WILLIAM L. STEWART
Vice President
Nuclear Operations

Nuclear Operations Department
Post Office Box 26666
One James River Plaza
Richmond, Virginia 23261

VIRGINIA POWER

April 1, 1985

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Attn: Mr. H. L. Thompson, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Serial No. 457
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NPF-4
NPF-7

Gentlemen:

VIRGINIA POWER
SAFETY EVALUATION OF ROD WITHDRAWAL
FROM SUBCRITICAL AT LOW FLOW CONDITIONS

Westinghouse notified Virginia Power on June 1, 1984, of a potential unreviewed safety question concerning the Uncontrolled Rod Withdrawal from Subcritical accident. In the current UFSAR analysis of this event, Westinghouse assumed operation of all reactor coolant pumps (N pump operation) in determining whether the results of the Uncontrolled Rod Withdrawal from Subcritical accident meet the Condition II (DNBR) acceptance criteria. This analysis showed the DNB criterion was met since the peak heat flux during the accident was less than the hot full power value. As expected, since this was a full flow (N pumps) case, the minimum DNBR was less limiting than that calculated for steady state hot full power conditions.

Surry and North Anna Technical Specifications allow operation with one or two pumps when the plant is in a subcritical condition. Thus, a potential inconsistency for these plants exists between the Technical Specifications and the FSAR accident analyses.

For your information, we are forwarding the attached safety evaluation performed for Surry to study the impact of low flow (one of three pumps running) for the Uncontrolled Rod Withdrawal from Subcritical Condition. Virginia Power performed this analysis using the RETRAN computer code, and the reactor system transient analysis methodology described in our topical report which was transmitted by letter from Mr. W. N. Thomas (Vepco) to Mr. H. R. Denton dated April 14, 1981 (Serial No. 215). The analysis results indicate that the Condition II criterion (minimum DNBR is greater than the limiting value of 1.30) continues to be met for the low flow conditions. No unreviewed safety question as defined in 10 CFR 50.59 exists. These results have been reviewed by both the Surry Station Nuclear Safety and Operation Committee and the Safety Evaluation and Control staff. The Surry UFSAR will be updated to include the results of this analysis.

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A similar analysis has been performed for Virginia Power's North Anna Power Station. This analysis was documented as part of Virginia Power's submittal to the NRC for allowing a positive moderator temperature coefficient (Virginia Power letter No. 666 dated February 7, 1985). This analysis also shows that the Condition II criterion is met (minimum DNBR is greater than the limiting value of 1.30).

If you require any additional information, please contact this office.

Very truly yours,

W. L. Stewart

W. L. Stewart

Attachment

1. Safety Evaluation of Surry Rod Withdrawal from Subcritical at Low Flow Conditions

cc: Dr. J. Nelson Grace
Regional Administrator
Region II

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

Mr. D. J. Burke
NRC Resident Inspector
Surry Power Station

Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing

Mr. M. W. Branch
NRC Resident Inspector
North Anna Power Station

ATTACHMENT 1

SAFETY EVALUATION OF SURRY ROD WITHDRAWAL
FROM SUBCRITICAL AT LOW FLOW CONDITIONS

UNCONTROLLED CONTROL-ROD ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL CONDITION

A control-rod assembly withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by the withdrawal of control-rod assemblies, resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical or at power. The "at power" case is discussed in UFSAR Section 14.2.2.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low-power level during startup by control-rod withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with control-rod assembly withdrawal. Control-rod assembly motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The control-rod drive mechanisms are wired into preselected banks, and these bank configurations are not altered during core life. The assemblies are therefore physically prevented from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The control-rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable-speed rod travel.

Should a continuous control-rod assembly withdrawal be initiated, the transient will be terminated by the following automatic safety features:

1. Source range flux level trip - actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either intermediate-range flux channel indicates a flux level above the source range cutoff power level. It is automatically reinstated when both intermediate-range channels indicate a flux level below the source range cutoff power level.

2. Intermediate-range control-rod stop - actuated when either of two independent intermediate-range channels indicates a flux level above a preselected, manually adjustable value. This control-rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately 10% of full power. It is automatically reinstated when three of the four power range channels are below this value.

3. Intermediate-range flux level trip - actuated when either of two independent intermediate-range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when 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.

