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June 7, 1985

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Dear Mr. Varga:

By letter dated February 28, 1985, we transmitted the document entitled "Amendment No. 2 to Application for Amendment to Operating License". That application requested a change to the previously proposed Indian Point Unit No. 2 Technical Specifications that would, among other things, revise the number of reactor coolant pumps required to be operating when the reactor coolant system is at hot shutdown and above 350°F. This change was proposed to achieve consistency between the safety analysis for the Uncontrolled Rod Withdrawal From Subcritical transient and the proposed Technical Specifications concerning the number of reactor coolant pumps in operation.

Subsequently, in a telephone conference on May 30, 1985, Ms. Slosson of your staff requested that we submit documentation in support of our February 28, 1985 submittal requiring two reactor coolant pumps to be operating when at hot shutdown and above 350°F. A summary of the supporting analysis is attached to this letter. This attached summary has been prepared for future inclusion into the updated FSAR; accordingly, all references contained therein are to the Indian Point Unit No. 2 FSAR.

Should you or your staff have any additional questions, please contact us.

Very truly yours,

John D. O'Toole

attach.

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14.1.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

14.1.1.1 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A Rod Cluster Control Assembly (RCCA) bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA's resulting in a power excursion. Such a transient could be caused by operator action or by a malfunction of the reactor control or rod control systems. This could occur with the reactor either subcritical, at hot zero power or at power. The "at power" case is discussed in Section 14.1.2. The low power startup condition assumed in this study (1×10^{-9} of nominal power) is less than the power level expected for any shutdown condition.

Although the reactor is normally brought to power from a subcritical condition by means of RCCA withdrawal, initial startup procedures with a clean core call for boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis (see Section 14.1.5, "Chemical and Volume Control System Malfunction").

The RCCA drive mechanisms are wired into preselected bank configurations which are not altered during reactor life. These circuits prevent the RCCA's from being automatically withdrawn in other than their respective banks. Power supplied to the banks is controlled such that no more than two banks can be withdrawn at the same time and in their proper withdrawal sequence. The RCCA drive mechanisms are of the magnetic latch type and coil actuation is sequenced to provide variable speed travel. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed, which is well within the capability of the protection system to prevent core damage.

The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the power excursion is of primary importance since it limits the power to a tolerable level during the delay time for protective action. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the following automatic features of the reactor protection system:

a. Source Range High Neutron Flux Reactor Trip - actuated when either of two independent source range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after an intermediate range flux channel indicates a flux level above a specified level. It is automatically reinstated when both intermediate range channels indicate a flux level below a specified level.

b. Intermediate Range High Neutron Flux Reactor Trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected manually adjustable value. This trip function is manually bypassed only after two of the four power range channels are reading above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

c. Power Range High Neutron Flux Reactor Trip (Low Setting) - actuated when two out of the four power range channels indicate a power level above approximately 25% of full power. This trip function may be manually blocked when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated only after three of the four channels indicate a power level below this value.

d. Power Range High Neutron Flux Reactor Trip (High Setting) - actuates when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

In addition, control rod stops on high intermediate range flux level (one out of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent the need to actuate the intermediate range flux level trip and the power range flux level trip, respectively.

14.1.1.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

METHOD OF ANALYSIS

The analysis of the Uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first an average core nuclear power transient calculation, then an average core heat transfer calculation, and finally the DNBR calculation. The average core nuclear calculation is performed using spatial neutron kinetics methods in TWINKLE to determine the average power generation with time, including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. The average heat flux and temperature transients are determined by performing a fuel rod transient heat transfer calculation in FACTRAN. The average heat flux with appropriate peaking factors is next used in THINC for transient DNBR calculations.

