$lb/hr\text{-}ft^2$]	MCHFR Multi-channel Model	MCHFR Multi-rod Correlation
7 x 7	790,000	1.54	1.58
8 x 8	740,000	1.56	1.53

Question 4: Calculational Accuracy

It has been stated that axial power shapes for the "centermelt fuel" bundles are known within 5% and that radial power shape for the highly enriched fuel spiked with depleted fuel rods has an uncertainty factor of 10%. Are these uncertainty factors included in the fuel temperature calculations during accident conditions? Are the performance limits based on minimum critical heat flux ratio of 1.5 at 122% overpower?

Answer: (In the first sentence of the question the word "radial" should read "total".)

Historically, the uncertainty factor for determining fuel rod powers has not been included when fuel temperatures are calculated for any condition, whether for steady state, or for postulated accident conditions. This applies to all fuel types that are presently licensed for operation in Big Rock Point Nuclear Plant as well as to the "centermelt" fuel assemblies proposed for insertion into the reactor.

The performance limits for "centermelt" fuel operating in Big Rock Point are as follows:

	8 x 8	7 x 7
1) Maximum Steady State Heat Flux, Btu/hr.-ft^2	500,000	500,000
2) Maximum Steady State Fuel Rod Power, kW/ft.	21.8	26.8
3) Minimum Critical Heat Flux Ratio at 122% of Rated Power	1.5	1.5

It should be pointed out that 1) and 2) above are given at rated conditions while 3) is at overpower.

CONSUMERS POWER COMPANY

By H R Wall
Vice President

Date: August 15, 1967

Sworn and subscribed to before me this 15th day of August, 1967.

Israel R. Warner
Notary Public, Jackson County, Michigan
My Commission Expires February 16, 1968

