Docket Nos. 50-250
50-251

December 17, 1973

Florida Power & Light Company
ATTN: Dr. Robert E. Uhrig
Director of Nuclear Affairs
P. O. Box 3100
Miami, Florida

Change No. 11
License Nos. DPR-31 and 41

Gentlemen:

By letter dated October 12, 1973, you proposed revisions to the Technical Specifications attached as Appendix A to Facility Operating Licenses DPR-31 and 41. This action is designated Change No. 11.

The requested change allows operation of Turkey Point Units 3 and 4 at power levels above 93% of rated power and, in order to achieve a longer first fuel cycle, at reduced reactor coolant system pressure. In support of this, you submitted a report on July 6, 1973, entitled "Fuel Densification, Turkey Point Units 3 and 4, Low Pressure Analysis."

We have reviewed your report and have determined that the effects of low pressure operation on DNB and the cladding creep rate have been appropriately considered and that a suitable schedule for increasing the maximum allowable power level as a function of burnup has been established. Our Safety Evaluation is enclosed as Enclosure 1.

Also, we have reviewed your request, submitted by letter dated July 12, 1973, to change the undervoltage trip setpoint from 70% to 60% of normal voltage. On the basis that the reactor coolant flow is 100% of rated when the pump motor voltage is 60% of nominal, we approve reducing the trip setpoint to 60%.

Changes to the limiting safety system settings have been made in Section 2.3 to reduce the low pressurizer pressure trip setting, and to revise the overtemperature and overpower ΔT trip settings in accordance with revised Figure 2.1-1. Changes to Section 3.2, Control Rod and Power Distribution Limits, have been made to paragraph 6, Power Distribution Limits, to define the allowable operating power and required power distribution measurements in accordance with Figure 3.2-4.

OFFICE➤						
SURNAME➤						
DATE➤						

We conclude that the changes do not involve a significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating Licenses DPR-31 and 41 are hereby changed as described in the Florida Power & Light Company letter of October 12, 1973, and as set forth in the revised pages which are enclosed.

The revised Technical Specifications as specified herein covering operation of Units 3 and 4 at a lower primary system pressure shall become effective for each unit at the discretion of the Florida Power & Light Company, however, the implementation for each unit cannot proceed until such time as the unit is withdrawn to a subcritical condition and the protection system instrumentation is reset in accordance with the revised Technical Specifications; until such time the current Technical Specifications shall apply and after such time the current Technical Specifications shall become null and void. This authorization contemplates a period when the operation of Units 3 and 4 will be governed by different Technical Specifications.

Sincerely,

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 2-1
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages

cc:
Mr. Jack Newman
Newman, Reis, and Axelrad
1100 Connecticut Avenue, NW
Washington, D. C. 20036

DISTRIBUTION:
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DATE	12/17/73	12/17/73	12/17/73	12/17/73		

Docket

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 17, 1973

Docket Nos. 50-250
50-251

Florida Power & Light Company
ATTN: Dr. Robert E. Uhrig
Director of Nuclear Affairs
P. O. Box 3100
Miami, Florida

Change No. 11
License Nos. DPR-31 and 41

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The requested change allows operation of Turkey Point Units 3 and 4 at power levels above 93% of rated power and, in order to achieve a longer first fuel cycle, at reduced reactor coolant system pressure. In support of this, you submitted a report on July 6, 1973, entitled "Fuel Densification, Turkey Point Units 3 and 4, Low Pressure Analysis."

We have reviewed your report and have determined that the effects of low pressure operation on DNB and the cladding creep rate have been appropriately considered and that a suitable schedule for increasing the maximum allowable power level as a function of burnup has been established. Our Safety Evaluation is enclosed as Enclosure 1.

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Changes to the limiting safety system settings have been made in Section 2.3 to reduce the low pressurizer pressure trip setting, and to revise the overtemperature and overpower ΔT trip settings in accordance with revised Figure 2.1-1. Changes to Section 3.2, Control Rod and Power Distribution Limits, have been made to paragraph 6, Power Distribution Limits, to define the allowable operating power and required power distribution measurements in accordance with Figure 3.2-4.