This accident is analyzed using Standard Thermal Design Procedures. Plant characteristics and initial conditions are discussed in Section 14.0.2.1. In order to give conservative results for a startup accident, the following assumptions are made:

a. Since the magnitude of the power peak reached during the initial part of the transient for any given rate of reactivity insertion is strongly dependent on the Doppler coefficient, conservatively low values as a function of power are used.

b. Contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time between the fuel and the moderator is much longer than the neutron flux response time. However, after the initial neutron flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A highly conservative value is used in the analysis to yield the maximum peak heat flux.

c. The reactor is assumed to be just critical at hot zero power (no load) T_{avg} (547°F). This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-water heat transfer coefficient, larger specific heats, and a less negative (smaller absolute magnitude) Doppler coefficient, all of which tend to reduce the Doppler feedback effect thereby increasing the neutron flux peak. The initial effective multiplication factor is assumed to be 1.0 since this results in the worst nuclear power transient.

d. Reactor trip is assumed to be initiated by power range high neutron flux (low setting). The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod cluster control assembly release, is taken into account. A 10% increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25% to 35%. Since the rise in the neutron flux is so rapid, the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition, the reactor trip insertion characteristic is based on the assumption that the highest worth rod cluster control assembly is stuck in its fully withdrawn position.

e. The maximum positive reactivity insertion rate assumed (75 pcm/sec) is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed (45 inches/minute). Control rod drive mechanism design is discussed in Section 3.2.3.4.

f. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high worth position, is assumed in the DNB analysis.

g. The initial power level was assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power). This combination of highest reactivity insertion rate and lowest initial power produces the highest peak heat flux.

h. Two reactor coolant pumps are assumed to be in operation. This is conservative with respect to DNB.

No single active failure in any systems or equipment available to mitigate the effects of the accident will adversely affect the consequences of the accident.

RESULTS

Figures 14.1.1 through 14.1.3 show the transient behavior for the uncontrolled RCCA bank withdrawal incident, with the accident terminated by reactor trip at 35% of nominal power. The reactivity insertion rate used is greater than that calculated for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. Figure 14.1.1 shows the neutron flux transient.

The energy release and the fuel temperature increases are relatively small. The thermal flux response, of interest for DNB considerations, is shown on Figure 14.1.2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux much less than the full power nominal value. There is a large margin to DNB during the transient since the rod surface heat flux remains below the design value, and there is a high degree of subcooling at all times in the core. Figure 14.1.3 shows the response of the hot spot fuel average temperature and the hot spot clad temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR at all times remains above the limit value.

The calculated sequence of events for this accident is shown on Table 14.1.14. With the reactor tripped, the plant returns to a stable condition. The plant may subsequently be cooled down further by following normal plant shutdown procedures.

The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain steam generator level through control of the main or auxiliary feedwater system.

Any action required of the operator to maintain the plant in a stabilized condition will be in a time frame in excess of ten minutes following reactor trip.

14.1.1.3 RADIOLOGICAL CONSEQUENCES

There are no radiological consequences associated with an uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power start-up condition event since radioactivity is contained within the fuel rods and the reactor coolant system is maintained within design limits. This is demonstrated by showing that the minimum DNBR remains above 1.30.

14.1.1.4 CONCLUSIONS

In the event of a RCCA withdrawal accident from the subcritical condition, the core and the reactor coolant system are not adversely affected, since the combination of thermal power and the coolant temperature result in a DNBR greater than the limit value. Thus, no fuel or clad damage is predicted as a result of DNB.

TABLE 14.1.14

TIME SEQUENCE OF EVENTS

Accident	Event	Time (sec.)
Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	Initiation of uncontrolled rod withdrawal from 10^{-9} of nominal power	0.0
	1 Power range high neutron flux low setpoint reached	10.3
	2 Peak nuclear power (121% of nominal) occurs	10.4
	3 Rod begins to fall into core	10.8
	4 Peak average clad temperature (658°F) occurs	12.4
	5 Peak heat flux (35.1% of nominal) occurs	12.1
	6 Minimum DNBR occurs	12.0
	7 Peak average fuel temperature (1655°F) occurs	12.6

