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SUBJECT:
FORWARDING SUPPLEMENT TO APPLICANT"S 07/10/78 PROPOSED AMEND TO LIC NOS DPR-31 & 41 RE FULL PWR OPERATION WITH UP TO 25% OF STEAM GENERATOR TUBES PLUGGED, CONSISTING OF "NON-LOCA ACCIDENTS SAFETY EVALUATION FOR HIGHER LEVELS OF STEAM GENERATOR TUBE PLUGGING

PLANT NAME: TURKEY PT #3
TURKEY PT #4

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FLORIDA POWER & LIGHT COMPANY

July 20, 1978
L-78-242

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Office of Nuclear Reactor Regulation
Attention: Mr. Victor Stello, Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Stello:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Supplement to Proposed Amendment to Facility
Operating Licenses DPR-31 and DPR-41

Florida Power & Light Company (FPL) letter L-78-230 of July 10, 1978, contained proposed amendments to the Turkey Point Units 3 and 4 Operating Licenses to allow full power operation with up to 25% of the steam generator tubes plugged. The proposed amendments resulted from an evaluation of non-LOCA transients. An analysis of the LOCA transients, which could lead to further proposed amendments, is scheduled to be submitted by August 10, 1978.

On July 12, after receipt of the FPL July 10 submittal, the NRC staff initiated a telephone conference call to discuss the FPL safety evaluation submitted in support of the proposed amendments. During the phone call, FPL representatives and the NRC staff scheduled a meeting for the purpose of continuing the discussion. The meeting was held on July 13, 1978. As a result of the meeting, it was agreed that FPL would submit a supplement to the July 10 letter containing a corresponding evaluation developed by the NSSS vendor. The NSSS vendor analysis is attached. The difference between it and the FPL analysis is as follows:

a) Section 3.2 - Evaluation

The NSSS vendor did not consider the effects of fuel rod bow in the evaluation and recommended that the generic rod bow penalty on $F_{\Delta H}$ be applied. Where appropriate, FPL modified the evaluation such that it included the effects of a rod bow penalty, consistent with a previous submittal (FPL letter L-77-106 of April 4, 1977) and showed that the rod bow penalty could be absorbed without a reduction in $F_{\Delta H}$. Any other differences are editorial in nature.

b) Section 3.3.1 - Uncontrolled Control Rod Assembly Withdrawal at Power

The NSSS vendor reanalysis did not include the development of new transient curves. Instead the NSSS vendor examined the setpoints and concluded that there was substantial margin in the current

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Please call if you have further questions regarding this or previous submittals.

Very truly yours,

Robert E. Uhrig

Robert E. Uhrig
Vice President

REU/MAS/RDH/cpc

Attachment

cc: Mr. James P. O'Reilly, Region II
Robert Lowenstein, Esquire

(FSAR) Overtemperature ΔT setpoint and that the Overpower ΔT setpoint should be changed. FPL performed a reanalysis of the transients with the setpoints developed by the NSSS vendor with the rod bow penalty included and showed that the DNBR limits were not violated. The Overpower ΔT and Overtemperature ΔT setpoints resulting from the two reanalyses are therefore identical.

c) Section 3.3.2 - Loss of Reactor Coolant Flow (Flow Coast-Down Accident)

The NSSS vendor reanalysis did not consider the effects of fuel rod bow. FPL modified the reanalysis to include a rod bow penalty.

Although the vendor reanalysis did not consider rod bow, the minimum DNBR obtained by the vendor during the transient is 1.48. This value shows that the effect of rod bowing can be accommodated without penalizing any operating parameter.

d) Section 4.0 - Technical Specifications

The only difference between the Technical Specifications proposed by the NSSS vendor and FPL are the Reactor Core Thermal and Hydraulic Safety Limits curves, Figure 2.1-1b. In generating these curves the vendor did not include the effects of rod bow penalty, but FPL did. The FPL curves were developed using the COBRA code exclusively, in conformance with the methods used to generate the companion curves, submitted in FPL letter L-78-217 on June 22, 1978, for lesser amounts of steam generator tube plugging. As the rod bow penalty is already included in these curves, there is no need for any further penalty to the $F_{\Delta H}$ limit or to any other operating parameter.

e) Section 6.0 - References

FPL requests a timely review of their Safety and Fuel Management Analysis Methods (Reference 6) and of the DYNODE-P Code (Reference 7), as these methods and codes are to be used in future analyses to be submitted by FPL in support of licensing actions.

A question was raised at the July 13 meeting concerning the NSSS vendor's use of a modified FLARE code on the Unit 4, Cycle 5 design (refer to Unit 4, Cycle 5 Reload Safety Evaluation, FPL letter L-78-210, dated June 19, 1978). We have discussed the use of this code with NSSS vendor representatives, and they have verified that all uses of the FLARE model for the cycle 5 design were backed up (confirmed) with the use of their standard methods.

NON-LOCA ACCIDENTS SAFETY EVALUATION
FOR HIGHER LEVELS OF STEAM GENERATOR TUBE PLUGGING

JUNE 1978

WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR ENERGY SYSTEMS DIVISION
P.O. BOX 355
PITTSBURGH, PENNSYLVANIA 15230

1.0 INTRODUCTION

The current Turkey Point 3 and 4 safety analyses are valid for steam generator tube plugging levels of up to 19% on Units 3 and 4.

Above these levels, the current LOCA-ECCS accident analysis as well as some of the current non-LOCA accident analyses are not valid because:

1. The RCS flow rate is reduced to below the thermal design value assumed in the current LOCA-ECCS and applicable non-LOCA analyses,

2. the adverse impact of higher steam generator plugging levels on the blowdown and reflood phases of LOCA-ECCS analysis has not been analyzed,

   the reduction in RCS volume (from the plugged steam generator tubes) can have an impact on some of the current non-LOCA analyses and must now be explicitly considered, and

4. the pump coast-down characteristics are more severe than those assumed in the current loss of flow analysis.

It is the purpose of this report to present an evaluation of the applicable non-LOCA accidents, considering the above factors, to demonstrate that with appropriate Technical Specifications changes, Turkey Point 3 and 4 can be operated safety from a non-LOCA accident standpoint with up to 25% of the steam generator tubes plugged. A LOCA-ECCS accident reanalysis for higher than 19% steam generator tube plugging and which considers the above factors is currently being performed, but is not included. It will, however, be submitted as a supplement to this report in the near future.

The evaluation provided in this report was conducted as follows:

1. Determine the RCS flow rate associated with 25% steam generator tube plugging.

2. Evaluate the impact of this tube plugging and the associated RCS flow rate on those significant parameters which influence the results of the applicable non-LOCA accident analyses.

3. Reanalyze those non-LOCA accidents which are either most limiting or most sensitive to the impacts resulting from 25% tube plugging level.

The remainder of this report is organized as follows:

Conservative flow rates versus level of steam generator tube plugging are developed in Section 2. The applicable non-LOCA accident evaluation and reanalyses are provided in Section 3. The required changes to the Technical Specifications are summarized in Section 4. Conclusions are given in Section 5. References are provided in Section 6.

2.0 FLOW RATE VERSUS LEVEL OF STEAM GENERATOR TUBE PLUGGING

Flow measurements have been taken at the Turkey Point Power Station for several levels of steam generator tube plugging. These data were than compared to the flow rates obtained from the analytical model used to calculate the best-estimate flow rate. Deviations between the model prediction and the measurement data points were conservatively accounted for by subtracting a constant bias (equal to the largest deviation between the measurement data and the design prediction) from the model prediction curve of flow rate versus steam generator tube plugging level. This measurement bias corrected curve was then further reduced by a factor of 1.02 to account for measurement instrumentation uncertainty (see Table 1).

The resulting curve of flow rate versus level of steam generator tube plugging is provided in Figure 1. This curve indicates that a tube plugging level of 25% will conservatively result in a flow rate of no more than 5% below the thermal design flow rate of 89,500 gpm per loop. This value, 85,025 gpm per loop, was then used, along with the tube plugging level of 25%, as the basis for the non-LOCA accident evaluation.

3.0 ACCIDENT ANALYSIS

3.1 INTRODUCTION

The impact of higher steam generator tube plugging levels of up to 25% on the non-LOCA accident analyses presented in Chapter 14 of the FSAR has been assessed. The basic approach used was to identify the important parameters for each accident, determine which of these parameters were affected by the higher steam generator tube plugging levels, and then determined how the impacted parameters affected the accident analysis. The resulting impacts were determined by either evaluating the accident to qualitatively demonstrate that the accident is not limiting or re-analyzing the affected accident (if the accident was found to be limiting or very sensitive to the impact of higher steam generator tube plugging levels). The evaluations were consistent with the following assumptions:

	Used In This Report	Used in the Currently Approved Analyses
Thermal design flow, gpm/loop	85,025	89,500
S. G. tube plugging, %	25	19
Maximum allowed power, Mwt	2200	2200
T_{avg} at 100% power, °F	574.2	574.2
ΔT at 100% power, °F	58.9	55.9
$F^N_{\Delta H}$	1.55	1.55 (1.75 FSAR)
F_Q maximum	2.05	2.55 (Non-LOCA)

In general, reanalysis and evaluation techniques were based on the assumptions and methods employed in the FSAR. Exceptions to this policy are noted in the text.

