TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
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May 14, 1986

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Project Director
PWR Project Directorate No. 4
Division of Pressurized Water Reactor (PWR)
Licensing A
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

| | | |
|---|---|---|
| In the Matter of the Application of | ) | Docket Nos. 50-390 |
| Tennessee Valley Authority | ) | 50-391 |

Please refer to J. W. Hufham's letter to E. Adensam dated June 21, 1985 concerning Watts Bar technical specification certification. In that letter, TVA provided a revised Final Safety Analysis Report (FSAR) analysis to support technical specification 3.4.1.2. The revised analysis for uncontrolled rod withdrawal from subcriticality justified acceptable departure from nucleate boiling (DNB) parameters with two reactor coolant pumps (RCP) in operation in mode 3. Our intention at that time was to include the revised analysis in the next FSAR amendment.

In August 1985, TVA was informally notified by Westinghouse of a discrepancy in the analysis they had performed. You were notified of this in J. A. Domer's letter dated January 10, 1986. Westinghouse had used the WBR-1 critical heat flux correlation in their analysis. The WBR-1 correlation has been approved by NRC for use in other plants; however, the W-3 R-grid correlation was used to form the licensing basis for Watts Bar Nuclear Plant. Westinghouse has completed the reanalysis of the uncontrolled rod withdrawal from subcriticality accident using the W-3 R-grid correlation. The reanalysis has confirmed the DNB design basis for two RCPs in operation. Enclosed are FSAR changes associated with the revised analysis. The changes will be included in a future amendment.

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Director of Nuclear Reactor Regulation May 14, 1986

If there are any questions, please get in touch with K. P. Parr at FTS 858-2681.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

R. Gridley, Director
Nuclear Safety and Licensing

Enclosure
cc: U.S. Nuclear Regulatory Commission (Enclosure)
Region II
Attention: Dr. J. Nelson Grace, Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

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15.1.9.6 TURTLE

TURTLE is a two-group, two-dimensional neutron diffusion code featuring a direct treatment of the nonlinear effects of xenon, enthalpy, and Doppler. Fuel depletion is allowed.

TURTLE was written for the study of azimuthal xenon oscillations, but the code is useful for general analysis. The input is simple, fuel management is handled directly, and a boron criticality search is allowed.

TURTLE is further described in Reference 17 .

15.1.9.7 TWINKLE

The TWINKLE program is a multi-dimensional spatial neutron kinetics code, which was patterned after steady state codes presently used for reactor core design. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two and three dimensions. The code uses six delayed neutron groups and contains a detailed multi-region fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. The code handles up to 2000 spatial points, and performs its own steady state initialization. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided, e.g. channelwise power, axial offset, enthalpy, volumetric surge, pointwise power, and fuel temperatures.

The TWINKLE Code is used to predict the kinetic behavior of a reactor for transients which cause a major perturbation in the spatial neutron flux distribution.

TWINKLE is further described in Reference 18 .

~~15.1.9.8 WIT~~

Delete

~~WIT is a one-region neutron kinetics program with a single axial lump description of thermal kinetics making it useful in the analysis of transients in a heterogeneous reactor core consisting of fuel rods, fuel rod clad, and water moderator and coolant. The code is basically a core model and therefore generally useful for fast reactivity transients which terminate before there is significant feedback from the remainder of the plant, i.e. transients shorter than the loop transit time.~~ 53

~~WIT is used in safety analysis of reactivity accidents from a subcritical condition.~~

~~WIT is further described in Reference 19 .~~



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15.1.9.9 PHOENIX

The PHOENIX code calculates the individual loop flows, core flow and pump speeds as a function of time subsequent to failure of any number of the reactor coolant pumps. The analysis is based on a momentum balance around each reactor coolant loop and across the reactor core. This momentum balance is combined with the continuity equation, a pump momentum balance and the pump characteristics. Any number of reactor coolant loops are accommodated up to a maximum of 6.

PHOENIX is further described in Reference [20].

15.1.9.10 THINC

The THINC Code is described in Section 4.4.3.4.

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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:

1. 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.

2. Intermediate Range High Neutron Flux Reactor Trip - actuated when either of two independent intermediate range channels indicates a neutron flux level above a preselected manually adjustable setpoint. This trip function may be manually bypassed only after two of the four power range channels are reading above approximately 10 percent of full power and is automatically reinstated when three of the four channels indicate a power level below this value. | 53

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

4. Power Range High Neutron Flux Reactor 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.

INSERT 15A →

In addition, control rod stops on high intermediate range flux level (one of two) and high power range flux level (one out of four) serve to discontinue rod withdrawal and prevent actuation of the intermediate range flux level trip and the power range flux level trip, respectively. | 53

15.2.1.2 Analysis of Effects and Consequences

Method of Analysis

~~This transient is analyzed by two digital computer codes. The WIT-6 3 Code is used to calculate the reactivity transient and~~

Delete, Insert 15B



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INSERT 15A

5. Power Range High Positive Neutron Flux Rate Trip - actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicate a rate above the preset setpoint. This trip function is always active.

INSERT 15B

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, a DNBR calculation. The average core nuclear power calculation is performed using spatial neutron kinetics methods, TWINKLE (3), 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 (4). The average heat flux is next used in THINC (described in section 4.4.3.4) for the transient DNBR calculation.