FIGURE 14.1.1

NEUTRON FLUX TRANSIENT FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

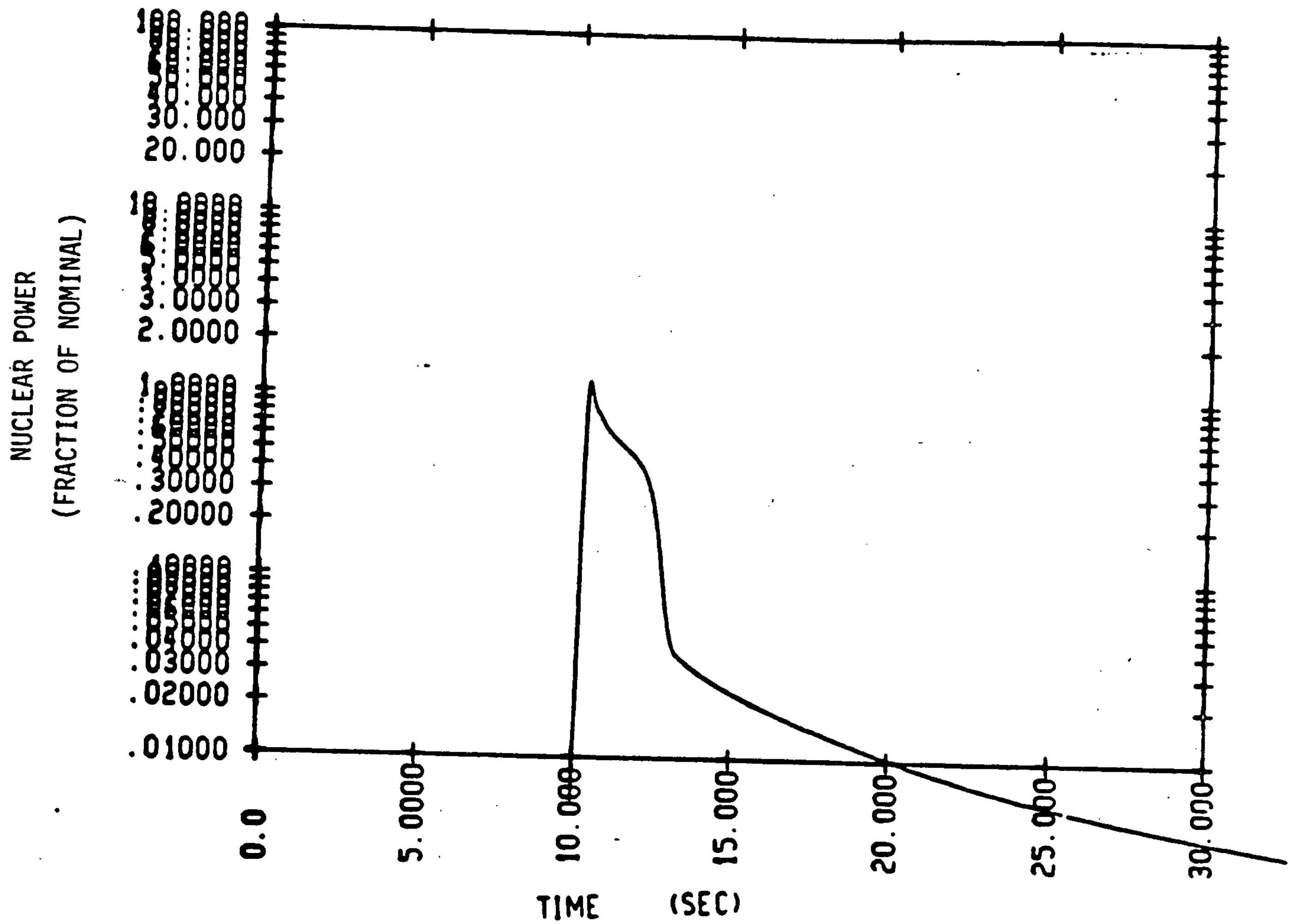

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FIGURE 14.1.2

THERMAL FLUX TRANSIENT FOR UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION