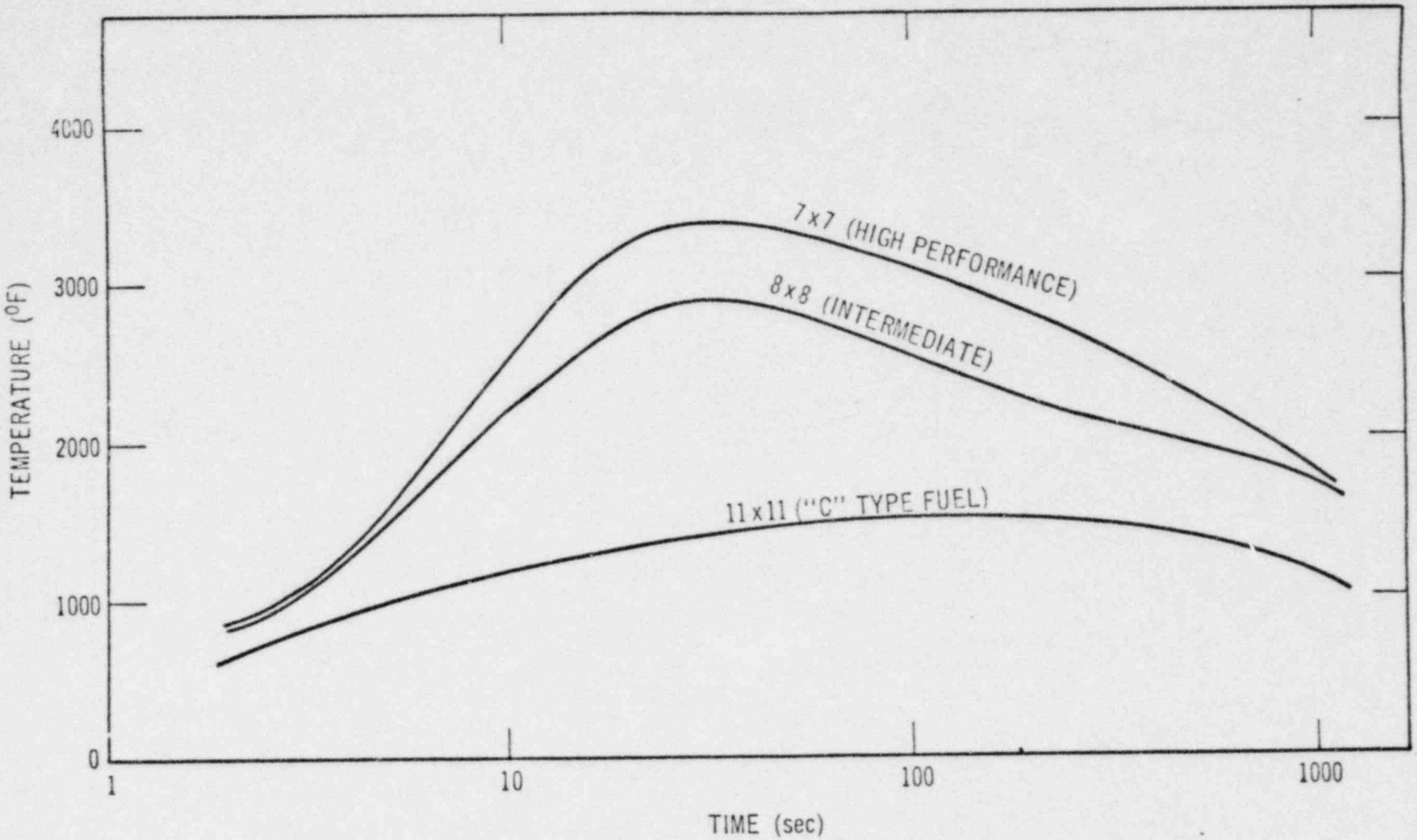

FIGURE 1. PEAK CLADDING TEMPERATURE FOR VARIOUS BUNDLES (L-O-C ACCIDENT)