The evaluation of the non-LOCA accidents is presented in Section 3.2. The results of this evaluation indicated that the following accidents were limiting (i.e., the results of the accident analysis are close to safety limits) or most sensitive to the impact of the higher steam generator tube plugging levels:

- Uncontrolled RCCA Withdrawal at Power
- Loss of Reactor Coolant Flow (Flow Coast-Down Accident)
- Chemical and Volume Control System Malfunction

The results of the reanalyses of these accidents are provided in Section 3.3.

3.2 EVALUATION

The following accidents were evaluated and found to have sufficient margin to the accident safety limits.

1. Uncontrolled RCCA Withdrawal From a Subcritical Condition[1]

A control rod assembly withdrawal incident when the reactor is subcritical results in an uncontrolled addition of reactivity leading to a power excursion (Section 14.1.1 of the FSAR). The nuclear power response is characterized by a very fast rise terminated by the reactivity feedback of the negative fuel temperature coefficient. The power excursion causes a heatup of the moderator. However, since the power rise is rapid and is followed by an immediate reactor trip, the moderator temperature rise is small. Thus, nuclear power response is primarily a function of the Doppler temperature coefficient.

The reduction in primary coolant flow is the primary impact which influences this accident. The reduced primary coolant flow results in a decreased core heat transfer coefficient which in turn results

in a faster fuel temperature increase than reported in the FSAR analysis.(1) The faster temperature increase would result in more Doppler feedback thus reducing the nuclear power heat flux excursion, as presented in Reference 1, which would partially compensate for the flow reduction. Therefore, the nuclear transient is only moderately sensitive to the impact of steam generator tube plugging.

The FSAR analysis(1) shows that for a 60 x 10^{-5} Δk/sec reactivity insertion rate, the peak heat flux achieved is 46.8% of nominal. This is conservative for the higher plugging situation for the reasons stated above. The resultant peak fuel average temperature was 951°F. A 5% reduction in flow and the associated reduction in core heat transfer coefficient would degrade heat transfer from the fuel by a maximum 5% and increase the rise in peak fuel and clad temperatures by a maximum of 5%. Therefore, the fuel and clad temperatures would be less than ~1000°F and ~630°F, respectively, for the present evaluation. These values are still significantly below fuel melt (5080°F) and zirconium-H_2O reaction (1800°F) limits, and the impact of increased steam generator tube plugging, up to 25%, would not result in a violation of safety limits.

2. Malpositioning of the Part Length Rods(1)

A malpositioning of a part length rod accident need not be addressed due to Technical Specification restrictions which prohibit power operation with the part length rods in the core.

3. Rod Cluster Control Assembly (RCCA) Drop(1)

The drop of a Control Rod Assembly results in a step decrease in reactivity with produces a similar reduction in core power, thus reducing the coolant average temperature. The highly negative moderator temperature coefficient (-35 pcm/°F) assumed in the analysis results in a power increase (overshoot) above the turbine power

runback value causing a temporary imbalance between core power and secondary power extraction capability. This analysis is potentially sensitive to steam generator tube plugging due to the reduced flow. The effect of a 5% reduction in initial RCS flow would be a smaller reduction in coolant average temperature. Thus the power overshoot would be less than the value shown in section 14.1.4 of the FSAR. Based on the FSAR transient, statepoints were evaluated consistent with a 5% reduction in flow. The results of this DNB evaluation showed that the DNBR limit of 1.30 can be accommodated with margin. Therefore, the impact of increased steam generator tube plugging on the Control Rod Assembly Drop Accident analysis would not appreciably affect the margin to the safety limits.

4. Startup of an Inactive Reactor Coolant Loop[1]

An inadvertent startup of an idle reactor coolant pump with loop stop valves open results in the injection of cold water into the core. This accident need not be addressed due to Technical Specifications restrictions which prohibit power operation with a loop out of service. However, evaluation shows, that the results presented in the FSAR would be conservative for any impacts associated with increased levels of steam generator tube plugging.

5. Reduction in Feedwater Enthalpy Incident[1]

The addition of excessive feedwater and inadvertent opening of the feedwater bypass valve are excessive heat removal incidents which result in a power increase due to moderator feedback. Increased levels of steam generator tube plugging would impact this analysis principally due to the reduced flow.

Section 14.1.7 of the FSAR presents two cases. The first case assumes a zero moderator coefficient, which is used to demonstrate inherent transient attenuation capability during a feedwater reduction. A reduction in flow will have a negligible effect on

stability since the reactivity insertion is identical to the FSAR case due to the zero moderator temperature coefficient. DNB is not a consideration for this case since DNBR's do not fall below the steady state value. This is due to the relatively large reduction in T_{avg}. The reduction in flow, however, will result in the initial steady state DNBR being reduced from 1.8 to ~1.72 (this value corresponds to the DNBR which was calculated at time zero in the Loss of Flow reanalysis using THINC). Thus, adequate margin to steady limits is retained.

The second case assumes a large negative moderator coefficient. The impact of increased steam generator tube plugging (reduction in flow) will result in a slower cooldown and, therefore, a lower reactivity insertion rate than in the FSAR analysis. The integral reactivity insertion due to moderator temperature reduction will be less than the FSAR case, thus producing a lower peak nuclear power. Therefore, the reduction in DNBR from the steady state value (~1.72 for increased steam tube plugging levels) would be no greater than that shown in the FSAR. The FSAR shows a DNBR reduction of ~0.04. Thus, the 5% flow reduction will result in a minimum DNBR of ~1.68, and considerable margin to safety limits.

Evaluation has shown that sufficient margin is available to the safety limits for the Feedwater System Malfunction Accident for increased levels of steam generator tube plugging. In addition, further protection is assured for both cases via the identified margin in the Overtemperature ΔT protection setpoints (see Section 3.3.1).

6. Excessive Load Increase Incident[1]

An excessive load increase event, in which the steam load exceeds the core power, results in a decrease in reactor coolant system temperature, which is very similar to the feedwater malfunction

analysis. As in the Feedwater Malfunction Accident, reduced flow is the principal impact on this accident due to increased levels of steam generator tube plugging.

Two Excessive Load Increase cases are presented in FSAR Section 14.1.8.

-10% Step load increase with manual control.
-10% Step load increase with automatic control.

The worst case results (automatic control) indicate that with no trip actuation, steady state conditions are reached with a minimum DNBR of >1.30. Margin is available in that the $F_{\Delta H}$ used in the FSAR analysis was 1.75 as compared to the current value of 1.55. This results in a DNB benefit of ~30%. The impact of increased steam generator tube plugging on the Excessive Load Increase Accident is an overall reduction in DNBR of approximately 5%. Thus there is considerable margin available given all of the assumptions in the original analysis remain valid. Tube plugging does result in small changes to the initial conditions, however, these changes tend to be in the conservative direction.

Another area of margin available to offset the flow reduction penalty is in the Overtemperature ΔT trip which is used for protection in the case with automatic control. Margin has been identified in the Overtemperature ΔT trip protection setpoints (see Section 3.3.1). As stated in the FSAR, the adequacy of this protection was verified in the Rod Withdrawal at Power Accident (see FSAR Section 14.1.2).

7. <u>Loss of Reactor Coolant Flow (Locked Rotor Accident)</u>[2]

The FSAR (Section 14.1.9) shows that the most severe Locked Rotor Accident is an instantaneous seizure of a reactor coolant pump rotor

at 100% power with three loops operating. Following the incident, reactor coolant system temperature rises until shortly after reactor trip.

The impact on the Locked Rotor Accident of increased steam generator tube plugging will be primarily due to the reduced flow. These impacts will not affect the time to DNB since DNB is conservatively assumed to occur at the beginning of the transient. The flow coast-down in the affected loop due to the Locked Rotor is so rapid that the time of reactor trip (low flow setpoint is reached) is essentially identical to that presented in Reference 2. Therefore, the nuclear power and heat flux responses will be the same as shown. However, the reduction in flow would result in slightly higher calculated system pressures and fuel and clad temperatures.

The currently applicable analysis[2] shows a calculated peak fuel temperature of 2940°F and a peak clad temperature of 1500°F. Peak temperatures are highly sensitive to the initial hot spot values assumed in the analysis. The above analysis was based on a hot spot heat transfer calculation which employed heat flux and fuel temperatures based on an F_Q of 2.55. LOCA considerations require that an F_Q limit of ≤ 2.05 (value assumes axial stack height and spike penalties) be used. This results in a 20% reduction in total energy input to the hot spot which will more than compensate for the 5% reduction in flow. Consequently, the expected peak fuel and clad temperatures would remain below the results of the currently applicable analysis.

It is estimated that the peak system pressure will increase ~80 psia above the previous value, however, the maximum calculated value was 2720 psia. This is significantly below the pressure at which vessel stress limits are exceeded (~400 psia exists to this limit), thus, considerable margin exists to absorb any slight pressure increase. (It should be noted that the 25% reduction in the

number of steam generator tubes would result in approximately a 10% reduction in primary coolant mass which would decrease the heat capacity of the RCS by the same amount. This would not result in higher peak temperatures or pressures, however, since the peak values are reached in considerable less than one loop transport time constant.)

Therefore, operation at reduced flow will not cause safety limits to be exceeded for a Locked Rotor Accident.

