

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2170

June 28, 1974

Mr. Norman C. Moseley, Director
Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Region II - Suite 818
230 Peachtree Street, Northwest
Atlanta, Georgia 30303

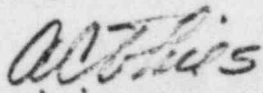
Re: RO:II:TNE
50-269/74-4

Dear Mr. Moseley:

Please find attached our responses to Items I.A.1.a and I.A.3.a and b
contained in RO Inspection Report 50-269/74-4.

Duke Power Company does not consider any information contained in
RO Inspection Report 50-269/74-4 to be proprietary.

Very truly yours,



A. C. Thies

ACT:vr
Attachment

8001310578

DUKE POWER COMPANY
OCONEE UNIT 1
RESPONSE TO AEC/RO REPORT NO. 50-269/74-4

I.A.1.a Asymmetric Rod Monitoring System

10CFR50.59 requires that proposed changes to the facility involving an un-reviewed safety question shall not be carried out unless authorized by the Atomic Energy Commission.

Contrary to the above, during the period from January 18, 1974 to April 26, 1974, Unit 1 was operated with as many as six asymmetric rod monitors for individual control rods turned off.

RESPONSE

In a February 28, 1974 letter to Mr. Angelo Giambusso, Deputy Director for Reactor Projects, Duke Power Company reported problems with the control rod drive absolute position indication (API) system at Oconee Nuclear Station. As identified in that report, most of the problems with the API have consisted of fluctuating or erratic indications. The effect of the erratic indications is to give an asymmetric rod signal to the integrated control system (ICS) which initiates a runback in power as long as the indicated fault exists provided the ICS is operated in the automatic mode. In order to avoid spurious runback due to erratic position indication signals, station personnel switched out of service the signal from the asymmetric rod monitoring bistable in the control rod drive system (CRD). Records indicate that six of these bistables were placed in the inoperative position.

Switching off the bistables had the following effect on the operation of the unit:

1. For those bistables switched off a runback in power would not have been initiated if those rods were asymmetric. However, in many cases, if a bypassed rod were dropped, runback would be initiated by other rods in the group due to the change in group average.
2. Rod withdrawal inhibit was bypassed on the particular rods for which bistables were switched off.
3. The statalarm window indicating "asymmetric rod" would have been inoperable for those rods which had the bistable switched off. However, position indication on the control rod drive PI panel would be available to the operator as well as an asymmetric alarm light on the control rod drive station. This light indicates a seven-inch asymmetric rod. It should be noted when the control rod drive station is in manual, the reactor runback and rod withdrawal inhibit features of the control rod drive and integrated control systems are blocked.

The effect of a dropped control rod, assuming no ICS or control rod drive system action, has been analyzed and reported in BAW-1387, "Oconee 1 Fuel Densification Report," January, 1973. Referring to Page 33, "The rod drop analysis results in a decrease in power initially after which the power returns to 100 percent. It has been shown previously that neither the withdrawal nor the drop of a single control element will cause perturbation of the flux shape sufficient to exceed design conditions at 112 percent. Therefore, this occurrence still does not present any thermal problems." AEC/Directorate of Licensing (DOL) in Supplement 3 to the Oconee Nuclear Station Unit 1 Safety Evaluation issued July 10, 1973 stated in Section 3.3 that the results of the control rod drop incident have been reviewed by the staff and this transient, "taking into account the effects of fuel densification concludes that they would not result in a reduction of core thermal margin, i.e., a DNB, less than 1.3." Consequently, although the words of Section 14.1.2.7 of the Final Safety Analysis Report have not been revised after the analysis was completed for fuel densification, the effect of a dropped rod with no credit being taken for the rod withdrawal inhibit or asymmetric rod runback features of the control rod drive and integrated control systems had been analyzed by B&W and confirmed by AEC/DOL in 1973. Also, Section 7.2.3.3 of the FSAR states the following:

Failure of the ICS does not diminish the safety of the reactor. None of the functions provided by the ICS are required for reactor protection or for actuation of the ESPS. The reactor protection criteria, used in the analysis of the accidents presented in Section 14, can be met irrespective of ICS action.

In May of 1974, B&W was requested by Duke Power Company to reconfirm by analysis that core protection limits are not exceeded assuming no control system action. The results of that analysis are as follows.