4. Power range flux level 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 bypassed when two of the four power range channels indicate a power level above approximately 10% of full power and is automatically reinstated when three of the four channels indicate a power level below this value.

5. Power range control-rod stop - actuated when one out of the four power channels indicates a power level above a preset setpoint. This function is always active.

6. Power range flux level trip (high setting) - actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

The nuclear power response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup incident since it limits the power to a tolerable level before external control action. After the initial power burst, the nuclear power is momentarily reduced and then if the incident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

The termination of the startup incident by the above protection channels prevents core damage. In addition, the reactor trip from high reactor pressure serves as a backup to terminate the incident before an overpressure condition could occur.

Method of Analysis

A detailed analysis using the RETRAN computer code and the reactor system transient analysis methodology described in our topical report, which was transmitted by letter from Mr. W. N. Thomas (Vepco) to Mr. H. R. Denton dated April 14, 1981 (Serial No. 215), has been performed in order to obtain the plant transient behaviour following an uncontrolled rod withdrawal from a subcritical condition. The analysis includes the simulation of the plant neutron kinetics, and the core thermal and hydraulic feedback equations. The RETRAN code calculates nuclear power, core heat flux, fuel, clad and coolant temperatures. The detailed core thermal-hydraulics analysis was performed using the COBRA computer code ("VEPCO Reactor Core Thermal-Hydraulic Analysis Using the COBRA-IIIC/MIT Computer Code" Topical Report VEP-FRD-33-A dated October, 1983) to generate MDNBRs (Minimum Departure from Nucleate Boiling Ratio).

In order to give conservative results from a startup incident, the following additional assumptions are made concerning the initial reactor conditions:

1. Since the magnitude of the nuclear power peak reached during

the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the Doppler power reactivity coefficient, a conservative fuel-temperature-dependent Doppler coefficient was used.

2. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux response constant. However, after the initial nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. A conservative value of +3 pcm/°F was used in the analyses since the positive value yields the maximum peak core flux (1 pcm = 10^{-5} dk/k).

3. The reactor is assumed to be at hot zero power. This assumption is more conservative than that of a lower initial system temperature. The higher initial system temperature yields a larger fuel-to-water thermal conductivity, a larger fuel thermal capacity, and a less negative (smaller absolute magnitude) Doppler coefficient. The less negative Doppler coefficient reduces the Doppler feedback effect, thereby increasing the nuclear flux peak. The high nuclear flux peak combined with a high fuel thermal capacity and large thermal conductivity yields a larger peak heat flux. Initial multiplication (ko) is assumed to be 1.0 since this results in the maximum nuclear power peak.

4. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and control-rod assembly release, are taken into account. A 10% increase has been assumed for the power range flux trip setpoint, raising it from the nominal value of 25% to 35%. Reference to Figure 1, however, shows that the rise in nuclear flux is so rapid that the effect of errors in the trip setpoint on the actual time at which the rods are released is negligible. In addition to the above, the rate of reactivity insertion corresponding to the trip action is based on the assumption that the highest worth control-rod assembly is stuck in its fully withdrawn position.

5. The maximum positive reactivity insertion rate assumed (75 pcm/sec) is greater than that for the simultaneous withdrawal of the combination of the two control banks having the greatest combined worth at maximum speed (45 in/min.).

6. The initial power level was assumed to be below the power level expected for any shutdown condition. The combination of highest reactivity insertion rate and lowest initial power produces the the highest peak heat flux.

7. This analysis is performed assuming only one pump operation.

Results

Figures 1, 2, and 3 show the transient behavior for a reactivity insertion rate of 75 pcm/sec with the incident terminated by reactor trip at 35% power. This insertion rate is greater than that for the two highest worth banks, both assumed to be in their highest incremental worth region. Figure 1 shows the nuclear power increase. The nuclear power overshoots to approximately 386% but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are small. The thermal flux response, of interest for departure-from-nucleate-boiling (DNB) considerations, is shown on Figure 2. The beneficial effect of the inherent thermal lag of the fuel is evidenced by a peak heat flux of only 38% of the 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 3 shows the response of the average fuel, cladding, and coolant temperature. The average fuel temperature increases to 763°F, which is lower than the nominal full-power value of about 1475°F. The average coolant temperature increases to only 585°F.