Delete ~~hence the nuclear power transient. This code includes the simulation of six delayed neutron groups and the core thermal and hydraulic feedback equations. The FACTRAN [4] Code is then used to calculate the thermal heat flux transient based on the nuclear power transient calculated by the WIT-6 Code. FACTRAN also calculates the fuel and clad temperatures.~~

In order to give conservative results for a startup accident, the following assumptions are made concerning the initial reactor conditions:

1. 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, conservative values (low absolute magnitude) as a function of ~~temperature~~ power are used. See Section 15.1.6 and Table 15.1-2.

2. Contribution of the moderator reactivity coefficient is negligible during the intial 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 conservative value ~~of +1 pcm/°F,~~ delete which is appropriate for beginning of core life at hot zero power, is used in the analysis to yield the maximum peak heat flux. ~~This value is conservative since the moderator coefficient in a rodded core is expected to be negative. See Section 15.1.6 and Table 15.1-2.~~ Delete

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-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.

4. 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 percent increase is assumed for the power range flux trip setpoint raising it from the nominal value of 25 percent to 35 percent. Previous results, however, show that rise in the neutron flux is so rapid that the effect of errors in the



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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 RCCA is stuck in its fully withdrawn position. See Section 15.1.5 for RCCA insertion characteristics. 53

5. The maximum positive reactivity insertion rate assumed 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 4.2.3. 53

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 highest peak heat flux.

INSERT 15C →

Results

The calculated sequence of events for this accident is shown on Table 15.2-1.

Figures 15.2-1 through 15.2-3 show the transient behavior for the indicated reactivity insertion rate with the accident terminated by reactor trip at 35 percent nominal power. This insertion rate is greater than that for the two highest worth sequential control banks, both assumed to be in their highest incremental worth region. ~~It is also greater (by more than a factor of 10) than the maximum insertion rate of the part length RCCA's~~. Delete

Figure 15.2-1 shows the nuclear power transient. The nuclear power overshoots the full power nominal value but this occurs for only a very short time period. Hence, the energy release and the fuel temperature increases are relatively small. The heat flux response, of interest for DNB considerations, is shown on Figure 15.2-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by a peak heat flux 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. Figures 15.2-3 and 15.2-3A show the response of the HOT SPOT average fuel, and cladding ~~and coolant~~ temperature. The average fuel temperature increases to a value lower than the nominal full power value. The minimum DNBR remains above the limiting value at all times.

15.2.1.3 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



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

8. Two reactor coolant pumps are assumed to be in operation.

is predicted

the coolant temperature result in a DNBR greater than the limiting value. ~~of 1.30.~~ Thus, ~~there will be~~ no cladding damage and no release of fission products to the Reactor Coolant System as a result of DNB. ~~This conclusion regarding the DNBR is based on the fact that the analysis shows that nominal full power values for the coolant average temperature, heat flux, and fuel pellet average temperature are not exceeded.~~ Delete

## 15.2.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

### 15.2.2.1 Identification of Causes and Accident Description

Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, in order to avert damage to the fuel clad the Reactor Protection System is designed to terminate any such transient before the DNBR falls below ~~1.30.~~ the limiting value.

The automatic features of the Reactor Protection System which prevent core damage following the postulated accident include the following:

1. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kw/ft) is not exceeded.

4. A high pressurizer pressure reactor trip actuated from any two out of four pressure channels which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip actuated from any two out of three level channels which is set at a fixed point.

Following reactor trip, the plant will approach a stabilized condition at hot standby; normal plant operating procedures may then be followed. The operating procedures would call for operator action to control RCS boron concentration and pressurizer level using the CVCS, and to maintain 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. 30

15.2.14.3 Conclusions

Results of the analysis show that spurious safety injection with or without immediate reactor trip presents no hazard to the integrity of the Reactor Coolant System.

DNBR is never less than the initial value.

If the reactor does not trip immediately, the low pressurizer pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown thereby expediting recovery from the incident. 53

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TABLE 15.1-2

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

| FAULTS | COMPUTER CODES UTILIZED | REACTIVITY COEFFICIENTS ASSUMED: MODERATOR TEMPERATURE (Δk/°F) | MODERATOR DENSITY (Δk/gm/cc) | DOPPLER | INITIAL NSSS THERMAL POWER OUTPUT ASSUMED* (MWt) |
|---|---|---|---|---|---|
| CONDITION II | | | | | |
| Uncontrolled RCC Assembly Bank Withdrawal from Subcritical Condition | Twinkle, ~~WIT,~~ FACTRAN, THINC | Refer to Section 15.2.1.2 (part 2) ~~+1 x 10^-5~~ | --- | Least negative Doppler power coefficient - Doppler defect = -.9 % ΔK ~~See Figure 15.1-5~~ | 0 |
| Uncontrolled RCC Assembly Bank Withdrawal at Power | LOFTRAN | --- | Figure 15.1-7 and 0.43 | lower and upper (1) | 3425 |
| RCC Assembly Misalignment | THINC, TURTLE LOFTRAN FRACTRAN | --- | Figure 15.1-7 | upper (1) | 3425 |
| Uncontrolled Boron Dilution | NA | NA | NA | NA | 0 and 3425 |
| Partial Loss of Forced Reactor Coolant Flow | PHOENIX, LOFTRAN THINC, FACTRAN | --- | Figure 15.1-7 | lower (1) | 2397 and 3425 |
| Startup of an Inactive Reactor Coolant Loop | MARVEL, THINC | --- | 0.43 | lower (1) | 2397 |
| Loss of External Electrical Load and/or Turbine Trip | LOFTRAN | --- | Figure 15.1-7 and 0.43 | upper (1) | 3425 |
| Loss of Normal Feedwater | BLKOUT | --- | NA | NA | 3579 |
| Loss of Off-Site Power to the Station Auxiliaries (Station Blackout) | BLKOUT | --- | NA | NA | 3425 |