8. Loss of External Electrical Load[1]

The result of a loss of load is a core power level which momentarily exceeds the secondary system power extraction causing an increase in core water temperature.

The impact of increased levels of steam generator tube plugging would be again principally due to the reduced flow and the decreased RCS mass inventory. Two cases, analyzed for both beginning and end of life conditions, are presented in Section 14.1.10 of the FSAR:

a. Reactor in automatic rod control with operation of the pressurizer spray and the pressurizer power operated relief valves; and

b. Reactor in manual rod control with no credit for pressurizer spray or power operated relief valves.

The FSAR analysis[1] results in a peak pressurizer pressure of 2517 psia following reactor trip and a minimum DNBR of 1.61. A reduction in loop flow and RCS mass inventory will result in a more rapid pressure rise than is currently shown. The effect will be minor, however, since the reactor is tripped on high pressurizer pressure. Thus, the time to trip will be decreased which will result in a lower total energy input to the coolant. Therefore,

although the initial margin to DNB will be reduced, the minimum transient DNBR will be only slightly affected and the margin to the safety limits will be maintained. In addition, the identified margin in the Overtemperature ΔT setpoints (see Section 3.3.1) will assure adequate margin to DNB.

9. Loss of Normal Feedwater(1)

This transient is analyzed to determine that the peak RCS pressure does not exceed allowable limits and that the core remains covered with water. These criteria are assured by applying the more stringent requirement that the pressurizer must not be filled with water.

Increased steam generator tube plugging levels would impact the accident principally due to reduced flow. The effect of these impacts would be a larger and more rapid heatup of the primary system. The resulting coolant density change would increase the volume of water in the pressurizer.

The analysis results presented in Section 14.1.11 of the FSAR show that considerable margin is available. This analysis shows that the peak pressurizer volume reached is 1100 ft^3 on an approximate 300 ft^3 change in volume. This result was due to a 26°F change in coolant average temperature. Using the highly conservative assumption that the average temperature delta would increase by 50% due to flow reductions, this would result in a maximum increase of less than 150 ft^3 in liquid volume. This is still below the 1300 ft^3 capacity of the pressurizer thus no reanalysis is necessary.

Therefore, the results of this reevaluation indicate that the impact of increased levels of steam generator tube plugging will allow sufficient margin to be maintained to the safety limit associated with the Loss of Normal Feedwater Accident.

10. Loss of AC Power (1)

This transient is analyzed to show that upon loss of all AC power to the unit auxiliaries, the auxiliary feedwater system is sufficient to remove stored and residual heat without water release through pressurized relief valves.

As in a Loss of Normal Feedwater Accident, increased steam generator tube plugging would impact the Loss of AC Power transient primarily due to reduced flow.

Upon the loss of power to the reactor coolant pumps, coolant flow neces- sary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. In the FSAR analysis, the natural circulation flow was calculated using an analytical method based on the condition of equilibrium flow and maximum loop resistance. The reduction in natural circulation flow rates due to the 5% reduction in thermal design flow is negligible.

In addition, due to the relatively long duration of the transient following trip, the results are highly sensitive to residual (decay) heat generation. Residual heat generation is directly proportional to initial power level preceding the trip. The accident assumed the power to be at 102% of the maximum turbine rating 2300 Mwt. Thus, the total energy input to the system would be ~4.3% less than assumed in the FSAR.

This re-evaluation indicates that the impact of increased levels of steam generator tube plugging will not adversely impact the Loss of AC Power Accident. Sufficient margin to the safety limits are maintained.

11. Rupture of a Steam Pipe (3)

The steamline break transient is analyzed for hot zero power, end of life conditions (Section 14.2.5 of the FSAR) for the following cases:

- Hypothetical Break (steam pipe rupture)
  - Inside Containment with and without power
  - Outside Containment with and without power

- Credible Break (Dump valve opening)

A steamline break results in a rapid depressurization of the steam generators which causes a large reactivity insertion to the core via primary cooldown. The acceptance criteria for this accident is that no DNB must occur following a return to power. This limit, however, is highly conservative since steam line break is classified as a Condition III event. As such, the occurrence of DNB in small regions of the core (~5%) would not violate NRC acceptance criteria.

The impact of increased levels of steam generator tube plugging would affect the accident principally due to the reduced flow, reduced RCS inventory, and reduced heat transfer coefficient. These impacts would result in changed cooldown and feedback reactivity characteristics such that the return to power as shown in the previous analysis[3] would be slightly conservative with respect to the lower initial flow conditions. In addition, the time of Safety Injection actuation would be unaffected by flow conditions for the Hypothetical Breaks. This coupled with the slightly slower return to power would result in a reduction in peak average power for the cases with and without power and indicate results conservative with respect to the current analysis.

Thus the impact of increased levels of steam generator tube plugging will not result in a violation of Westinghouse or NRC safety limits.

12. Rupture of a Control Rod Drive Mechanism Housing, RCCA Ejection[2]

The rupture of a control rod drive mechanism housing which allowed a control rod assembly to be rapidly ejected from the core would result in a core thermal power excursion. This power excursion would be limited by the Doppler reactivity effect as a result of the increased fuel temperature and would be terminated by a reactor trip activated by high nuclear power signals.

The rod ejection transient is analyzed at full power and hot standby for both beginning and end of life conditions (Section 14.2.6 of the FSAR). Reduced core flow is the primary impact resulting from increased levels of steam generator tube plugging. This impact would result in a reduction in heat transfer to the coolant which would increase clad and fuel peak temperatures. The current analysis[2] results and inputs are summarized in Table II. As is shown, all cases have significant margin to fuel failure limits. The effect of reducing flow by 5% is to primarily increase the peak clad temperatures by $\sim 50^{\circ}F$. The current analysis shows that for all cases a value of at least $400^{\circ}F$ can be accommodated before peak clad limits are reached ($2700^{\circ}F$). The fuel temperatures will also increase, however, they will increase much less than the clad increase due to the rapid nature of the transient.

In addition, there is a significant degree of conservatism in the inputs. The ejected rod worths and post ejection peaking factors are >5% above the calculated Turkey Point reload values. Also for the fuell power cases, the initial hot spot fuel temperatures were calculated assuming an F_Q of 2.55. Due to LOCA considerations, the F_Q limit will be <2.05. This results in more than a $175^{\circ}F$ reduction in initial fuel temperature which translates into a $\sim 75^{\circ}F$ reduction in peak transient fuel temperatures which will compensate for the reduction in thermal design flow.

Therefore, the impact of increased levels of steam generator tube plugging on the Rod Ejection Accident will not significantly reduce the margin to the safety limit due to the conservative inputs and large margin to the limits.

3.3 REANALYSIS

The following accidents were reanalyzed because they were either limiting or were sensitive to the impact of increased steam generator tube plugging.

1. Uncontrolled Control Rod Assembly Withdrawal at Power[1]

An uncontrolled control rod assembly withdrawal at power produces a mismatch in steam flow and core power, resulting in an increase in reactor coolant temperature. Increased steam generator tube plugging will impact the analysis principally due to the influence of the reduced flow, the elevated outlet temperature, and the increased loop transit time. The first two impacts will result in less initial margin to DNB. The third impact requires new values of lead/lag time constants to be determined for the Overtemperature ΔT setpoint equation.

The Uncontrolled Rod Withdrawal at Power Accident analysis utilized the overtemperature ΔT trip for protection during "slow" reactivity insertion events and the high neutron flux trip for "fast" reactivity insertion rates. For the fast or high insertion rates, the DNBRs reported in the FSAR would be reduced by $\sim$5% due to the flow reduction. The DNBR's reported in the FSAR were analyzed assuming an $F_{\Delta H}$ of 1.75; however, the current $F_{\Delta H}$ limit is 1.55. This results in greater than 20% addition DNBR margin. Thus the flow reduction penalty is more than offset by $F_{\Delta H}$ benefit for the fast reactivity insertions.

For the slow reactivity insertion rates, the overtemperature ΔT trip function is used. This trip is based on the Reactor Core Thermal and Hydraulic Limits. To assure adequate core protection, the Reactor Core Thermal and Hydraulic Safety Limits have been recalculated consistent with the reduction in RCS flow. Based on these new protection lines, the Overtemperature and Overpower ΔT setpoint equations constants have been recalculated consistent with the new Core Limits and the methods outlined in Reference 5.

Table III gives a comparison of the original (FSAR) and the recalculated constants. For the overtemperature equation, the FSAR values are significantly more limiting than the recalculated values. This is directly due to the $F_{\Delta H}$ reduction from 1.75 to 1.55 in recalculating the Reactor Core Thermal anhd Hydraulic Safety Limits. To offset the effects of an RCS flow reduction, the FSAR overtemperature ΔT trip constants will be maintained in the Technical Specifications (see Section 4.0).

For the most limiting case shown in the FSAR (FSAR Figure 14.1.2-6), the DNBR is reduced from a steady state value of 1.62 down to ∿1.30. Assuming a reactor coolant flow of 85025 gpm/loop and a 1.55 $F_{\Delta H}$ limit, the steady state DNBR is >1.7. By using the overtemperature setpoints derived in the FSAR, approximately the same reduction in DNBR would occur for the limiting reactivity insertion rate. Thus a minimum DNBR of ∿1.5 is anticipated. Therefore, considerable margin is available to the DNB limit to offset minor system changes due to a flow reduction.