Conclusion

Taking into account the conservative assumptions used in the 75-pcm/sec reanalysis, it is concluded that, in the unlikely event of a control-rod assembly withdrawal incident, the core and reactor coolant system are not adversely affected, since the thermal power reached is only approximately 38% of the nominal value and the core water temperature increases to only 585°F. This combination of thermal power and core water temperature results in a DNBR well above the limiting value of 1.30. The peak average clad temperature is less than the nominal full-power value of 633°F and thus there is no clad damage.

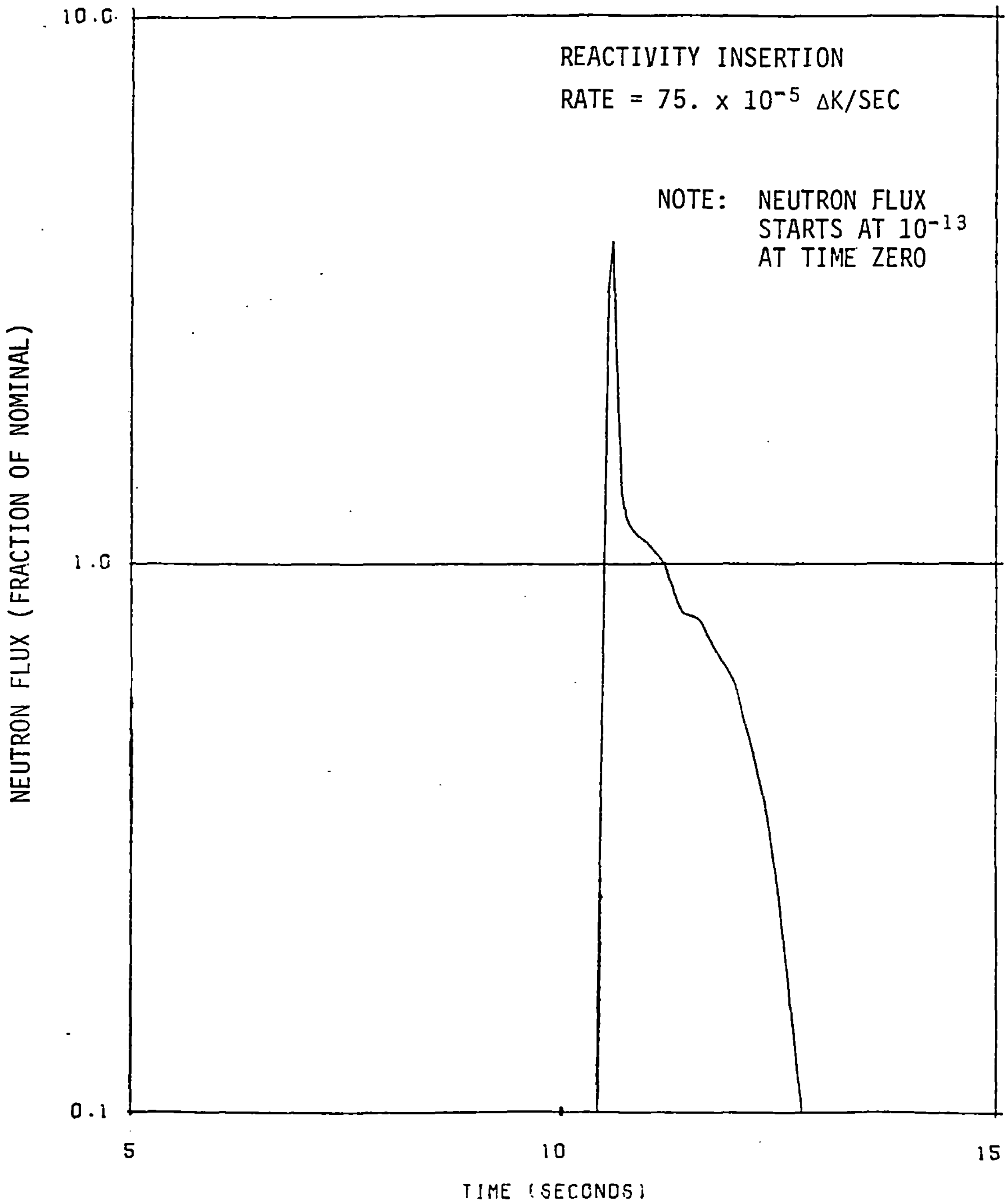

Figure 1 Uncontrolled Rod Withdrawal from a Subcritical Condition, Neutron Flux versus Time