The maximum increase in peaking for a dropped control rod is 22.3 percent with an incore detector quadrant power tilt of 14.9 percent. Startup test analysis for Oconee 1 of the worst-case dropped rod observed an incore detector tilt of 15 percent and consistent peaking relationships. The analysis indicates that if a dropped rod occurred during normal operation the center fuel line melt and DNER criteria would not be exceeded. The following is a list of assumptions for the analysis:

1. The dropped rod is the worst case
2. Initial power level is 102 percent
3. Central fuel melt limit is 20.1 kW/ft
4. Fuel densification is included
5. Power peaking is at the highest level for normal equilibrium operation during life (BOL)
6. There is no initial quadrant tilt
7. After the dropped rod, the tilt increases to 15 percent (peaking increases by 22.4 percent)

With the preceding assumptions, the initial heat rate is 15.0 kW/ft. After the rod drop, the linear heat rate would increase to 18.4 kW/ft assuming no control reactivity compensation and a return to 102 percent power. If power

remains constant at 102 percent and the control rods compensate for the reactivity decrease, linear heat rate increases to 18.0 kW/ft. The minimum margin occurs in the case without reactivity compensation and is 9.2 percent. The loss in DNBR peaking margin for the tilt at 15 percent is 17.8 percent. For the same conditions as the preceding, the minimum DNBR peaking margin is 19.5 percent at 102 percent power leaving an excess margin of 1.4 percent.

The above analysis by B&W assumes no action by either the control systems or by the operator. However, Technical Specification 3.5.2.2e states the following:

If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60 percent of the thermal power allowable for the reactor coolant pump combination.

This, in effect, requires the operator to manually runback the plant to 60 percent power assuming no control system action. Appropriate action by the operator assures additional conservatism with regard to the above analysis.

The violation as issued alleged that an unreviewed safety question was involved. From 10CFR50.59, the definition of an unreviewed safety question is as follows:

A proposed change, test or experiment shall be deemed to involve an unreviewed safety question:

- (1) if the probability of occurrence or consequences of an accident or a malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or
- (2) if a possibility for an accident or malfunction of a different type than that evaluated previously in the safety analysis report may be created; or
- (3) if the margin of safety as defined in the bases of any technical specification is reduced.

In view of the above analysis concerning a dropped control rod without control rod drive or integrated control system action, Duke does not consider that the bypassing of the control rod drive asymmetric rod bistable monitors to be an unreviewed safety question.

Duke, however, is concerned that appropriate station management personnel were not advised and did not have an opportunity to review the bypassing of the bistables. Mr. J. E. Smith, Oconee Nuclear Station Superintendent, in a letter dated June 14, 1974 to all supervisors and engineers stated that the bypassing "of these functions was performed without written procedures and without proper review." Smith further writes, "while we do not consider the bypassing of these functions to be an unreviewed safety question worthy of a Category I violation, it does point out the potential

that a possibility exists in other areas for the bypassing of important functions which could result in a compromise of safety or the result in negating of important functions. In the future, such use of bypass switches is not to be permitted without proper review and a written approved procedure."

While Duke does not consider the action performed to be an unreviewed safety question, it is realized that the description in Section 14.1.2.7 of the Final Safety Analysis Report may not lead one to the proper conclusion concerning the required action of the control rod drive and integrated control systems: specifically, that core protection criteria are met without asymmetric rod runback or control rod drive inhibit. Therefore, a revision to Section 14.1.2.7 has been forwarded to AEC/DOL and is attached for information.

Duke is concerned that this matter was not identified to station management as a violation during the exit interview by the RO inspectors on April 26, 1974. Nor was Duke advised that RO considered this to be a violation prior to receipt of Inspection Report 50-269/74-4 dated May 28, 1974. We believe that appropriate identification of apparent violations at the earliest possible date will provide sufficient opportunity for discussion prior to written issuance of the apparent violation. We believe that a full discussion of this incident would not have resulted in its being identified as a Category I violation. Although Duke believes that proper procedural steps were not taken at the station, we respectfully request that the Category I nature of this violation be rescinded.

I.A.3.a Specific Conductance and pH Monitoring

As noted in the details section of the inspection report, there was one day during the period from December 1, 1973 to April 18, 1974 (and it has been subsequently verified from April 18, 1974 through June 1, 1974) when pH and specific conductance monitoring was not properly done. It has not been determined why this sample was not taken.

It is considered that appropriate action has already been taken to prevent recurrence of this violation. A status board has been prepared which gives the last date on which the sample was taken and the date on which the next sample is due. In addition, additional emphasis has been given by the station chemist to the regular gathering of these samples for analysis or the reporting of basin status if samples cannot be taken. Since monitoring has been done regularly each day since December 22, 1973, no further corrective action is anticipated.