(Sheet 1 of 4)

Revised by Amendment 53

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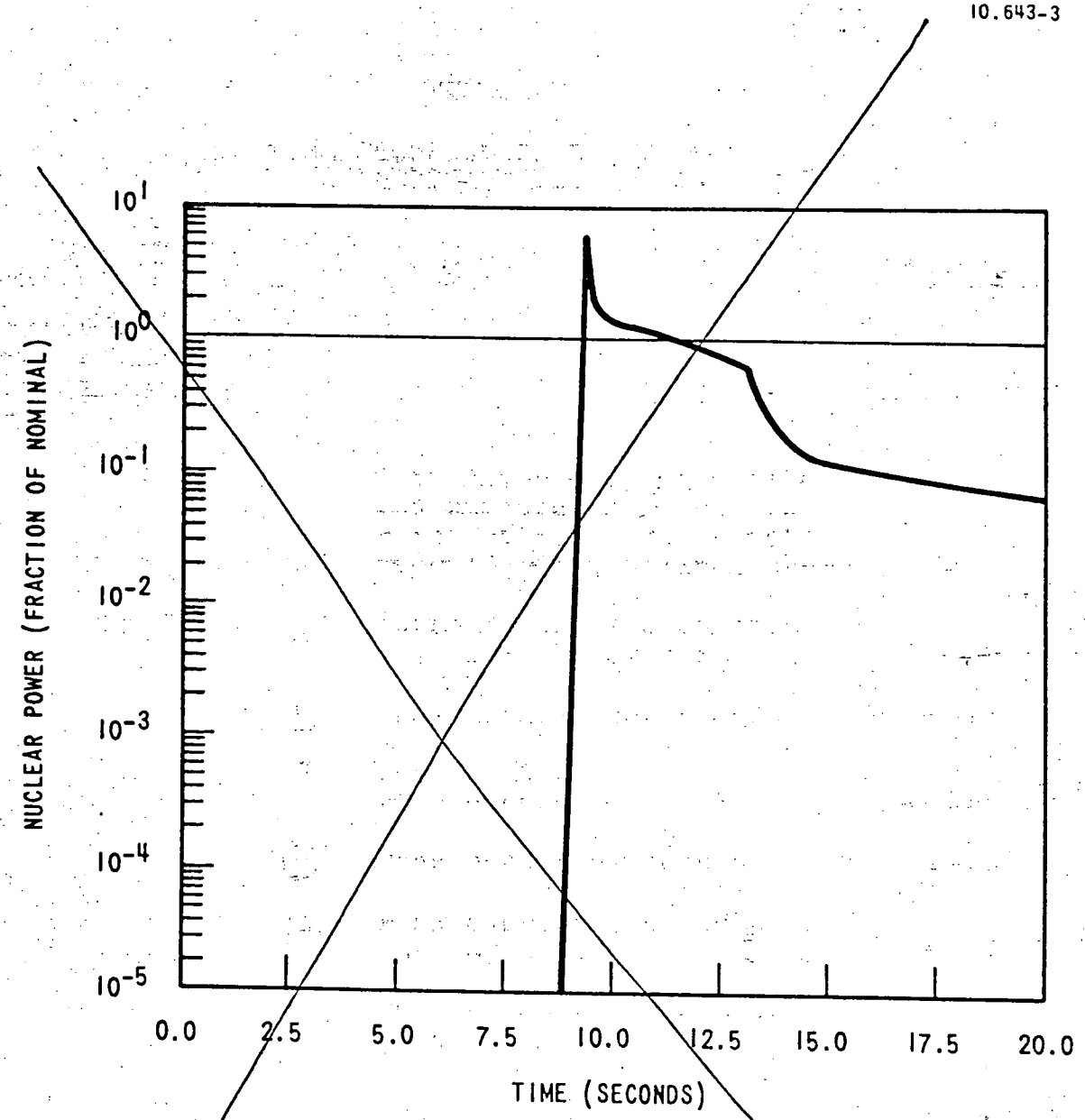
TABLE 15.2-1

TIME SEQUENCE OF EVENTS FOR CONDITION II EVENTS

| Accident | | Events | Time (sec.) |
|---|---|---|---|
| Uncontrolled RCCA Withdrawal from a Subcritical Condition | | Initiation of uncontrolled rod withdrawal ~~90~~ 60 pcm/sec reactivity insertion rate from $10^{-\cancel{13}\,9}$ of nominal power | 0 |
| | 1 | Power range high neutron flux low setpoint reached | ~~9.2~~ 12.74 |
| | 2 | Peak nuclear power occurs | ~~9.3~~ 12.90 |
| | 5 | Peak clad temperature occurs | ~~10.1~~ 15.28 |
| | 4 | Peak heat flux occurs | ~~10.1~~ 14.75 |
| | 6 | Peak average fuel temperature occurs | ~~11.2~~ 15.49 |
| | 3 | Rods begin to fall into core | ~~9.7~~ 13.24 |
| | 44 | Minimum DNBR occurs | 14.75 |

Arrange times in chronological order. SHOWN.