As anticipated, Table III shows that the overpower ΔT setpoints have become more limiting with the flow reduction (see Reference 5 for a discussion of this function). This will require a Technical Specification change as shown in Section 4.0. However, no additional analysis is required since the overpower ΔT trip function is not explicitly used in any of the FSAR transients analyzed. Rather this trip function is considered a diverse means of trip for the Power Range Nuclear Instrumentation System.

The above evaluation shows that the conclusions presented in the FSAR are still valid. That is, the core and reactor coolant system are not adversely affected since the nuclear flux and overtemperature ΔT trips prevent the minimum DNB ratio from falling below 1.30 for this incident.

2. Loss of Reactor Coolant Flow (Flow Coast-Down Accident)[2]

As demonstrated in the FSAR, Section 14.1.9 the most severe loss of flow transient is caused by the simultaneous loss of electrical power to all three reactor coolant pumps. This transient was reanalyzed to determine the effect of steam generator tube plugging on the minimum DNBR reached during the incident. Tube plugging will result in a significant decrease in margin to safety limits due to the following effects:

- High loop resistances result in a more rapid flow coast-down.

- Lower initial flows result in less margin to the 1.30 DNBR limits.

Thus reanalysis is required.

Analysis methods and assumptions used in the reanalysis were consistent with those employed in the FSAR. These assumptions include:

a. Initial operating conditions most adverse with respect to the margin to DNBR, i.e., maximum steady state power level (102% of nominal), minimum pressure (2220 psia), and maximum core average temperature (578.2°F);

b. Maximum Doppler power and temperature coefficient and most positive moderator temperature coefficients;

c. Time from loss of power to all pumps to the initiation of control rod assembly motion (reactor trip) of 1.6 seconds; and

d. 4% Δk trip reactivity from full power using frozen feedback.*

e. Loop resistances equivalent to 25% tube plugging.

The flow coastdown was calculated by the PHOENIX[6] code, and the resultant system transient was simulated using the LOFTRAN[7] code. The calculation was performed for a zero constant moderator coefficient value.

The minimum value of DNBR presented in the FSAR for this incident was 1.31. The DNB evaluation performed in the reanalysis was based on the evaluation model described in WCAP-8075[4]. Figures 2 through 6 show the flow coastdown, nuclear power, heat flux, and minimum DNB ratio vs. time. Figure 6 clearly shows that the effect of rod bowing can be accommodated with margin in this analysis.

Steam generator tube plugging appreciably affects the results of the complete loss of flow transient; however, the minimum DNBR remains

*The reactivity rate versus control rod insertion following a reactor scram used in this analysis is slightly less (i.e., more conservative than that used in the previously applicable analyses. (Please note that the conclusion of the previously applicable analyises are not affected by the use of the new reactivity rate insertion curve).

above 1.30 for this incident. The three pump case was reanalyzed since it is the most limiting one presented in the FSAR. Loss of a single pump with all loops in service or with a single loop out of service were less limiting.

3. Chemical and Volume Control System Malfunction

Section 14.1.5 of the Turkey Point Units 3 & 4 FSAR, shows that for a boron dilution event the operator has sufficient time to identify the problem and terminate the dilution before the reactor returns critical or loses shutdown capability. The standard acceptance criteria and FSAR calculated values for operator action are summarized below:

Mode	FSAR (minutes)	Acceptance Criteria
Refueling	> 120	30
Startup	> 240	15
Power		
a. Manual Control	> 15	15
b. Automatic Control	21	15

Steam generator tube plugging has no affect on the analysis at refueling conditions since only the reactor vessel volume is assumed active. The coolant loop volume are conservatively assumed stagnant.

For dilution during startup and at power, there is an effect due to the reduction in primary coolant volume. The effective volume of primary coolant in the steam generator tubes is assumed to be reduced by 25% ($\sim$ 510 ft^3). Thus the total volume assumed in the analysis has been reduced from 7800 ft^3 to 7290 ft^3. This translates into approximately a 7% reduction in the originally calculated dilution-time from startup conditions (240 minutes). The result is still significantly greater than the required operator action time, therefore no safety concern exist.

For dilutions during power operation a highly conservative reactivity insertion rate of 1.1×10^{-5} δk/sec was assumed in the FSAR consistent with an initial boron concentration of 1200 ppm. FSAR Figure 14.1.5-1 (Reactivity Insertion Rate vs Boron Concentration) has been recalculated consistent with the lower primary value and is shown in Figure 7. The results show that the reactivity insertion rate assumed in the FSAR is still conservative. Therefore no additional analysis is required. It should be noted, however, that the FSAR analysis is still highly conservative with respect to the current cycles since the analysis assumed that only 1% shutdown margin is available. The Turkey Point Units have been designed such that > 2.5% shutdown margin is always available for BOL conditions. The result is that operator action times would be > 70 minutes with more a realistic value.

Thus for all cases it has been shown that steam generator tube plugging up to 25% will not affect the safety conclusions of the FSAR for boron dilution.

4.0 TECHNICAL SPECIFICATIONS

This section contains the technical content of proposed changes to the Technical Specifications. These changes are consistent with the plant operation necessary for the design and safety evaluation conclusions stated previously to remain valid. This report is contingent upon approval of these changes.

Delete the following three pages of the technical specifications:

1. TS 2.3-2a
2. TS 3.1-7a
3. TS 2.3-3

Add the appropriately marked pages which follow.

Reactor Coolant Temperature

Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - 0.0107\ (T-574) + 0.000453\ (P-2235) - f(\Delta q)\right]$

$\leq \Delta T_o$ = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta q)$ = a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during startup tests such that:

For $(q_t - q_b)$ within +10 percent and -14 percent where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, $f(\Delta q) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds +10 percent, the Delta-T trip setpoint shall be automatically reduced by 3.5 percent of its value at interim power.

For each percent that the magnitude of $(q_t - q_b)$ exceeds -14 percent, the Delta-T trip setpoint shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.095*

(Two Loop Operation) = 0.88

---

*K_1 = 1.095 for steam generator tube plugging $\leq$ 25 percent

Over-power $\Delta T \leq \Delta T_o \left[1.11^{*} - K_1 \frac{dT}{dt} - K_2 (T - T') - f(\Delta q) \right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

T' = Indicated average temperature at nominal conditions and rated power, F

K_1 = 0 for decreasing average temperature, 0.2 sec./F for increasing average temperature

K_2 = 0.00068 for T equal to or more than T'; 0 for T less than T'

$\frac{dT}{dt}$ = Rate of change of temperature, F/sec

$f(\Delta q)$ = As defined above

Pressurizer

Low Pressurizer pressure - equal to or greater than 1835 psig.

High Pressurizer pressure - equal to or less than 2385 psig.

High Pressurizer water level - equal to or less than 92% of full scale.

Reactor Coolant Flow

Low reactor coolant flow - equal to or greater than 90% of normal indicated flow.

Low reactor coolant pump motor frequency - equal to or greater than 56.1 Hz

Under voltage on reactor coolant pump motor bus - equal to or greater than 60% of normal voltage

Steam Generators

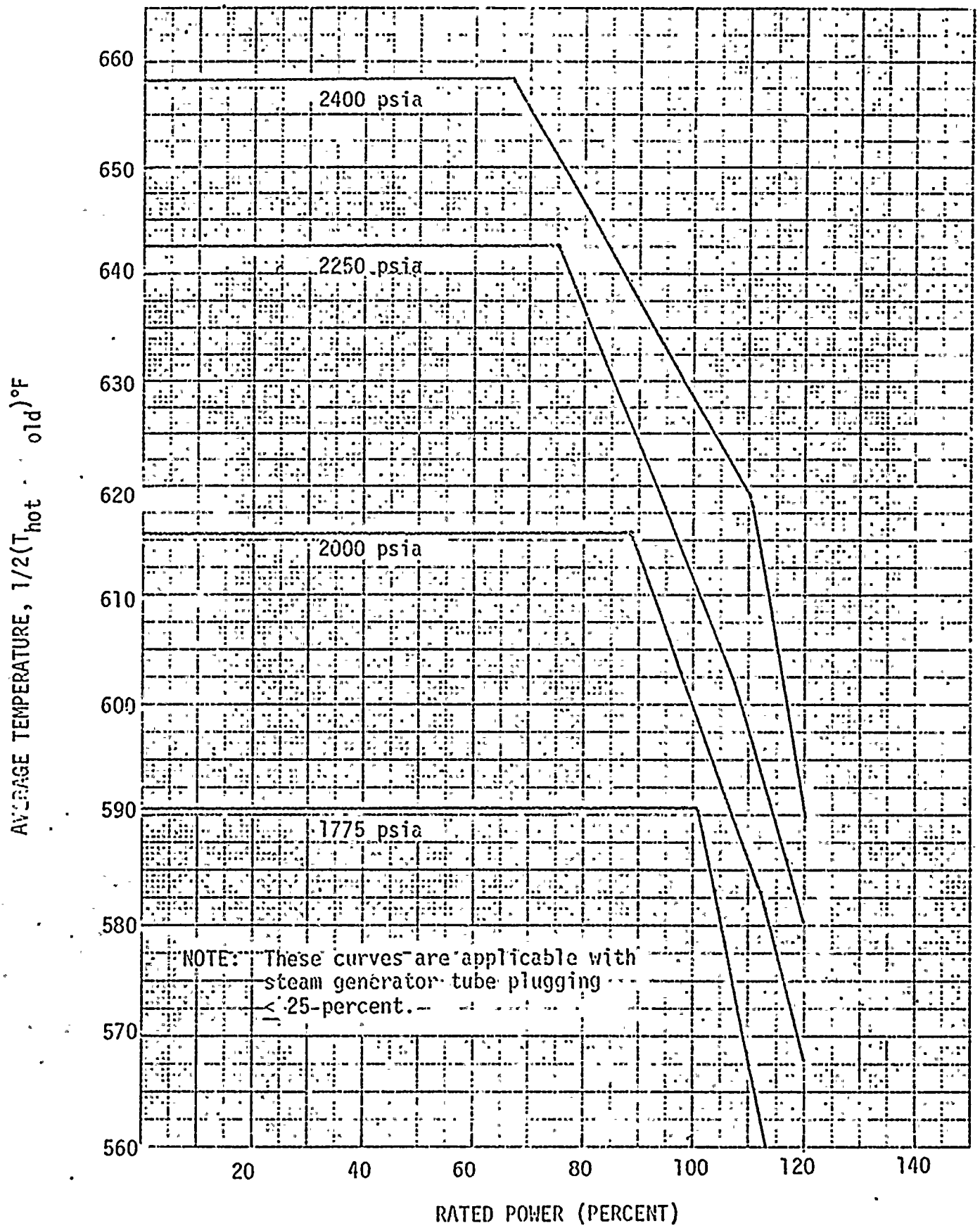
Low-low steam generator water level - equal to or greater than 5% of narrow range instrument scale