I.A.3.b Waste Collection Basin Effluent pH

The proposed revision to Technical Specification 1.2B is under review by the Directorate of Licensing. Furthermore, work is in progress to enlarge the wastewater handling and collection facility at Oconee. Full compliance will require satisfactory completion of both of these items.

14.1.2.7 Stuck-Out, Stuck-In, or Dropped-In Control Rod Accident

14.1.2.7.1 Identification of Cause

In the event that a control rod cannot be moved, localized power peaking and subcritical margin must be considered.

If a control rod is dropped into the core while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution might be distorted due to the new control rod pattern. In the presence of a distorted power distribution, the return to full power might lead to localized power densities and heat fluxes in excess of design limitations.

14.1.2.7.2 Protective Basis

Adequate hot subcritical margin is provided by requiring a subcriticality of 1 percent $\Delta k/k$ with the control rod of greatest worth fully withdrawn from the core. The nuclear analysis reported in 3.2.2 demonstrates that this criterion can be satisfied. This criterion has been analyzed in terms of the minimum tripped rod worth available in the loss-of-coolant-flow, startup, rod withdrawal, and steam-line-failure accidents. In all cases, the available rod worth is sufficient to provide margins below any damage threshold.

For protective purposes a dropped control rod is defined as the deviation of a control rod from its group reference position by more than a maximum of 9 inches. This definition then covers both the action of dropping a rod and sticking a rod while moving a group. The action taken by the ICS is:

- a. All rod-out motion is inhibited.
- b. The steam generator load demand is run back to 60 percent of rated load at 5%/s.

The details of these actions are described in Section 7.2.2 and 7.2.3.

Although these ICS actions are available to mitigate the consequences of the accident, they are not required functions for safe plant operation as the results of the accident analysis demonstrate.

The criterion for plant protection during this transient is that the DNB ratio will not be less than 1.3 and the system pressure will not exceed code limits.

14.1.2.7.3 Method of Analysis

The transient response to a dropped control rod has been analyzed using a detailed B&W digital model. This program includes fuel pin, point kinetics, pressurizer, and loop models, including the steam generators.

The reactor is assumed to be operating at 100 percent of rated power when the control rod is dropped. In order to achieve the most adverse response the most negative values of moderator coefficient [3.0×10^{-4} ($\Delta k/k$)/F] and Doppler coefficient [-1.3×10^{-5} ($\Delta k/k$)/F] occurring at end-of-core life were

used. The maximum reactivity worths expected to occur during full operation were used to examine the effects of ICS protective action. These rod worths correspond to operation at full power without xenon (0.46% $\Delta k/k$) and with xenon (0.36% $\Delta k/k$). It was assumed that the steam generator load demand was reduced linearly to 60 percent at 5%/s. The effects of a dropped rod without ICS action were also examined for a very conservative rod worth of 0.65% $\Delta k/k$. The rod was assumed to drop to 2/3 insertion in 1.4 seconds.

14.1.2.7.2 Results of Analysis

The results of the analysis with ICS action are presented in Figures 14-18 and 14-19. Figure 14-18 shows the response to a 0.46% $\Delta k/k$ dropped rod. The neutron power decreases rapidly to about 55 percent of rated power. This causes rapid decreases in the core moderator temperature and fuel temperature. These temperature decreases over-compensate for the worth of the control rod, and the power rises until the reduced steam generator demand begins to increase the inlet temperature and decreases the power. The thermal power levels out briefly at about 78 percent of its initial value but soon begins to decrease in response to the decreased steam generator demand. The pressurizer pressure swing is about ± 60 psi before returning to equilibrium.

Figure 14-19 shows the results of the 0.36% $\Delta k/k$ rod drop. The initial neutron power decrease is slightly less in this case, resulting in the thermal power leveling off at 83 percent, a slightly higher value than in the 0.46% $\Delta k/k$ case. The pressurizer pressure peaks at a higher value due to this higher thermal power.

Figure 14-19a shows the results of a 0.65% $\Delta k/k$ dropped rod analysis conservatively based on no ICS action and operation at higher than rated power level of 2772 MWt. The neutron power decreases causing a rapid decrease in both the core moderator temperature and the fuel temperature. These temperature decreases overcompensate for the worth of the control rod, and the neutron power rises slightly above the initial neutron power level. The neutron power then decreases to below the initial power level and eventually levels out at the initial power level. The thermal power response is similar to the neutron power; however, the thermal power level never exceeds the initial power value. Both the core moderator temperature and pressurizer pressure decrease during the transient and level out at a value lower than the initial value.

Several cases have been run for rod drops at beginning of life conditions. These transients yield new power levels that are lower than the end of life conditions and may result in reactor trip. These are therefore not included in this discussion because they represent less severe conditions.