Delete, Insert New Figure 15.2-1

Figure 15.2-1 Uncontrolled Rod Withdrawal from a Subcritical Condition
Nuclear Power Versus Time

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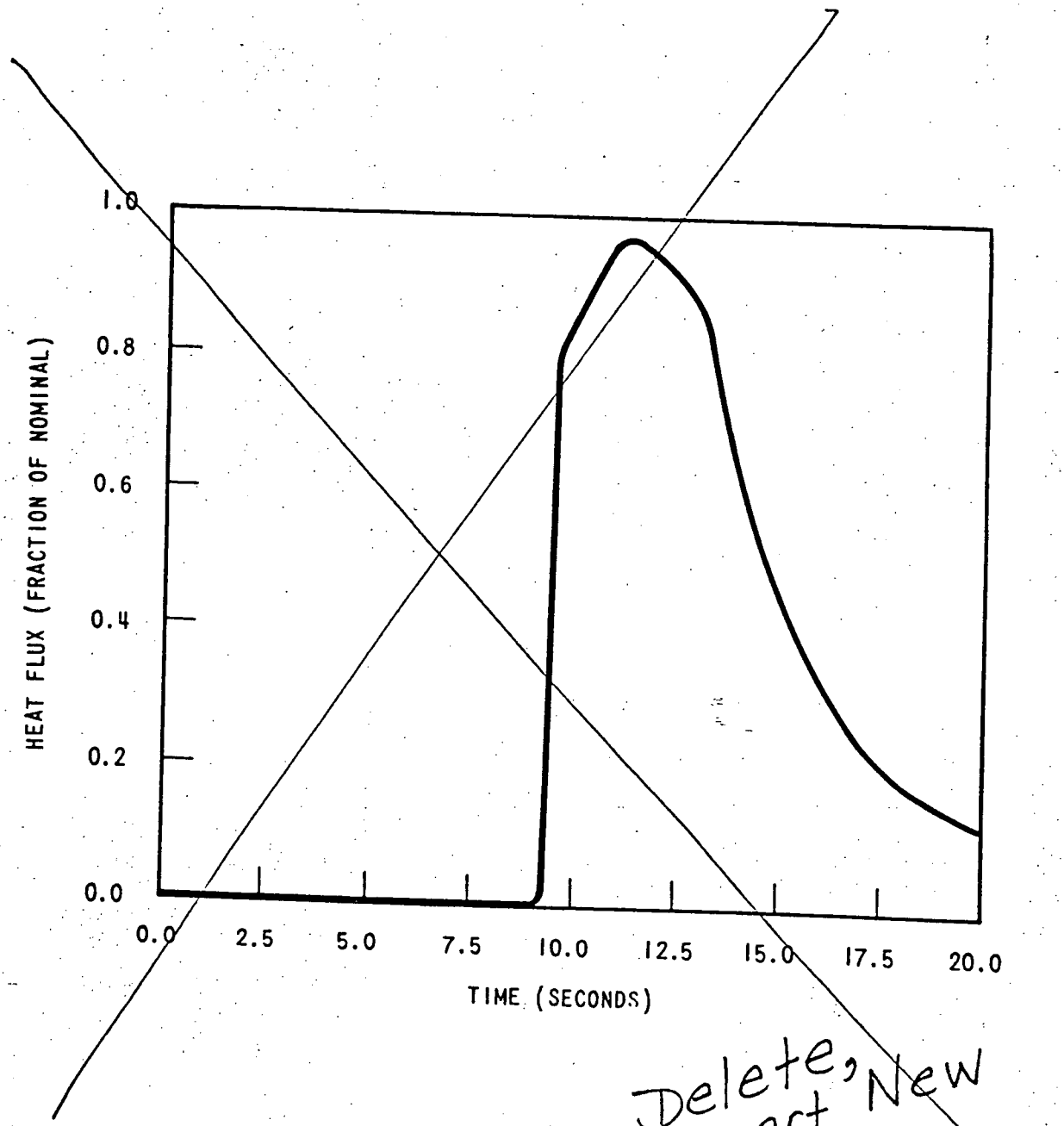


Delete, New Insert Figure 15.2-2

Figure 15.2-2 Uncontrolled Rod Withdrawal from a Subcritical Condition
Heat Flux Vs. Time

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Delete, Insert New Figures 15.2-3 & 15.2-3A