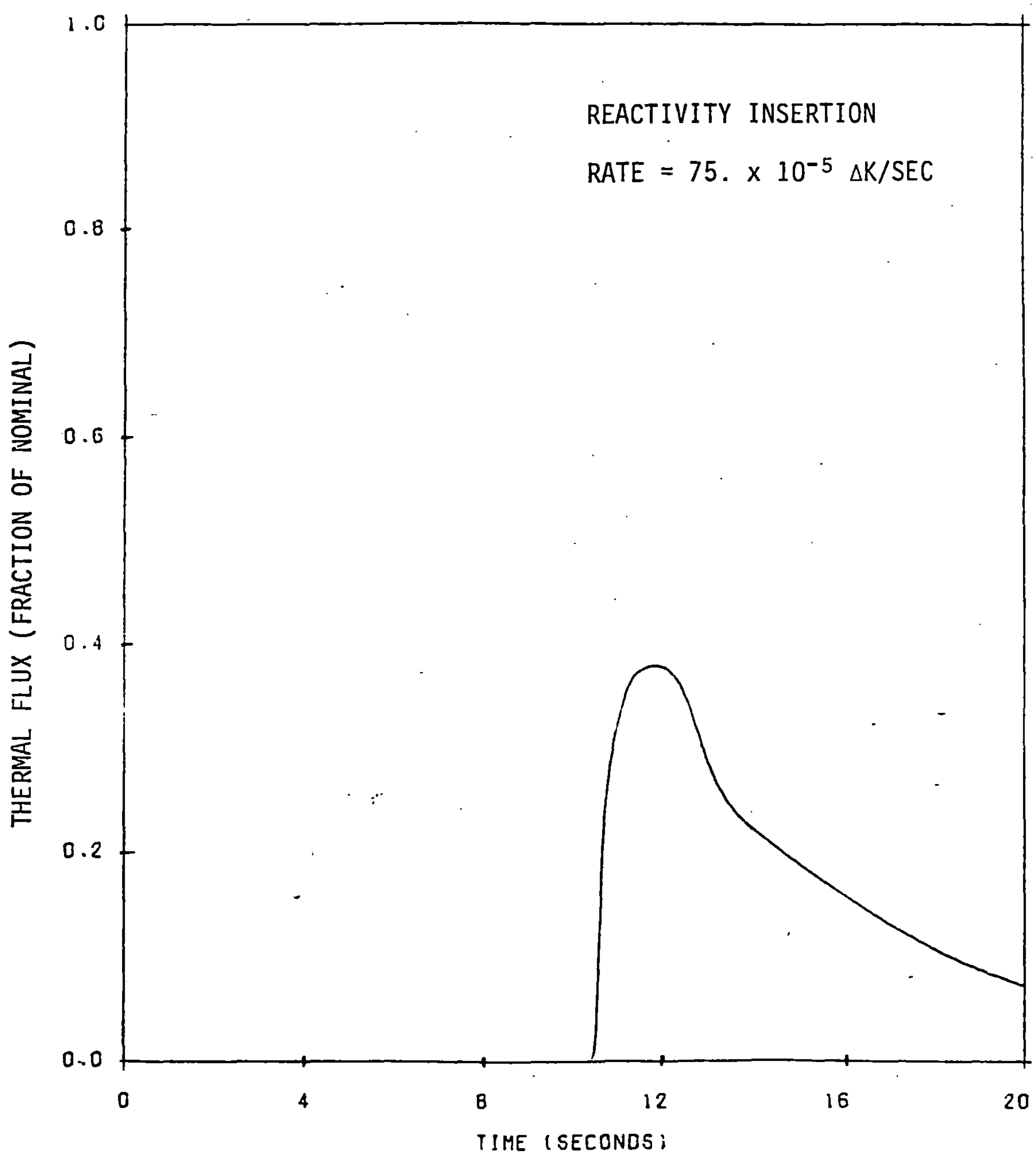

Figure 2 Uncontrolled Rod Withdrawal from a Subcritical Condition, Thermal Flux versus Time