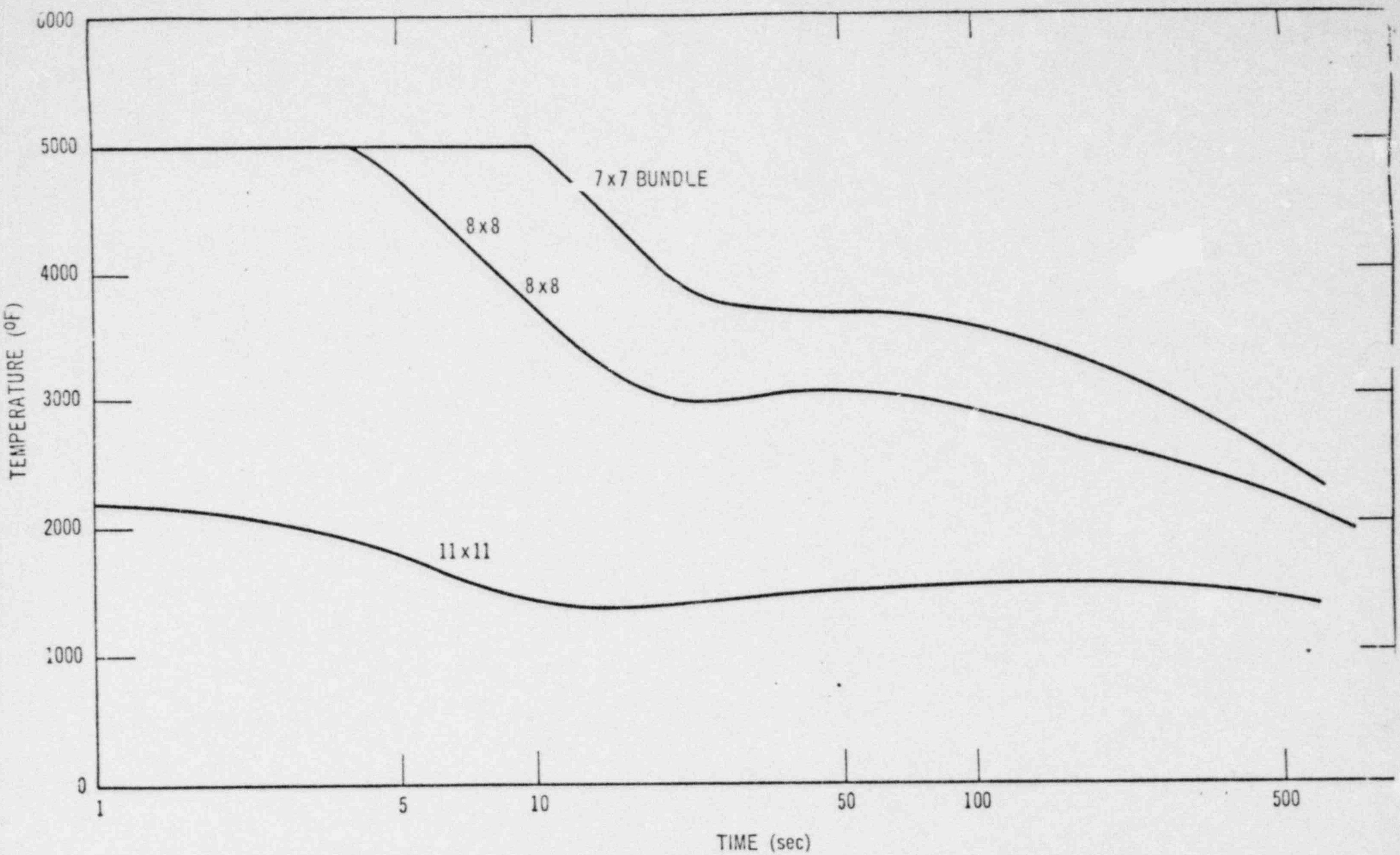

FIGURE 2. PEAK FUEL TEMPERATURE FOR VARIOUS BUNDLES (L-O-C ACCIDENT)