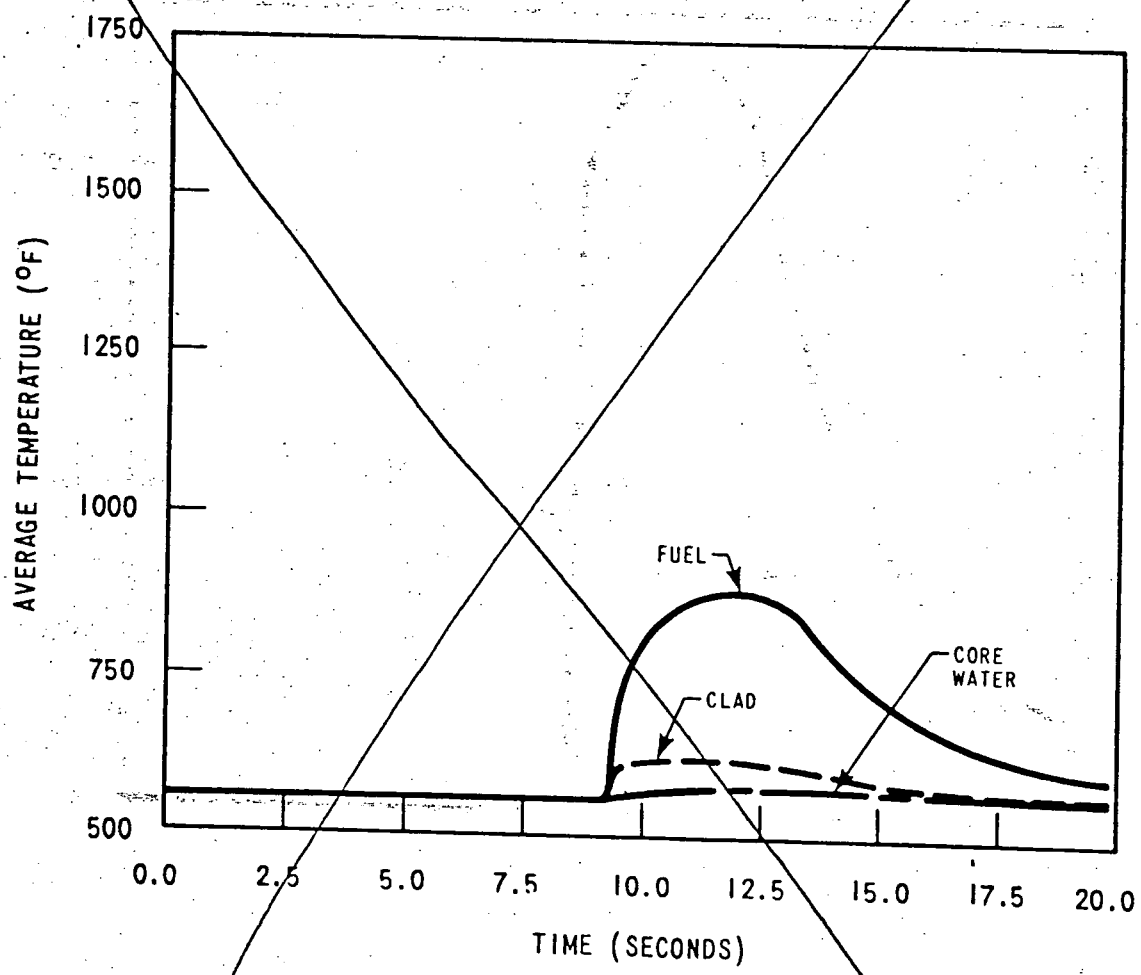


Figure 15.2-3 Uncontrolled Rod Withdrawal from a Subcritical Condition, Temperature Versus Time, Reactivity Insertion Rate $90 \times 10^{-5}$ $\Delta K/Sec$