We conclude that the changes do not involve a significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered. Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications of Facility Operating Licenses DPR-31 and 41 are hereby changed as described in the Florida Power & Light Company letter of October 12, 1973, and as set forth in the revised pages which are enclosed.

The revised Technical Specifications as specified herein covering operation of Units 3 and 4 at a lower primary system pressure shall become effective for each unit at the discretion of the Florida Power & Light Company, however, the implementation for each unit cannot proceed until such time as the unit is withdrawn to a subcritical condition and the protection system instrumentation is reset in accordance with the revised Technical Specifications; until such time the current Technical Specifications shall apply and after such time the current Technical Specifications shall become null and void. This authorization contemplates a period when the operation of Units 3 and 4 will be governed by different Technical Specifications.

Sincerely,

R C DeYoung

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 2-1
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages

cc:
Mr. Jack Newman
Newman, Reis, and Axelrad
1100 Connecticut Avenue, NW
Washington, D. C. 20036

ENCLOSURE 1

SAFETY EVALUATION FOR OPERATION OF TURKEY POINT UNITS 3 & 4

AT POWER LEVELS UP TO 100% OF RATED AND AT REDUCED PRIMARY SYSTEM PRESSURE

We have reviewed the report "Fuel Densification - Turkey Point Units 3 and 4 - Low Pressure Analysis, "WCAP-8136, dated June 1973. This report was submitted by Florida Power & Light in support of proposed changes to the Technical Specifications which would permit (1) power level escalation with time (burnup) to 100% of rated power, and (2) operation at a reduced primary system pressure, 1900 psia.

We have determined that three areas require assessment: minimum DNB ratio, stored energy, and time-dependent creep collapse of fuel cladding.

The DNB ratio analysis was performed using the methods described in the FSAR. The minimum value of the DNB ratio during normal operation and anticipated transients is limited to a value of 1.30. A reduction in the core inlet temperature of the reactor coolant from 546.2°F to 538.6° F is required for the reduction in reactor coolant system pressure from 2250 psia to 1900 psia in order that the calculated DNB ratio is above 1.30 for power operation up to 100% of rated power, including the anticipated transients. We find this acceptable and conclude that the DNB margins are similar to those at 2250 psia because the reactor coolant temperatures have been reduced appropriately.

For determining the stored energy, the creep rate associated with the reduced pressure of 1900 psia was used. This provides less fuel-to-cladding gap closure and increases the stored energy and is, therefore, conservative and acceptable.

We have reviewed and accepted the Westinghouse time-dependent creep-collapse model in connection with our earlier review of fuel densification for VEPCO's Surry plant (Docket Nos. 50-280/281). The schedule for increasing the maximum allowable power levels for Turkey Point Units 3 and 4 as a function of burnup has been developed with this model. Reducing the primary system pressure will reduce the creep rate with the direct result that the time to clad collapse will be extended.

In summary, we have determined that the effects of fuel densification and lower primary system operation have been adequately analyzed. We conclude that the changes to the limiting safety system settings embodied in the proposed Technical Specification are appropriate and will assure that the consequences of reactor transients or postulated accidents are not significantly different from those previously judged acceptable by us.

On the basis of our review we have determined that the proposed change does not involve a significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered.

Paul S. Check

Paul S. Check, Senior Project Manager
Light Water Reactors Project Branch 2-2
Directorate of Licensing

Karl Kniel

Karl Kniel, Chief
Light Water Reactors Project Branch 2-2
Directorate of Licensing

Date: DEC 1 7 1973

P.O. BOX 3100 MIAMI, FLORIDA 33101

FLORIDA POWER & LIGHT COMPANY

October 12, 1973

RECEIVED
OCT 16 1973
U.S. Atomic Energy Commission
Regulatory Mail Section

Mr. R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Re: Turkey Point Plant Units 3 and 4
Docket Nos. 50-250 and 50-251

Dear Mr. DeYoung:

In accordance with 10 CFR 50.59, Florida Power & Light Company (FPL) submits herewith three signed originals and 19 additional copies of a request for authorization of a change in Technical Specifications attached as Appendix A to Facility Operating Licenses DPR-31 and 41.