*This factor is 1.11 for steam generator tube plugging $\leq$ 15 percent

This factor is 1.10 for steam generator tube plugging > 15 percent and $\leq$ 19

This factor is 1.08 for steam generator tube plugging $\leq$ 25 percent

REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS,
THREE LOOP OPERATION



5.0 CONCLUSIONS

Based on an analytical and empirical study of the Turkey Point Units 3 and 4 flow characteristics, a 25% steam generator plugging level was conservatively determined to result in a flow reduction 5% below thermal design flow.

The impact of this higher steam generator tube plugging level on the non-LOCA accident analyses applicable to Turkey Point Units 3 and 4 has been assessed. The reevaluation has indicated that the impacts associated with increased steam generator tube plugging can be accommodated with margin to the safety limits for the following accident analyses:

- Uncontrolled RCCS Withdrawal from a Subcritical Condition
- Malpositioning of a Part Length Rod
- Rod Cluster Control Assembly (RCCA) Drop
- Start-up of an Inactive Reactor Coolant Loop
- Reduction in Feedwater Enthalpy Incident
- Excessive Load Increase Incident
- Loss of Reactor Coolant Flow (Locked Rotor Accident)
- Loss of External Electrical Load
- Loss of Normal Feedwater
- Loss of A.C. Power
- Rupture of a Steam Pipe
- Rupture of a Control Rod Mechanism Housing

Several accidents were determined to be either limiting or most sensitive to the impact of higher steam generator tube plugging levels. The following accidents were reanalyzed or required partial reanalysis:

- Uncontrolled Control Rod Withdrawal at Power
- Loss of Reactor Coolant Flow (Flow Coast-down Accident)
- Chemical and Volume Control System Malfunction

6: DNB PARAMETERS

The following DNB related parameters limits shall be maintained during power operation:

a. Reactor Coolant System Tavg < 578.2°F
b. Pressurizer Pressure ≥ 2220 psia*
c. Reactor Coolant Flow > 268,500 gpm+

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power using normal shutdown procedures.

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours.

Compliance with c. is demonstrated by verifying that the parameter is within its limits after each refueling cycle.

* Limit not applicable during either a THERMAL POWER ramp increase in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step increase in excess of (10%) RATED THERMAL POWER.

\+ Reactor Coolant Flow < 268,500 gpm for steam generator tube plugging < 15%.

Reactor Coolant Flow ≥ 263,130 gpm for steam generator tube plugging > 15% and ≤ 19%.

Reactor Coolant Flow ≥ 255,075 gpm for steam generator tube plugging > 19% and ≤ 25%.

Based on these reanalyses, it was determined that the results meet the appropriate safety limits with the conservative Technical Specifications changes to the Reactor Core Thermal and Hydraulic Safety Limits.

Therefore, it is concluded that the impact of increased steam generator plugging levels, up to 25%, can be accommodated without adversely affecting the safe operation of Turkey Point Units 3 and 4.

6.0 REFERENCES

1. "Final Safety Analysis Report, Turkey Point Plant, Units 3 and 4," Docket Nos. 50-250, 50-251.

2. "Reload Safety Evaluation, Turkey Point Plant, Unit 4, Cycle 4," January 1977.

3. "Reload Safety Evaluation, Turkey Point Plant, Unit 4, Cycle 5," June 1978.

4. "Fuel Densification, Turkey Point Plant Unit No. 3," WCAP-8074 (Proprietary) and WCAP-8075 (Non-Proprietary), February 1973.

5. "Design Basis for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," WCAP-8745 (Proprietary) and WCAP-8746 (Non-Proprietary), March 1977.

6. "Calculation of Flow Coastdown After Loss of Reactor Coolant Pump (PHOENIX Code)," WCAP-7551, August 1970.

7. LOFTRAN Code Description, WCAP-7878, March 1972.

TABLE 1

REACTOR COOLANT FLOW MEASUREMENT UNCERTAINTY

Parameter Uncertainty		RC Loop Flow Uncertainty
Feedwater Flow	±1.25%	±1.25%
Feedwater Temp.	±1.0°F	±0.2%
Steam Pressure	±30 psi	±0.1%
T_{Hot}	±0.5°F	±0.9%
T_{Cold}	±0.5°F	±0.9%
RCS Pressure	±50 psi	±0.2%
		±1.8% Σ^2

TABLE II

	BOL	BOL	EOL	EOL
Power Level, %	102	0	102	0
Ejected rod work, %Δk	.35	.71	.30	.84
Delayed neutron fraction, %	.50	.50	.44	.44
Feedback reactivity weighting	1.20	1.30	1.2	2.6
Trip rod shutdown, %Δk	5	2	4	2
F_Q before rod ejection	2.55	–	2.55	–
F_Q after rod ejection	5.48	7.0	5.52	14.3
Number of operating pumps	3	2	3	2
Initial fuel temperature, °F	2660	547	2475	547
Maximum fuel pellet average temperature, °F	3990	2040	3490	3080
Maximum clad average temperature, °F	2370	1580	2125	2270
Fuel Pellet Melting, %	0	0	0	0

TABLE III

ΔT Protection Setpoints

Overtemperature ΔT

Technical Specification Section 2

Setpoint $\leq \Delta T_o [K_1 - A (T-574) + B(P-2235) - f (\Delta q)]$

	FSAR Analysis Values	Current Limit (19% S.G. Plugging)	Recalculated Values (25% S.G. Plugging)
K_1	1.095	1.08	1.18
	.0107	.0107	.00959
B	.000453	.000453	.000420