14.1.2.7.5 Conclusions

Control rod malfunctions are accommodated by the core design without ICS action. Since the most severe case analyzed for the dropped rod does not result in reactor trip nor does the thermal power exceed its initial value, core and reactor coolant system boundary protection is assured. Additional protection for the dropped rod accident is provided through the ICS which detects a dropped rod and inhibits out-motion of the control rods. The ICS

is designed to run back the steam generator load demand upon receiving the dropped rod signal from the rod drop detection circuitry. The reactor thermal power will assume a lower value that matches the load demand and will provide additional margin toward not exceeding any design limit.

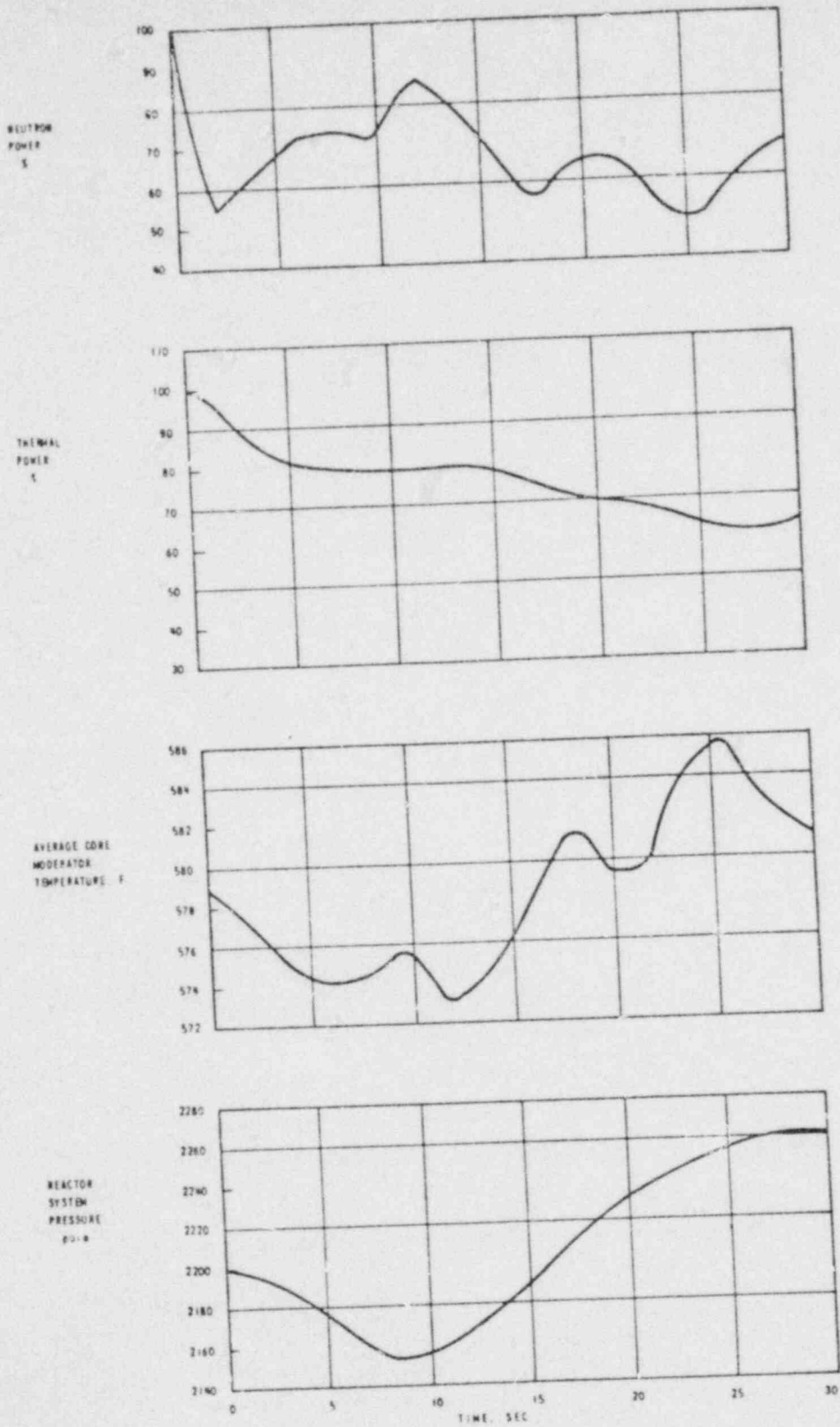
14.1.2.8 Loss of Electric Power

14.1.2.8.1 Identification of Cause

Each unit is designed to withstand the effects of a loss of electric load or electric power. Emergency power systems are described in 8.2.3. Two types of power losses are considered:

- a. A loss of load condition, caused by separation of the unit from the transmission system.
- b. A hypothetical condition which results in a complete loss of all system and station power.