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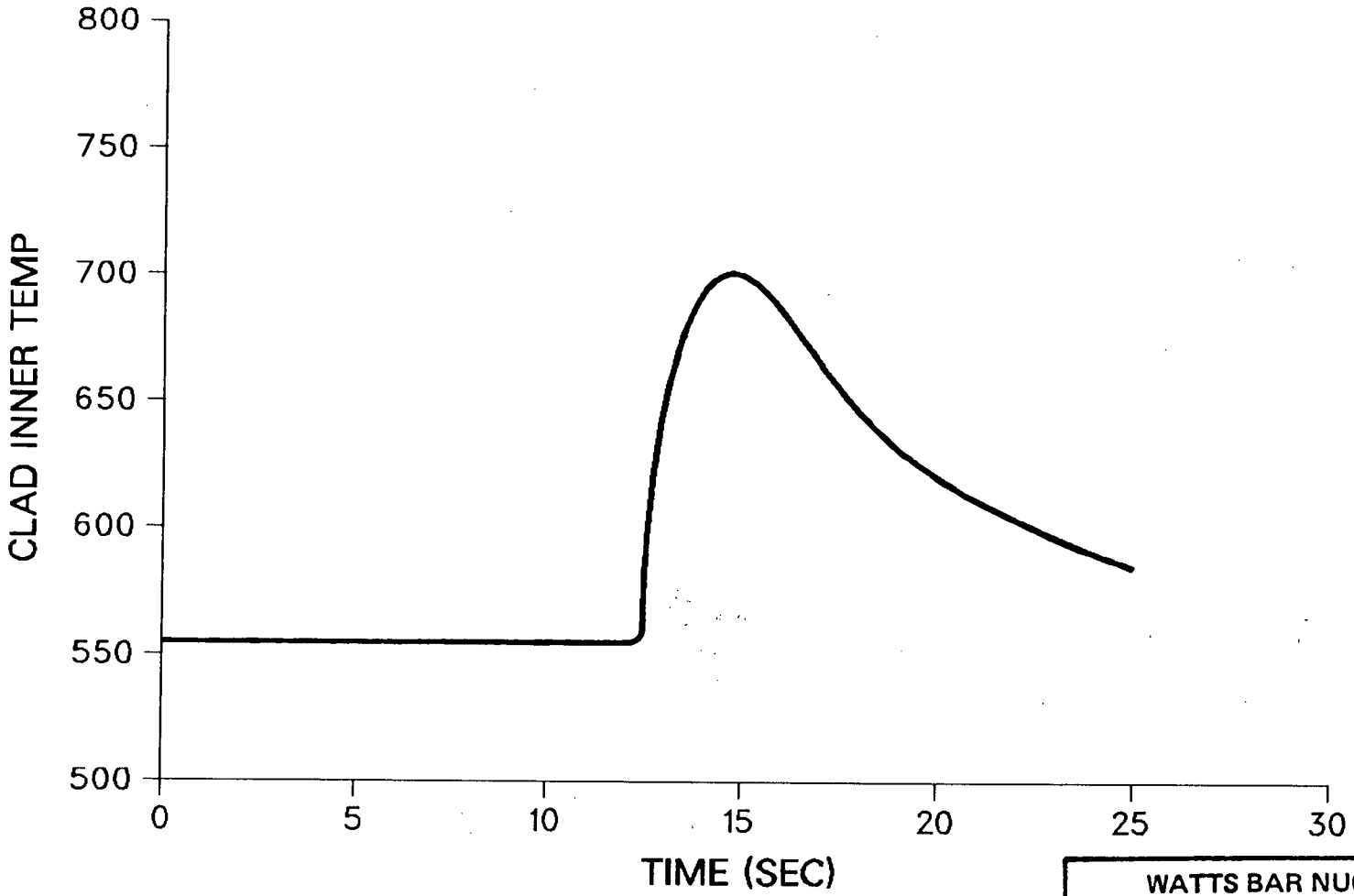
# WAT UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL HOT SPOT CLAD INNER TEMP VS TIME