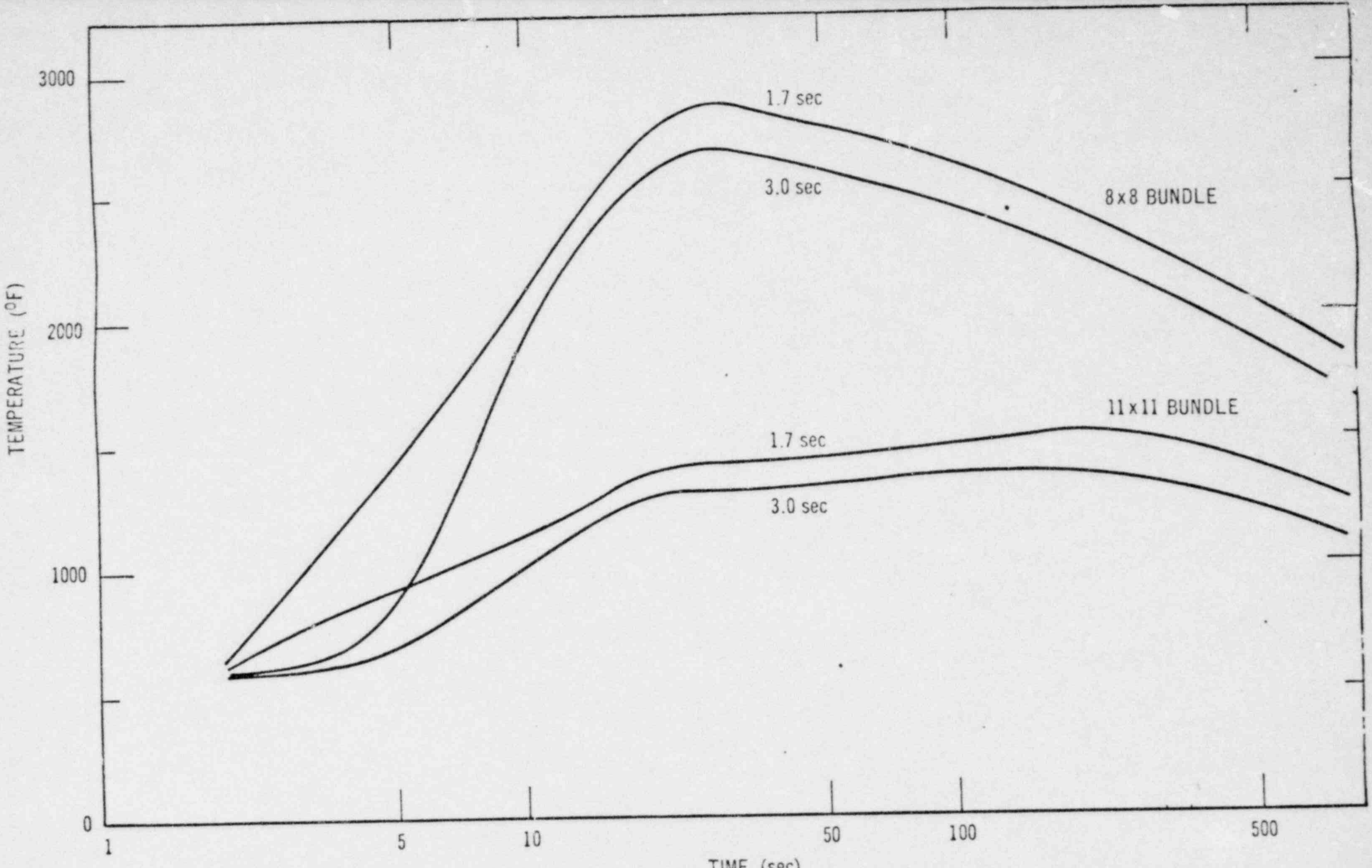

FIGURE 3. MAXIMUM CLADDING TEMPERATURE FOR DIFFERENT DRYOUT TIMES

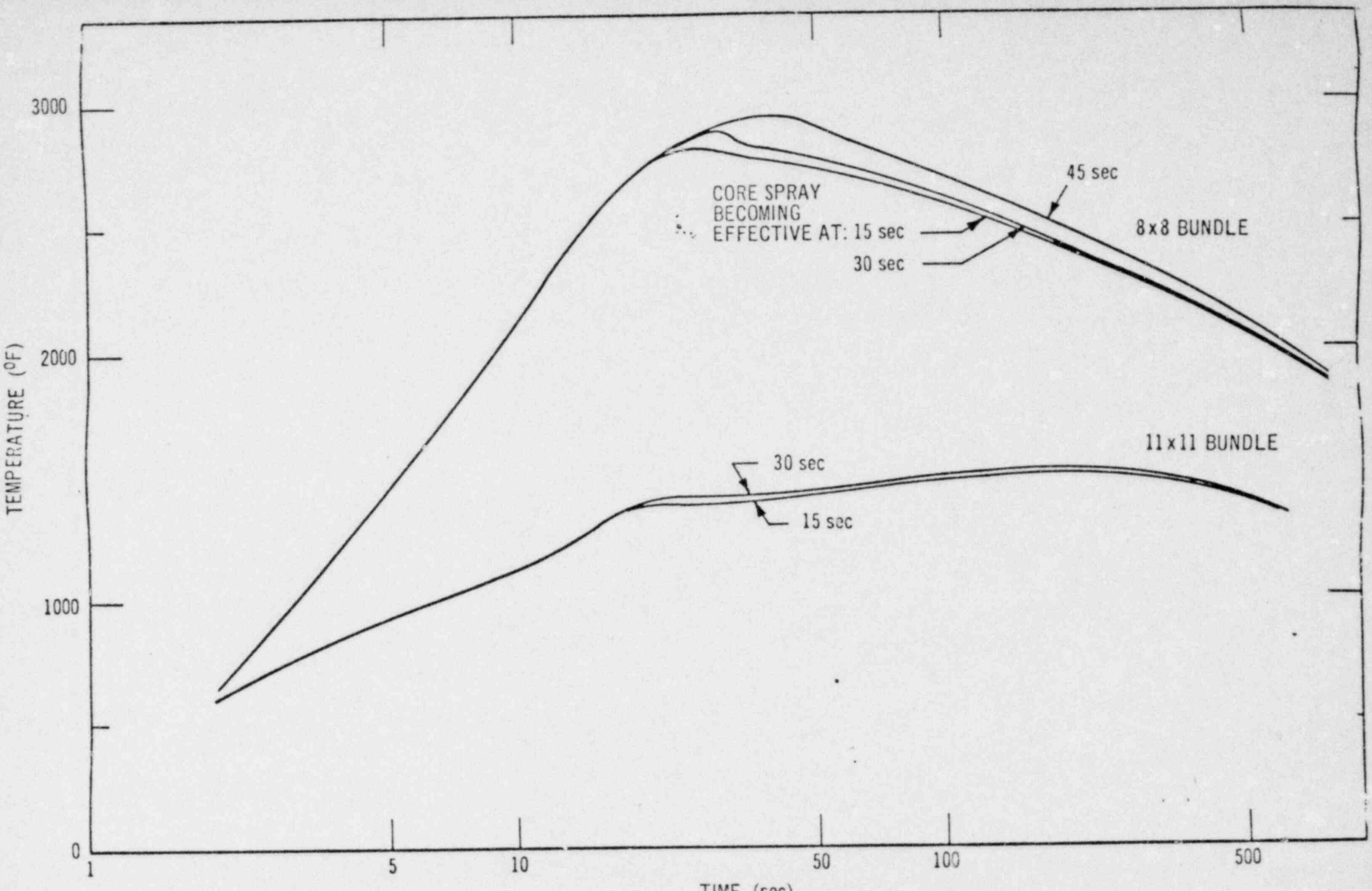

FIGURE 4. MAXIMUM CLADDING TEMPERATURE FOR DIFFERENT CORE SPRAY EFFECTIVENESS TIMES (L-O-C ACCIDENT)