The reactor protection criteria for those conditions are that fuel damage will not occur from an excessive power-to-flow ratio nor will the reactor coolant system pressure exceed design pressure.



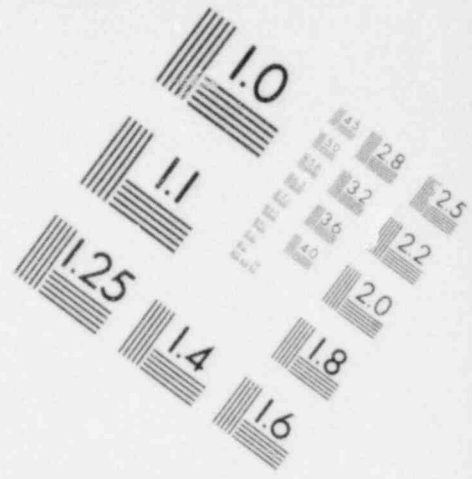
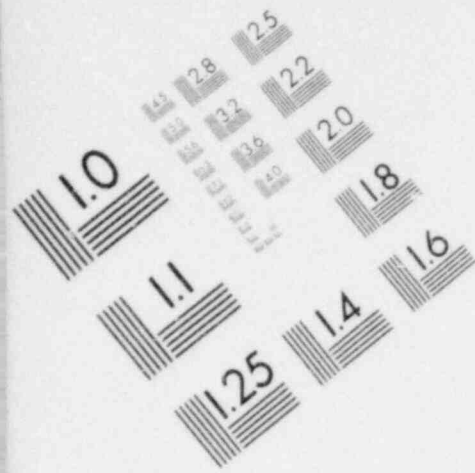
0.46' DR/N ROD DUG FROM RATED POWER
 WITH AUTOMATIC RUNBACK TO 60% DEMAND
 IN 12 SECONDS

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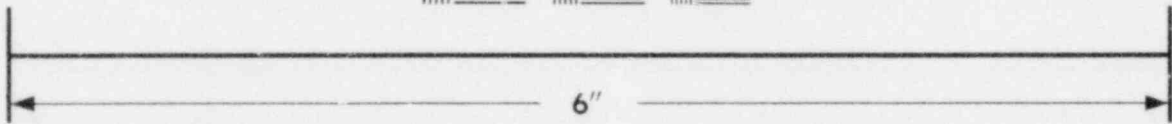


OCONEE NUCLEAR STATION

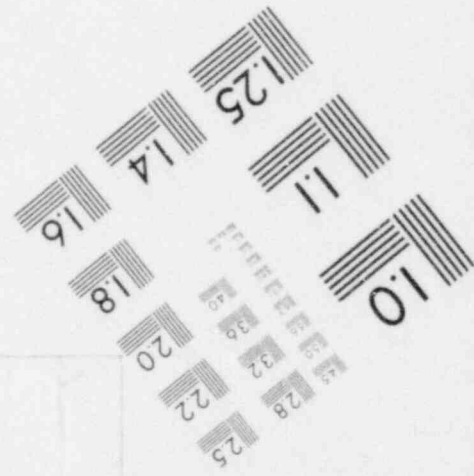
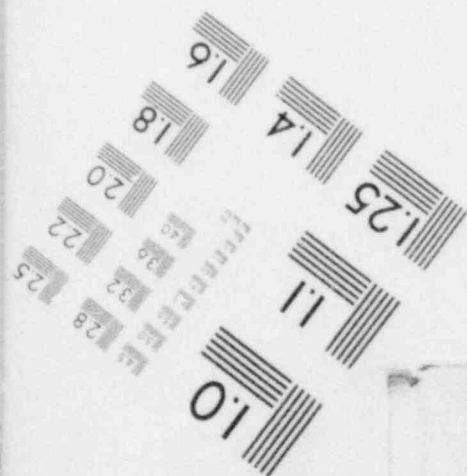
Figure 14-18