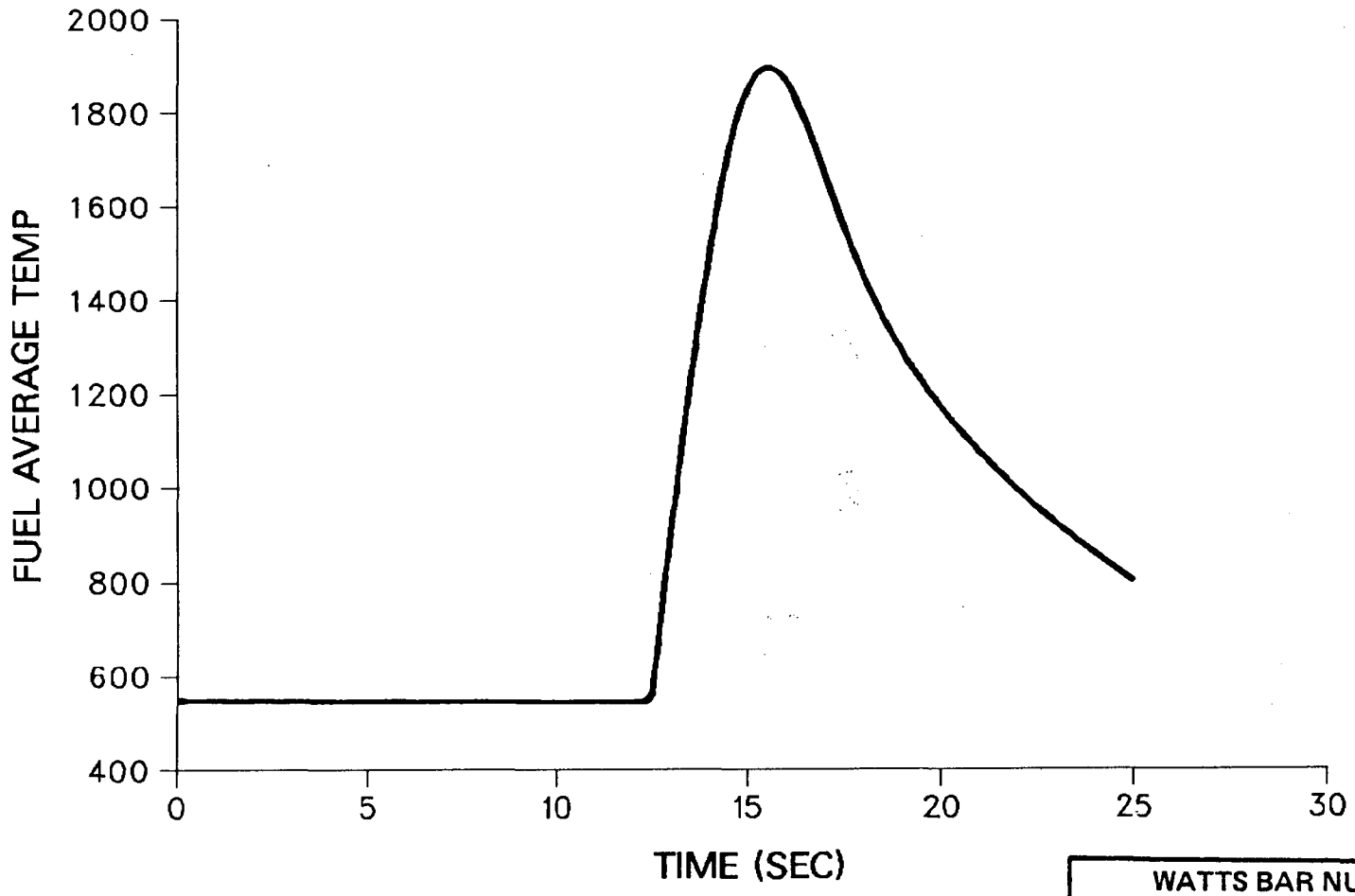



**WATTS BAR NUCLEAR PLANT**
**FINAL SAFETY**
**ANALYSIS REPORT**

UNCONTROLLED ROD WITHDRAWAL FROM A SUBCRITICAL CONDITION, CLAD INNER TEMPERATURE VERSUS TIME, REACTIVITY INSERTION RATE $60 \times 10^{-5}$ $\Delta K/SEC$

FIGURE 15.2-3A

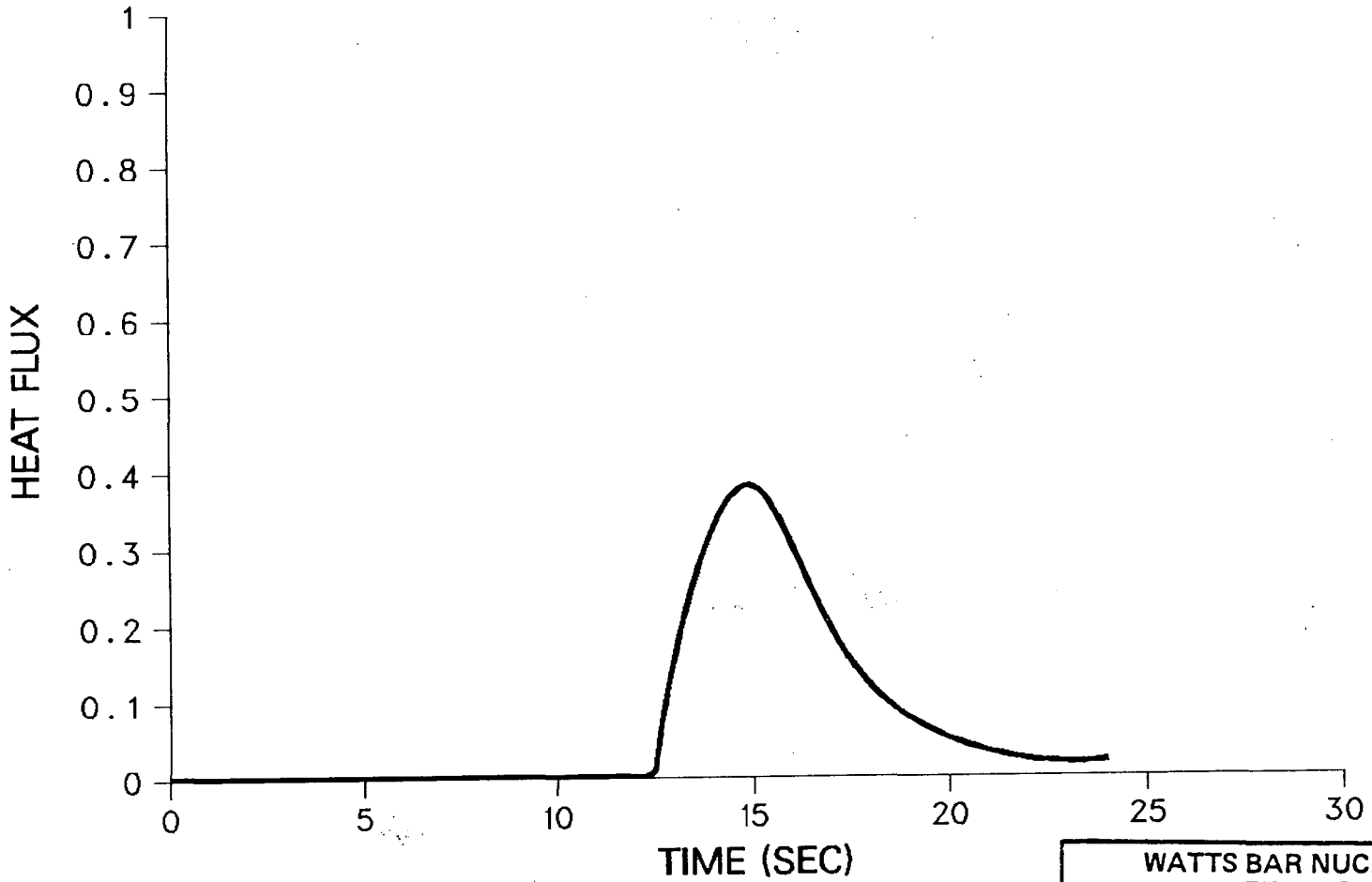
# WAT UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL HOT SPOT FUEL AVG TEMP VS TIME