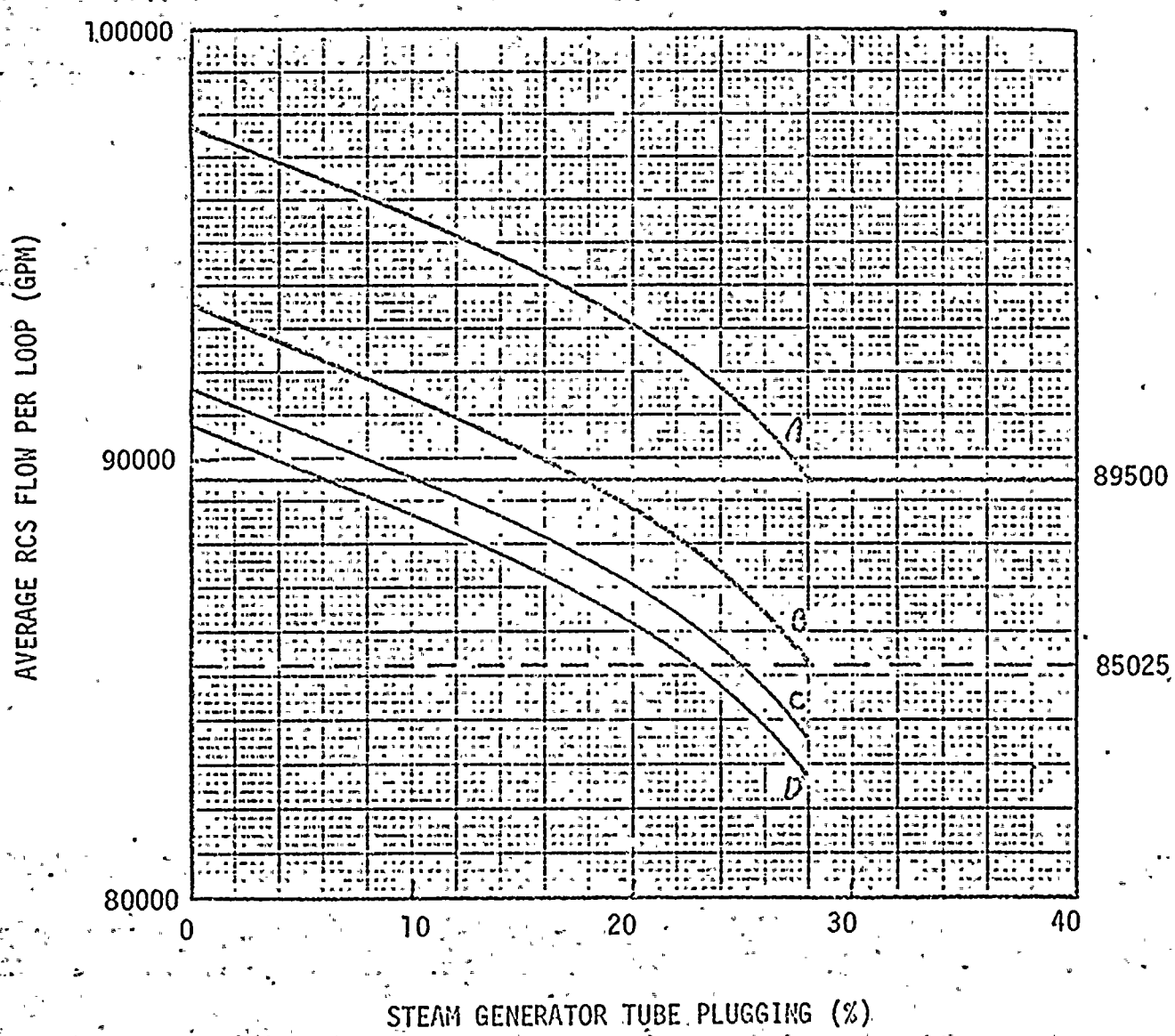
Overpower ΔT

Technical Specification Section 2

Setpoint $\leq \Delta T_o \left[K_o - K_1 \frac{dT}{dt} - K_2 (T-T') - f(\Delta q)\right]$