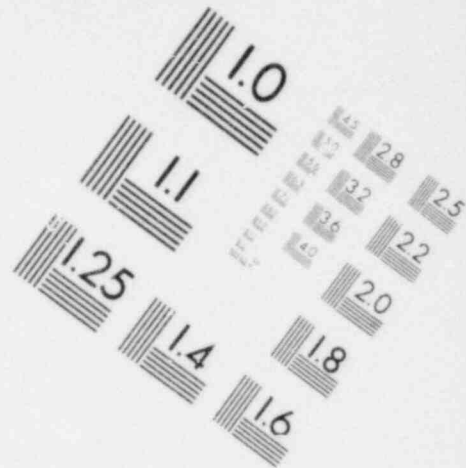
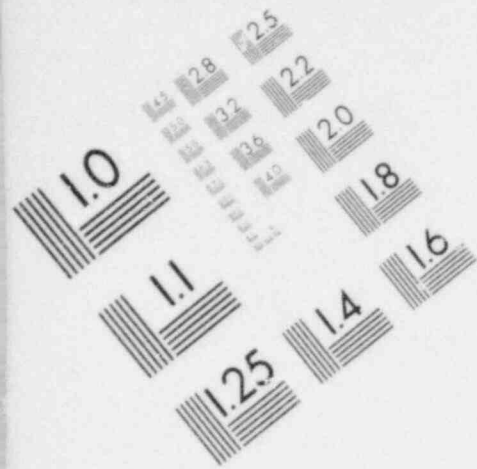


**IMAGE EVALUATION
TEST TARGET (MT-3)**

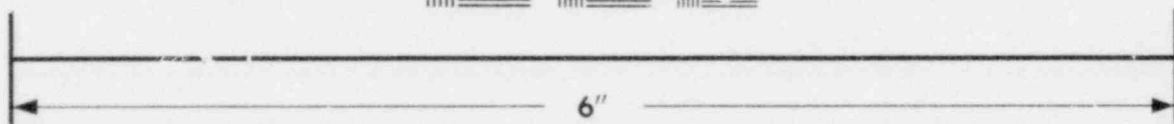
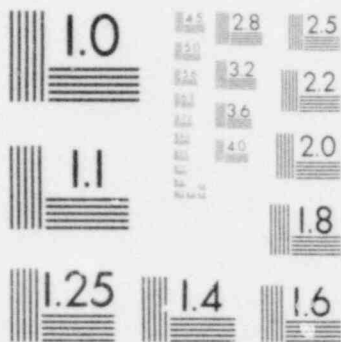


MICROCOPY RESOLUTION TEST CHART

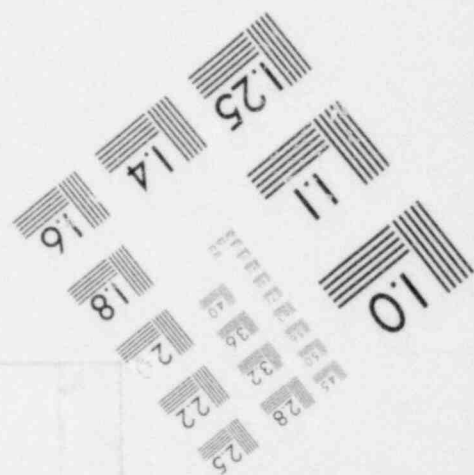
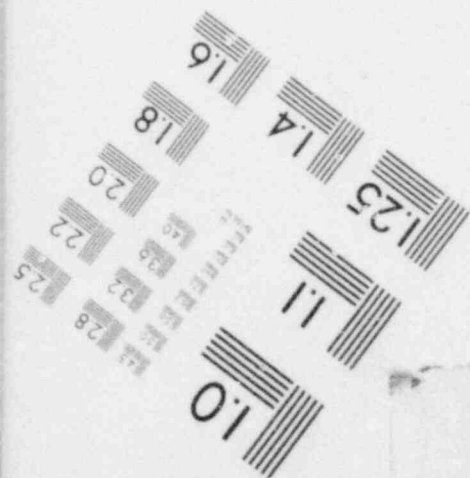