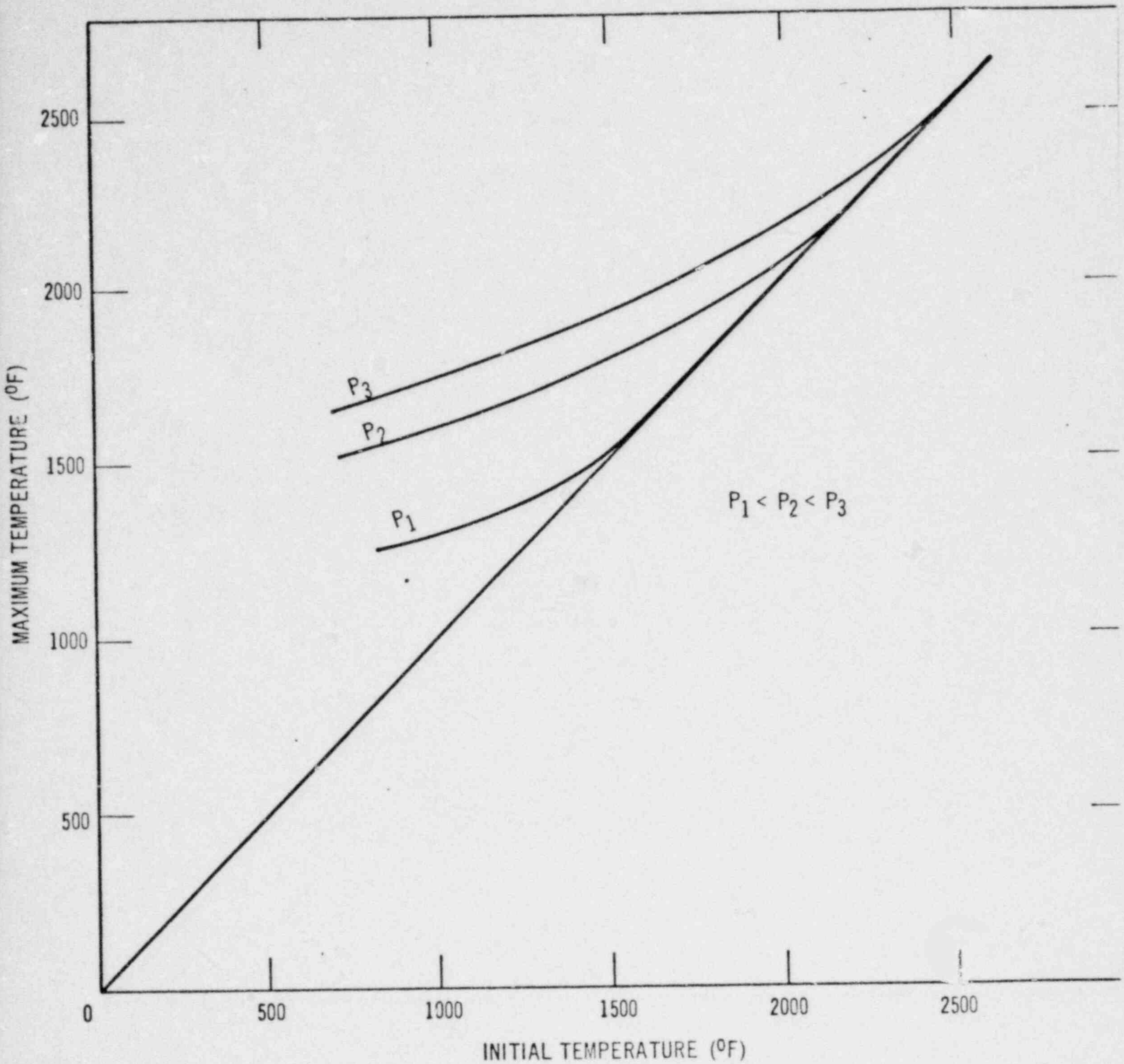

FIGURE 5. SPRAY COOLING
EFFECT OF INITIAL TEMPERATURE

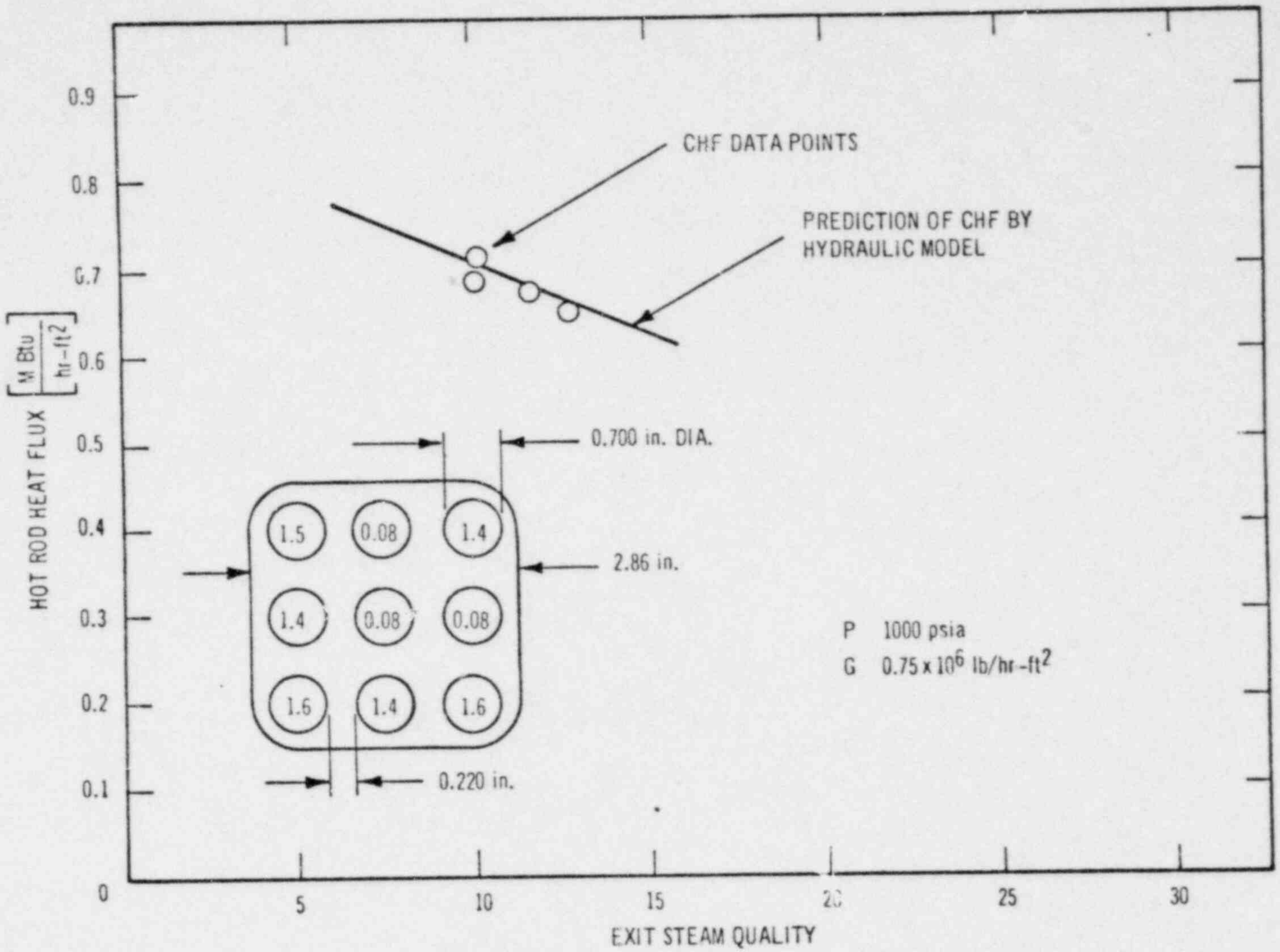

FIGURE 6. 9-ROD CHF TEST – LARGE RODS AND HIGH ROD-ROD PEAKING

Consumers Power Co.
Jackson, Michigan
(Robert L. Haueter)

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Dr. Morris

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Ltr. trans. the following in response to our 7-21-67 ltr. relative to their Proposed Change No. 13 dated 5-26-67:

ENCLOSURES: (3 signed & 37 conf'd cys.)

Answers to Additional Info Requested Regarding Proposed Change No. 13, dated and notarized 8-15-67

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