K_o	1.11	1.10	1.08
K_1	0.2	0.2	0.2
K_2	.00068	.00068	.00106

FIGURE 1
TURKEY POINT UNITS 3 AND 4
RCS FLOW VS STEAM GENERATOR TUBE PLUGGING



A -- Best Estimate Calculated RCS Flow

B -- Curve A Minus 4200 GPM: Lowest Expected Measurement

C -- Curve B Times 0.98

D -- Curve B Times 0.97

FIGURE 2

LOSS OF FLOW

CORE FLOW VS TIME

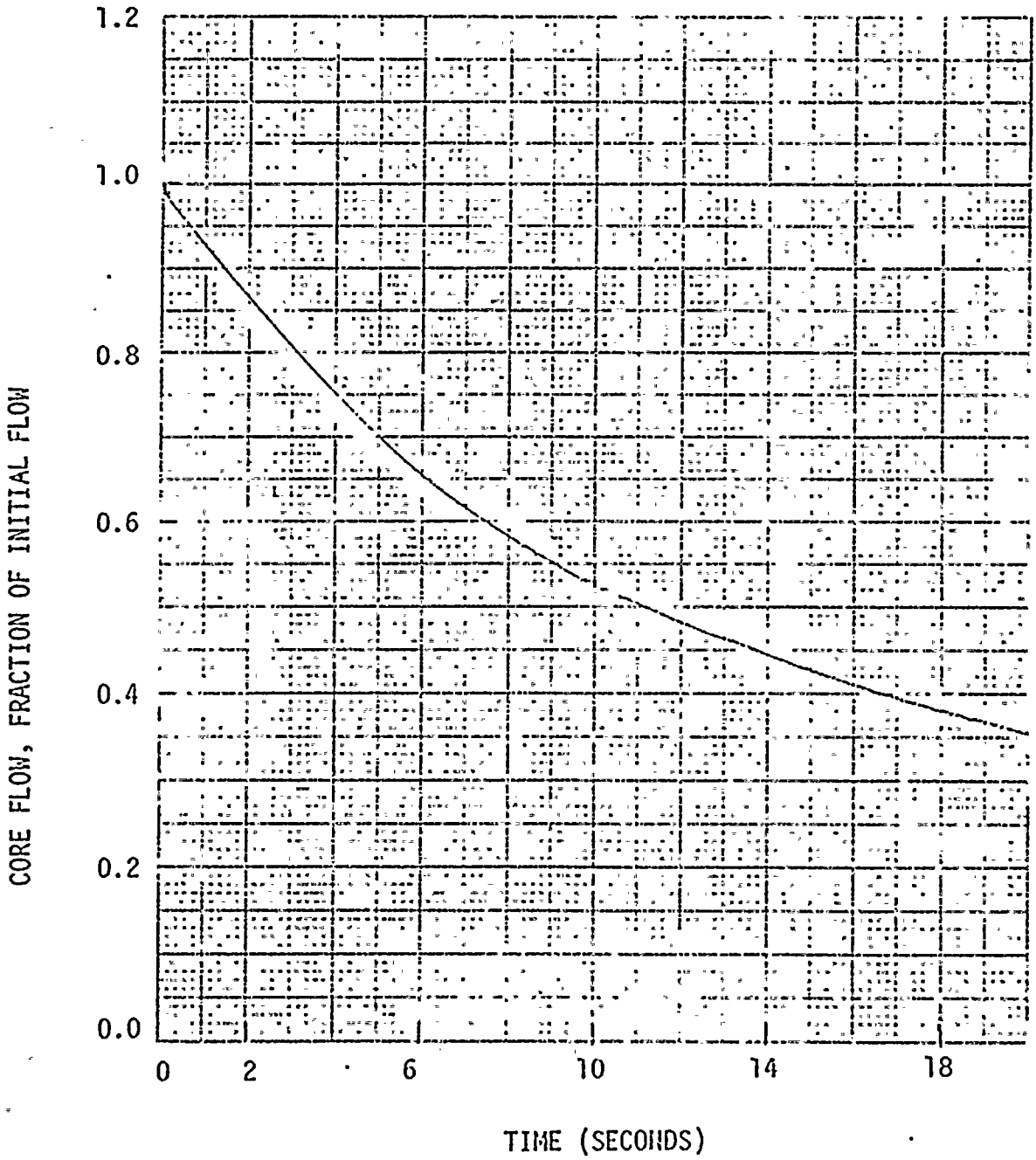


FIGURE 3

LOSS OF FLOW

NUCLEAR POWER VS TIME

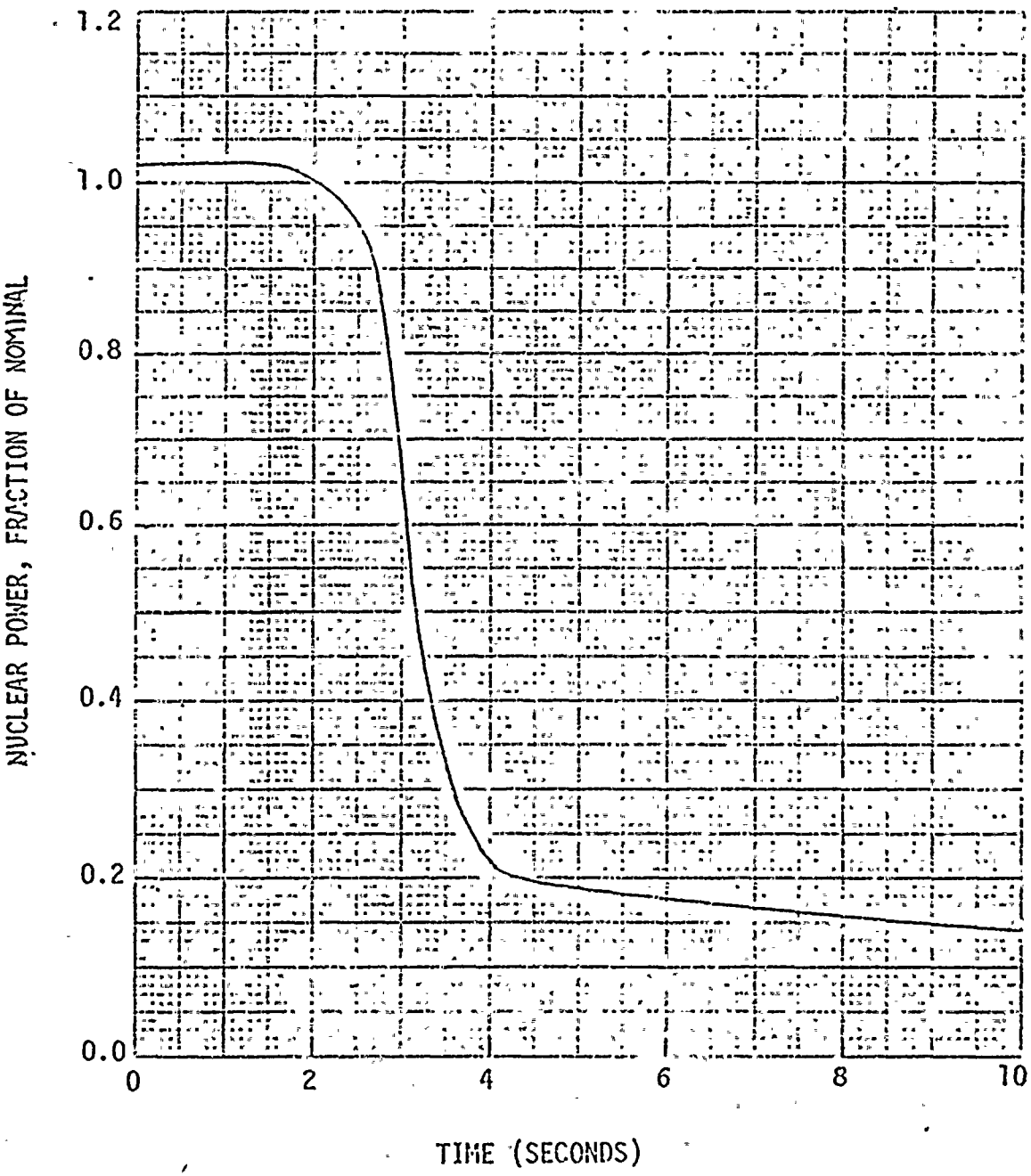


FIGURE 4

LOSS OF FLOW

HOT CHANNEL HEAT FLUX VS TIME

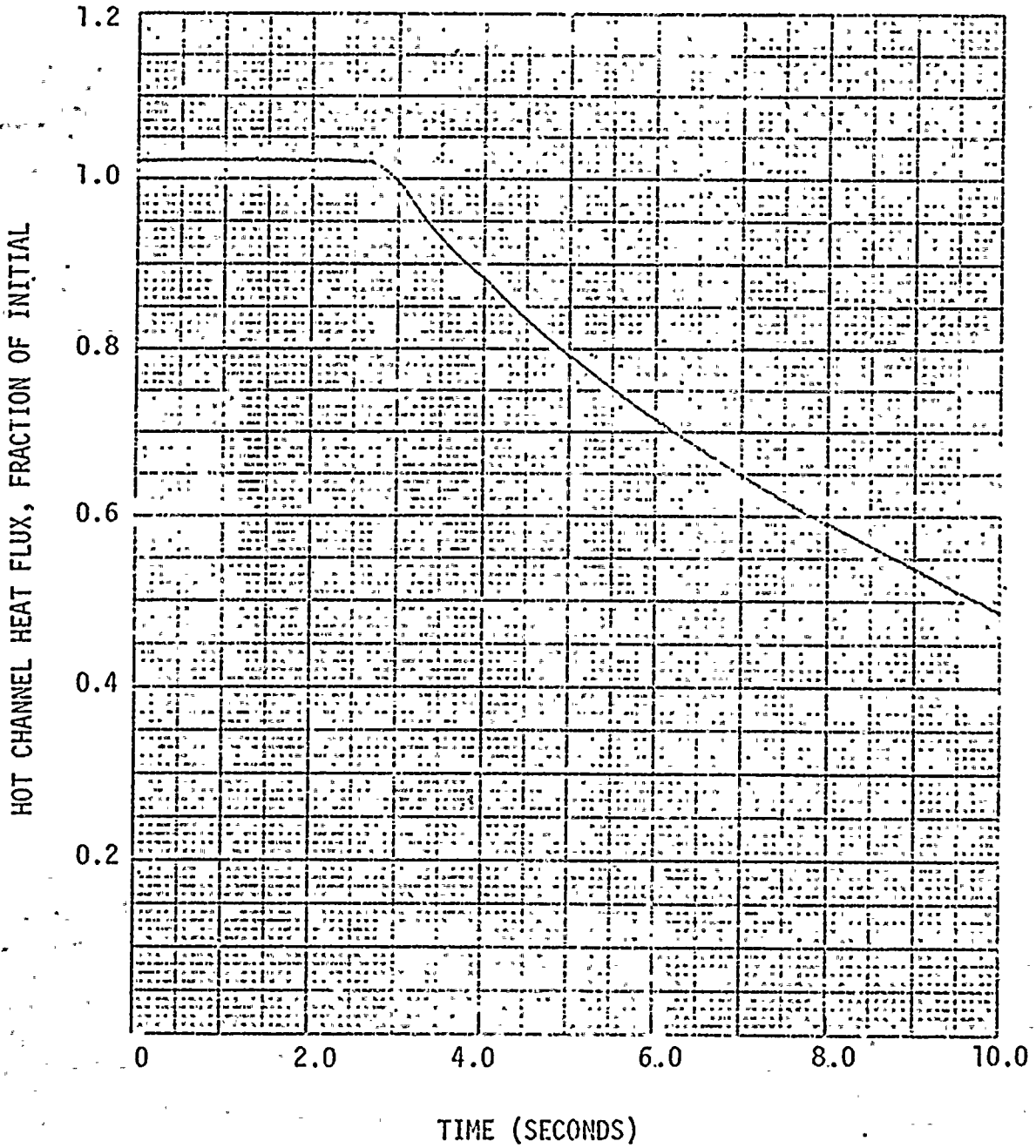
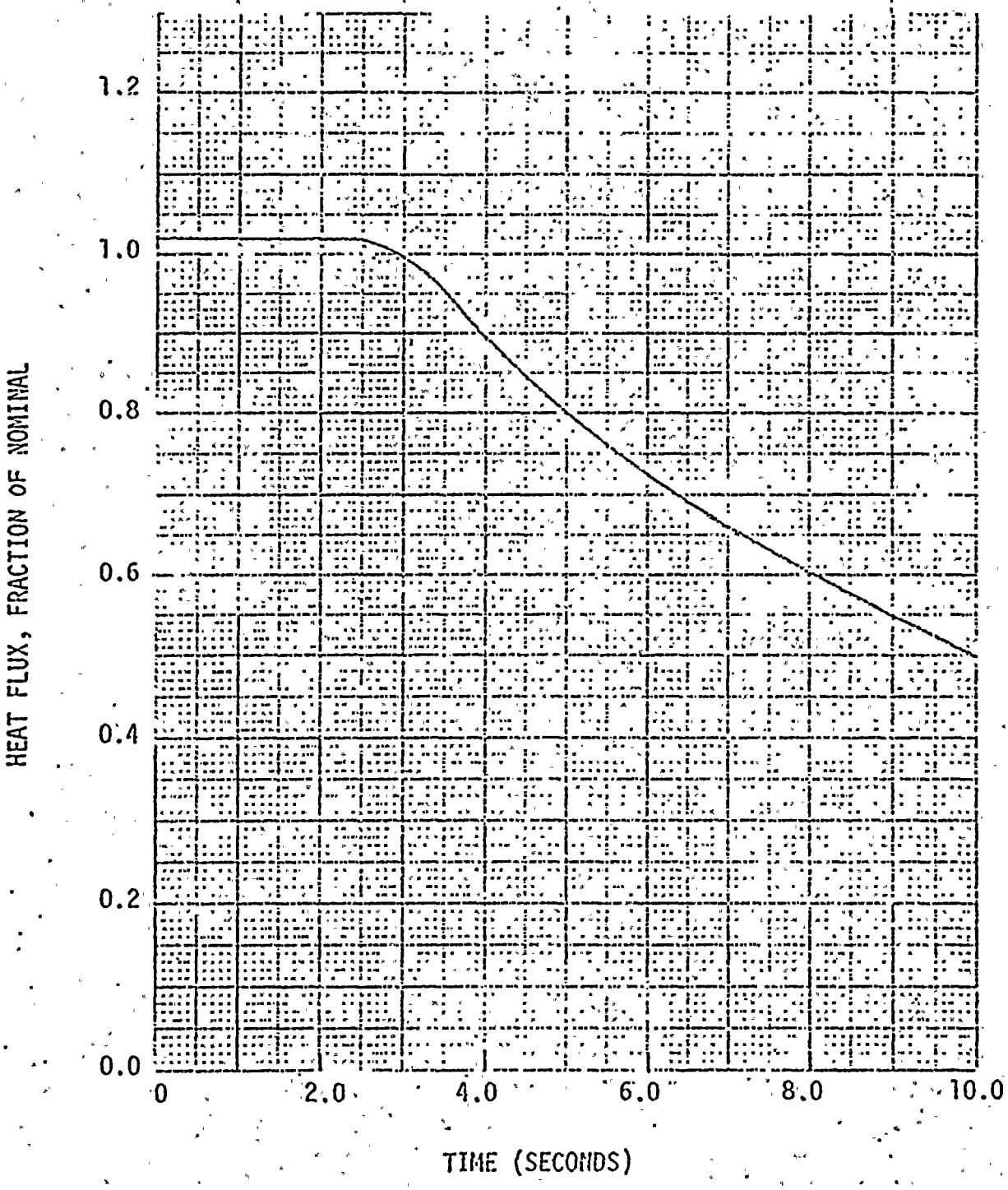


FIGURE 3

LOSS OF FLOW
HEAT FLUX VS TIME



LOSS OF FLOW
DNBR VS. TIME
(THIMBLE CELL)