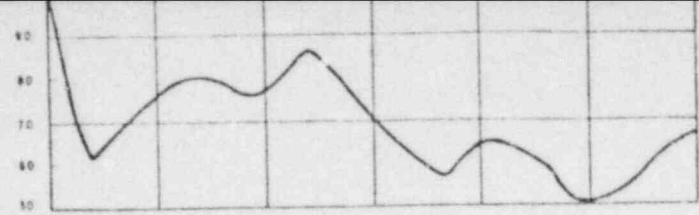
**IMAGE EVALUATION
TEST TARGET (MT-3)**



MICROCOPY RESOLUTION TEST CHART



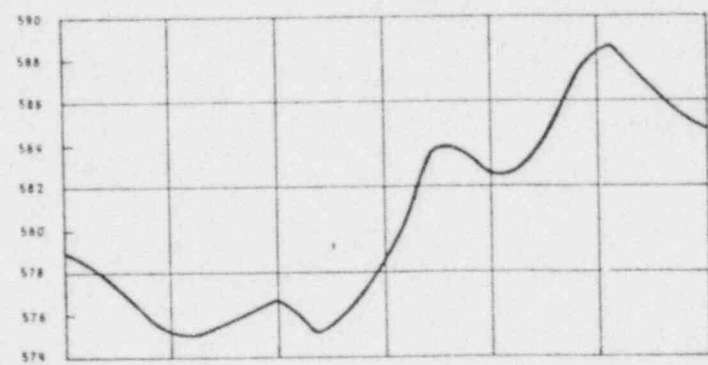
NETURON
POWER
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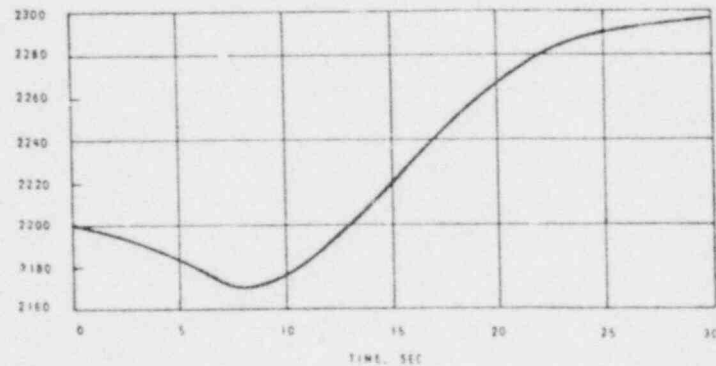
THERMAL
POWER
%



AVERAGE CORE
MODERATOR
TEMPERATURE, F



REACTOR
SYSTEM
PRESSURE
PSIA



0.365 AX/M ROD DROP FROM RATED
POWER WITH AUTOMATIC PWRBACK TO
60% DEMAND IN 12 SECONDS

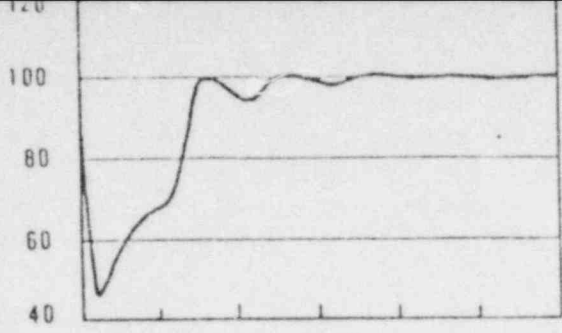


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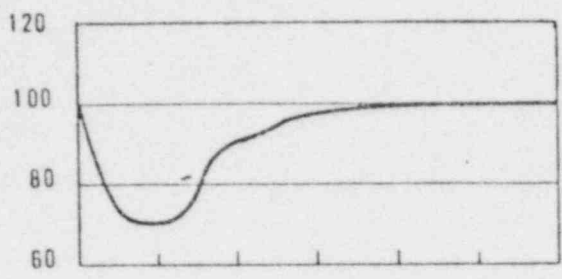
Figure 14-19

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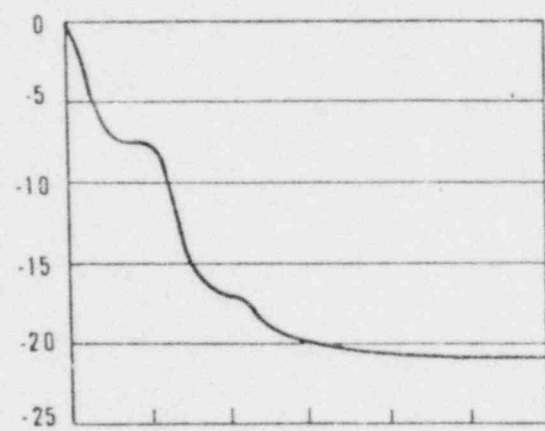
Neutron Power, %