**WATTS BAR NUCLEAR PLANT**
**FINAL SAFETY**
**ANALYSIS REPORT**

UNCONTROLLED ROD WITHDRAWAL FROM
A SUBCRITICAL CONDITION,
TEMPERATURE VERSUS TIME,
REACTIVITY INSERTION RATE
FUEL AVERAGE $60 \times 10^{-5}$ $\Delta K/SEC$
FIGURE 15.2-3

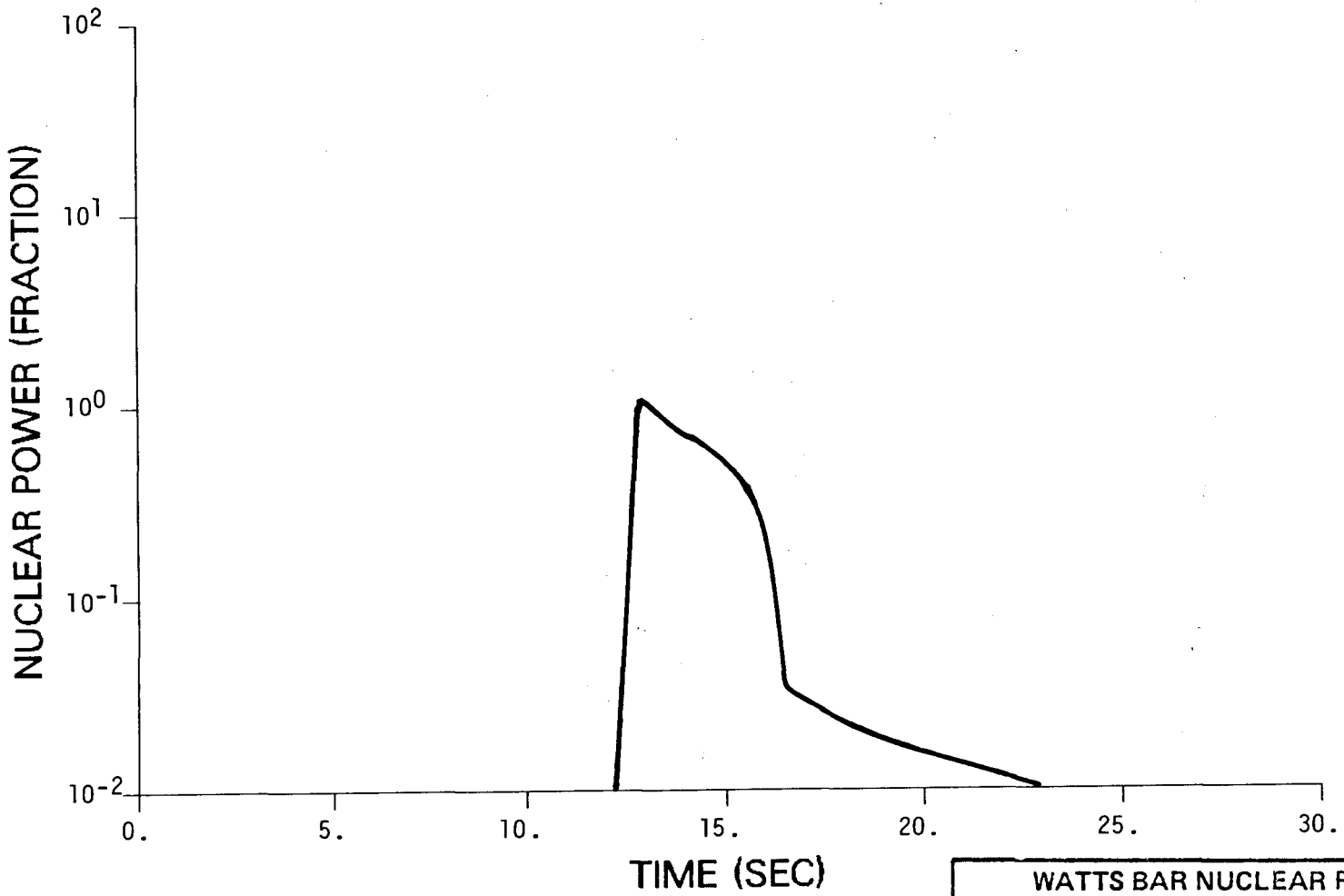
# WAT UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL HEAT FLUX VS TIME




WATTS BAR NUCLEAR PLANT
FINAL SAFETY
ANALYSIS REPORT

UNCONTROLLED ROD
WITHDRAWAL FROM A
SUBCRITICAL CONDITION
HEAT FLUX VS TIME
FIGURE 15.2-2

# WAT UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL NUCLEAR POWER VS TIME




| WATTS BAR NUCLEAR PLANT FINAL SAFETY ANALYSIS REPORT |
| --- |
| UNCONTROLLED RCCA BANK WITHDRAWAL FROM SUBCRITICAL FIGURE 15.2-1 |