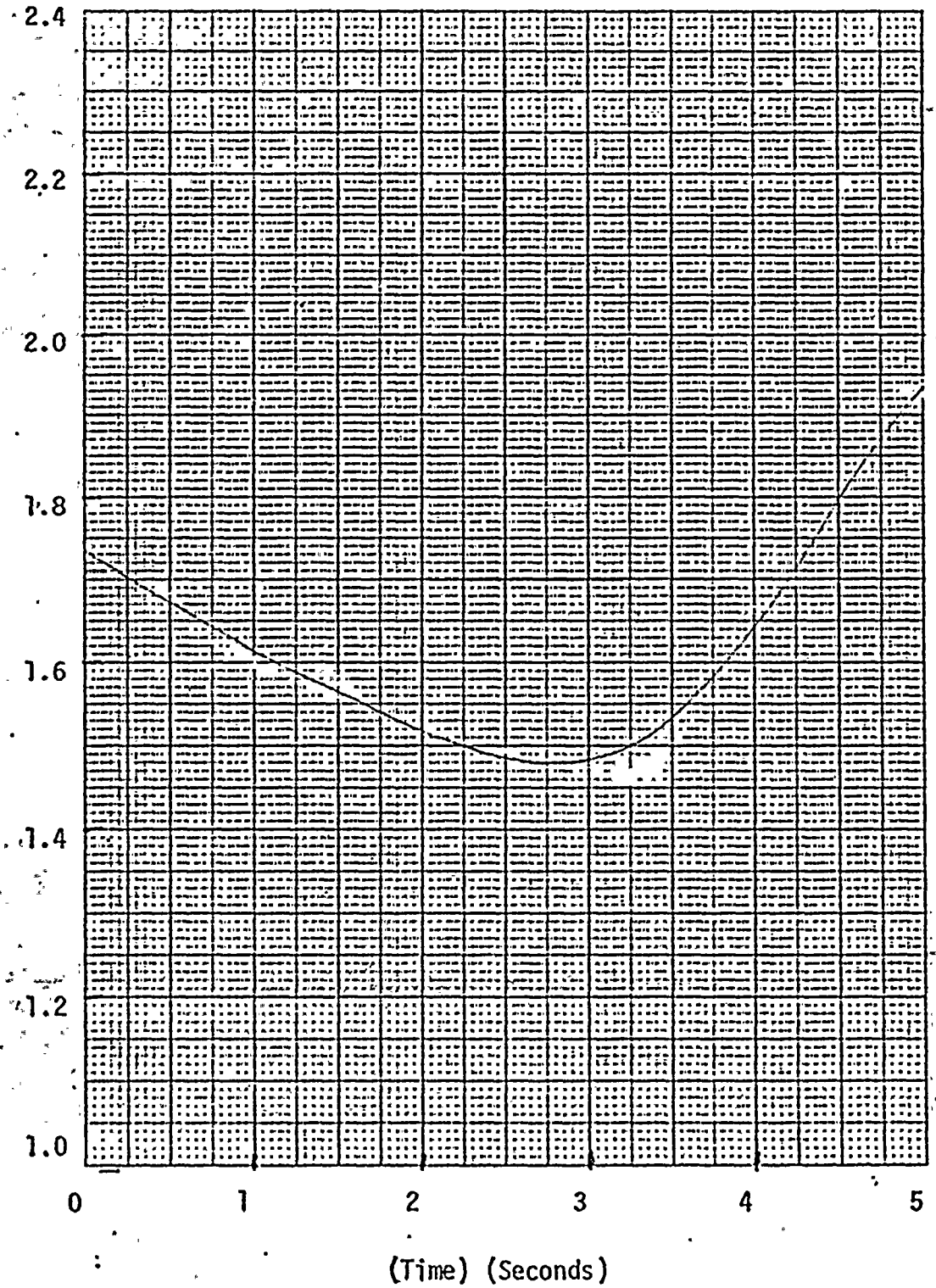


FIGURE 7

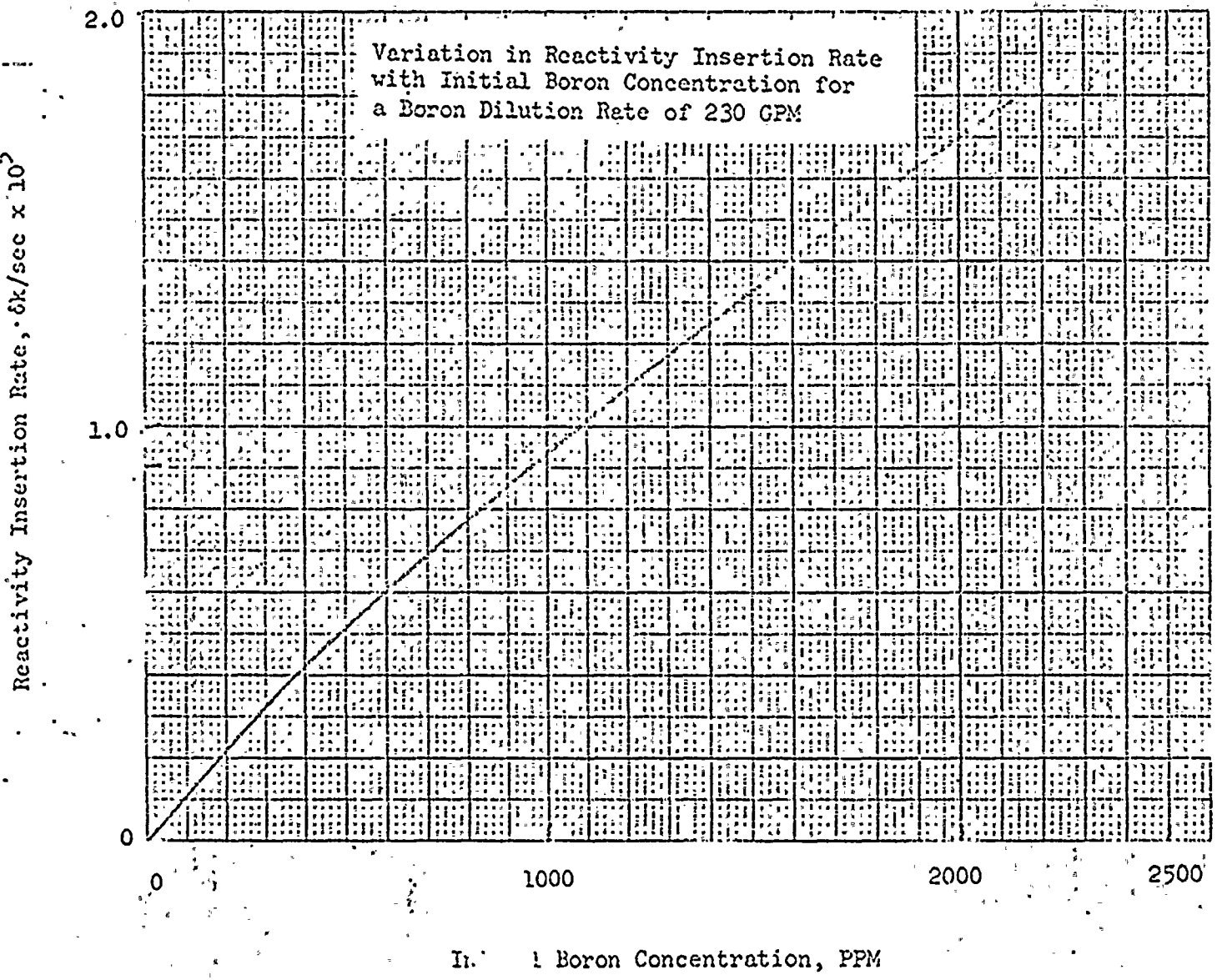


FIGURE 14.1.5

The standard design methods utilized in this 95% thermal design flow analysis of Cycle 5 of Turkey Point Unit 4 ensure that the generic DNB margins (Reference 1) exist in the Cycle 5 design and are available to offset DNBR reductions due to fuel rod bow. Utilizing these generic DNB margins to offset the current NRC rod bow penalties for low parasitic 15x15 fuel would result in the $F_{\Delta H}^{N}$ reductions vs. burnup curve of Reference 2 being applicable for Cycle 5.

These generic DNB margins are, however, more than enough to offset the rod bow penalties calculated in Reference 3 (partial rod bow test results) for burnups up to 33,000 MWD/MTU. For fuel regions with high burnups, the depletion of fissile nuclides and buildup of fission products greatly reduces power production capacity. These combined burnup effects reduce $F_{\Delta H}^{N}$ sufficiently to cover residual rod bow penalties beyond a region average burnup of 33,000 MWD/MTU.

REFERENCES

1. NRC's "Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors", Revision 1) Feb. 16, 1977.

2. W letter (J. Rolin) to FP & L (J. M. Cleveland) FP-FP-375 dated May 9, 1978.

3. W letter (C. Eicheldinger) to NRC (D. R. Ross, Jr.), NS-CE-1580, dated Oct. 24, 1977.

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

Robert E. Uhrig, being first duly sworn, deposes and says:

That he is a Vice President of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this said document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Robert E Uhrig
Robert E. Uhrig

Subscribed and sworn to before me this

20th day of July, 1978

Betty Brittain
NOTARY PUBLIC, in and for the County of Dade, State of Florida

My commission expires: NOTARY PUBLIC STATE OF FLORIDA at LARGE MY COMMISSION EXPIRES MARCH 27, 1982 BONDED THRU MAYNARD BONDING AGENCY

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