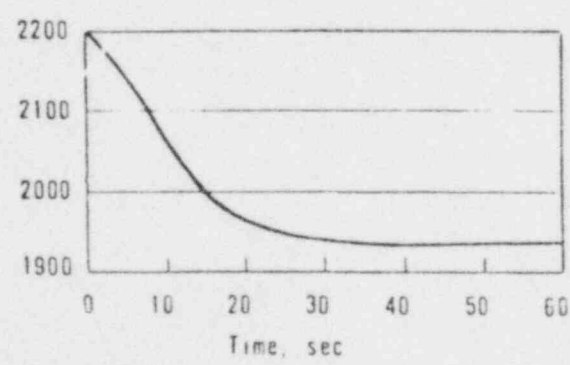
Thermal Power, %



Average Core Moderator Temperature Change, F



Reactor System Pressure, psia



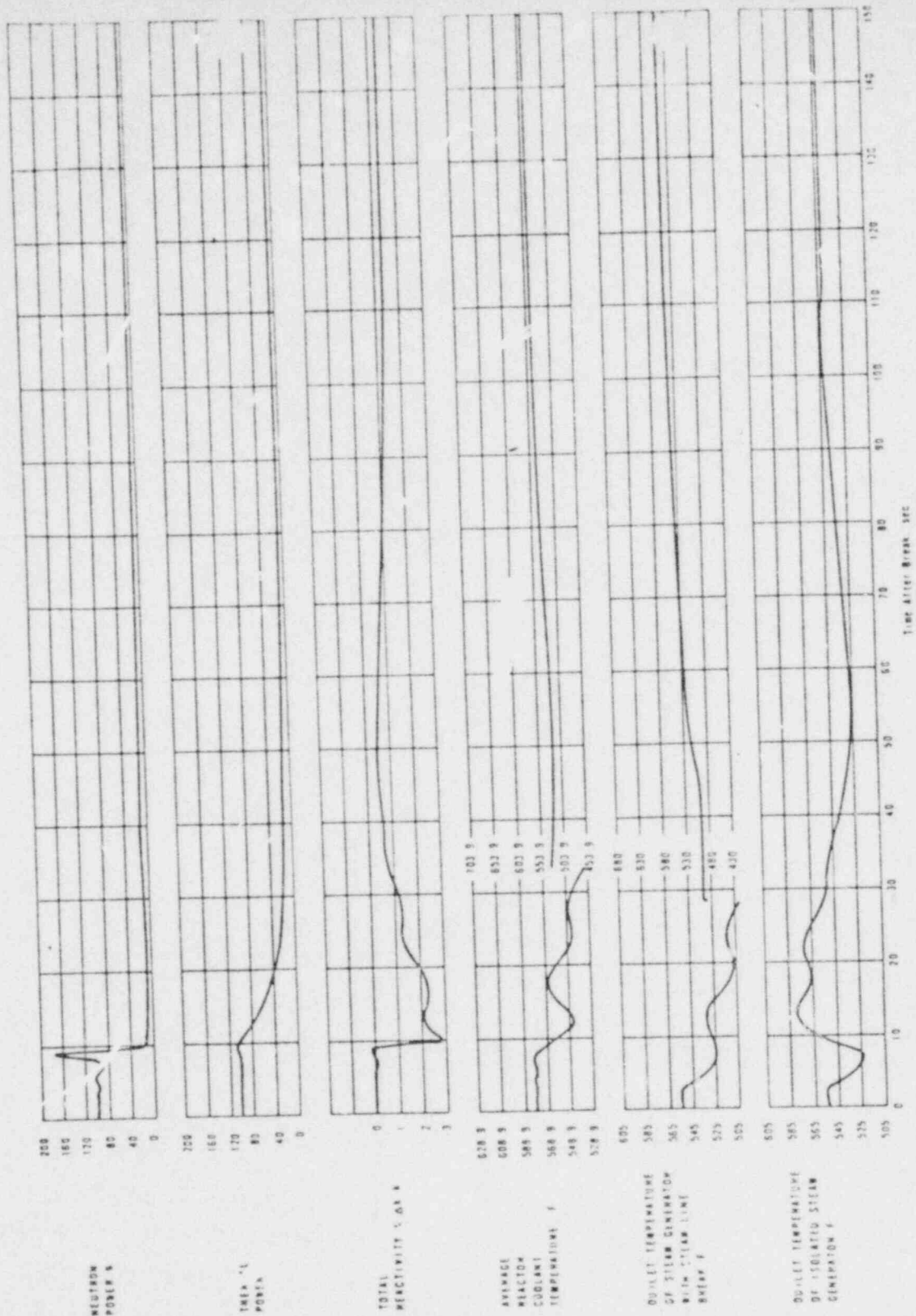
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0.65% Ak/k ROD DROP FROM RATED POWER AT EOL CONDITIONS (2772 MWT)

OCONEE NUCLEAR STATION

Figure 14-19a



DOUBLE END OF RUPTURE OF 34 IN. STEAM LINE BETWEEN STEAM GENERATOR AND STOP VALVE (WITH RECOOLANT ISOLATION)



OCCONEE NUCLEAR STATION

Figure 14 - 20

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