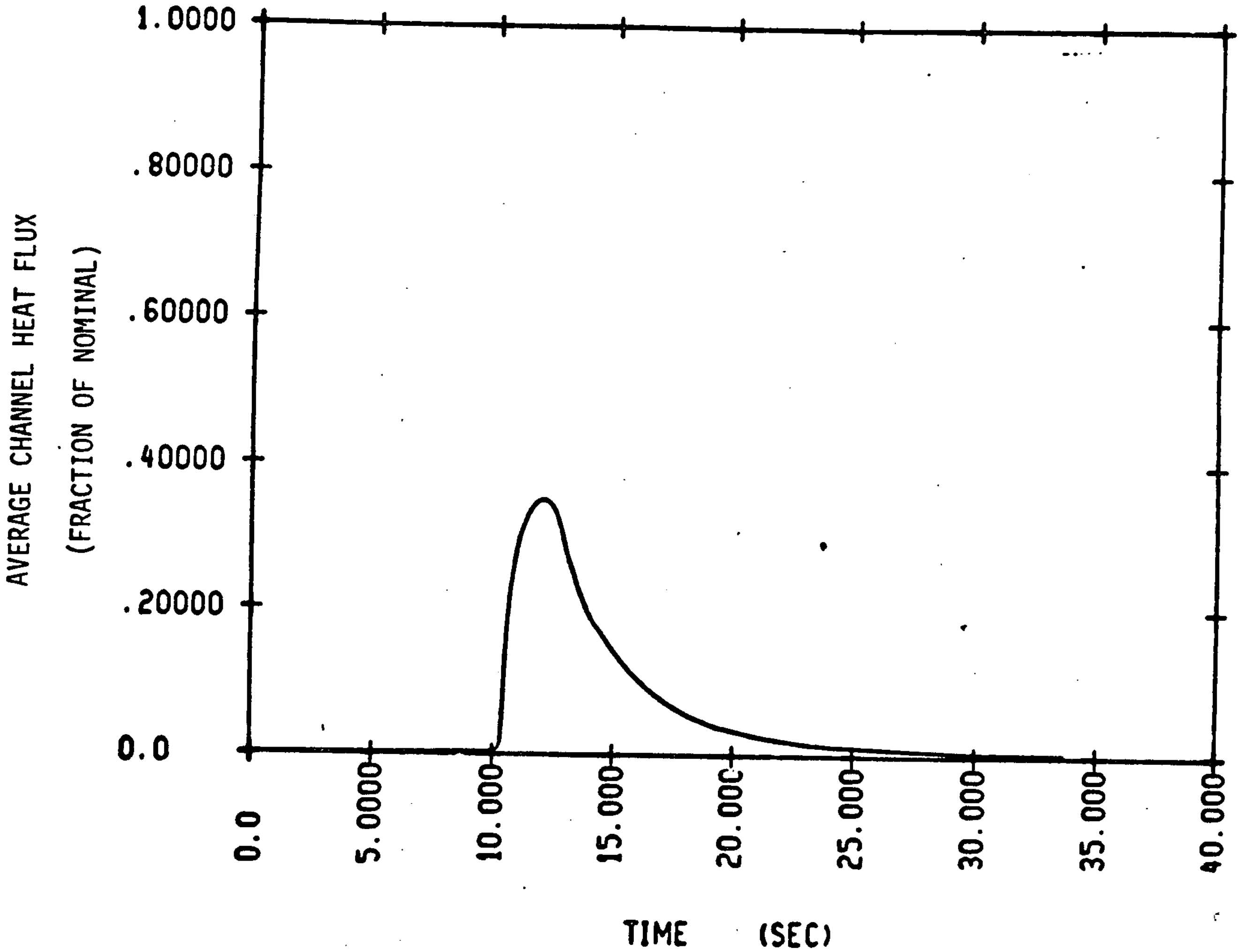

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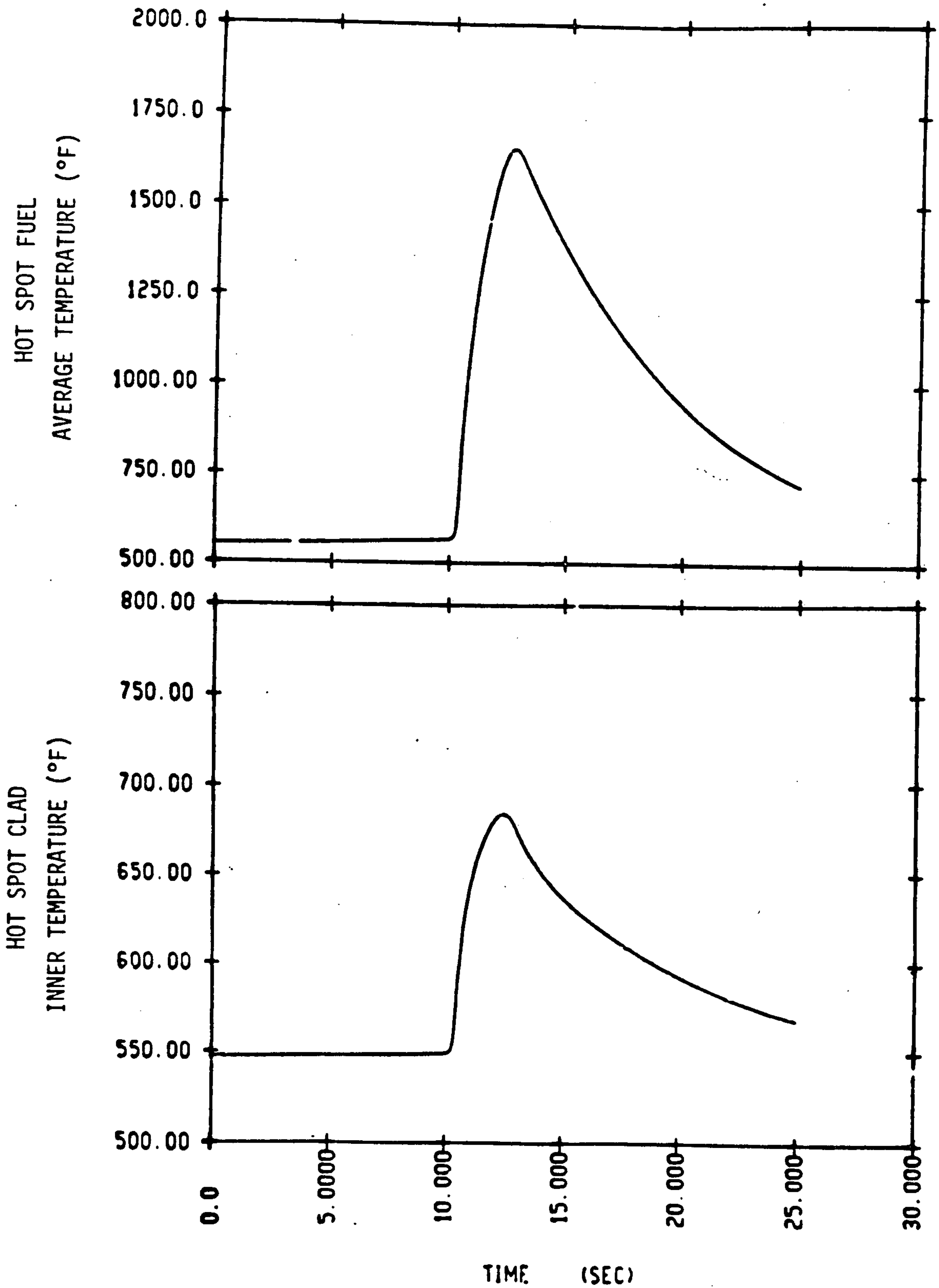
FIGURE 14.1.3

FUEL AND CLAD TEMPERATURE TRANSIENTS FOR
UNCONTROLLED ROD WITHDRAWAL FROM A
SUBCRITICAL CONDITION



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