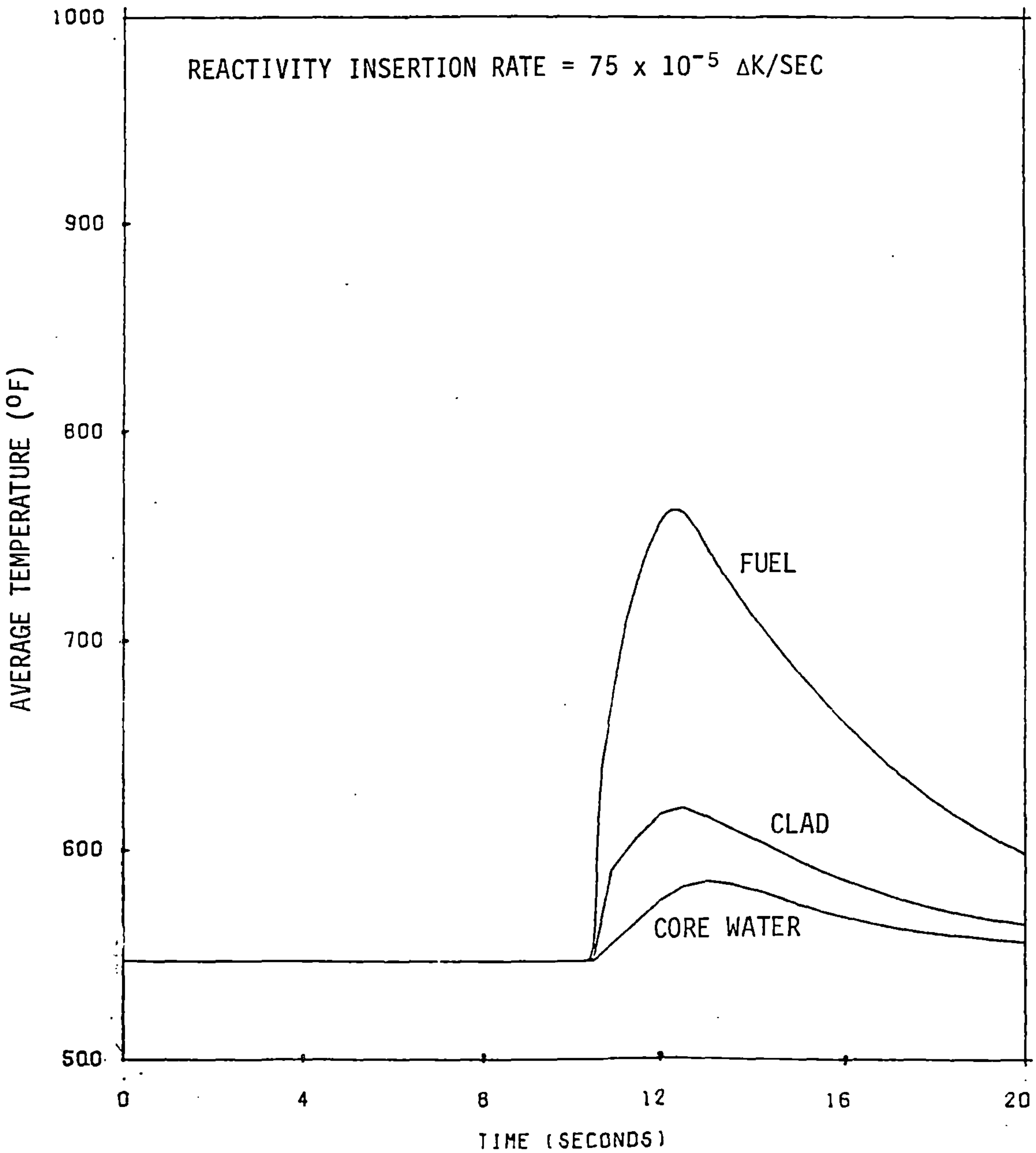

Figure 3 Uncontrolled Rod Withdrawal from a Subcritical Condition, Temperature versus Time