The proposed changes are to escalate power above 93% of rated power and to operate at reduced reactor coolant system pressure.

The changes are as set forth in the attached revised Technical Specification pages and figures bearing this date in the lower right hand corner, and are as described below:

Page v

The titles of replacement Figures 2.1-1 and 2.1-2 have been entered.

Figure 3.2-3 (new) has been added to the list.

Figure 3.2-3 (Power Spike Factor versus Elevation) added in Change No. 3 should be removed from the specifications.

Figure 3.2-4 (new) has been added to the list.

Figure 4.12-1 has been added to the list in accordance with Change No. 8.

Page 1-6

In 1.15 the second sentence has been changed and the third sentence added.

Figures 2.1-1 and 2.1-2

These are replacement figures.

Page 2.3-2

The Overtemperature ΔT formula constants have been modified.

An "F" has been added at the end of the ΔT_o definition.

"Rated" has been substituted for "interim" in the second paragraph of the f (ΔT) defnintion.

K_1 for three loop operation has been changed (at the botton of the page.)

Page 2.3-3 (retyped)

In the Overpower ΔT formula "1.11" has been changed to "1.09".

K_2 has been changed from "0.00068" to "0.00134".

The low pressurizer pressure value has been changed to 1715 psig.

Note that the reactor coolant pump motor undervoltage trip value has been left at 60% as proposed in the FPL submittal of July 12, 1973.

Page 3.2-2 (retyped)

The word "rated" has been deleted in the fourth line of 2.

Page 3.2-3 (retyped)

No changes have been made.

Pages 3.2-4 and 3.2-5 (retyped)

Paragraph c. has been rewritten. Minor editing has been done to paragraph f. (formerly on page 3.2-5). Paragraph g. has been rewritten.

Page 3.2-6 (retyped)

No change has been made to the text.
NOTE: The retyping of the pages listed above eliminated old page 3.2-7.

Figures 3.2-3 and 3.2-4

These are new figures.
NOTE: Figure 3.2-3 (Power Spike Factor) should be removed from the specification.

Page B2.3-1

In the second line of the last paragraph 112% has been changed to 118%. A 9% instrument error is used to arrive at this value.

Pages B3.2-3 through B3.2-6 (retyped)

The Bases have been made consistent with the proposed specifications.

When the Technical Specifications are changed, it is requested that a reasonable period of time (no less than 30 days) be allowed FPL to put them into effect. To date, from the time a Change has become effective, until FPL has received it, five to 17 days have elasped. Technically FPL is in violation of license conditions during this period. Further, the subject changes will involve time consuming recalibration of instruments and controls, and FPL wishes to make these in a deliberate unhurried manner.

Very truly yours,

Robert E Uhrig

Robert E. Uhrig
Director of Nuclear Affairs

REU:nch
Enclosures

cc: Mr. Jack R. Newman

APPROVED:

E A Adomat

E. A. Adomat
Executive Vice President

STATE OF FLORIDA)
)   SS
COUNTY OF DADE   )

E. A. ADOMAT, being first duly sworn, deposes and says:

That he is an Executive Vice PResident of Florida Power & Light Company, the Licensee herein:

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

E. A. Adomat

E. A. Adomat

Subscribed and sworn to before me

this 12th day of October, 1973.

Margaret C. Hart

Notary Public in and for the County
of Dade, State of Florida

[illegible] PUBLIC STATE OF FLORIDA AT LARGE
[illegible]SION EXPIRES FEB. [illegible] 1977

My Commission expires [illegible] GENERAL INSURANCE UNDERWRITERS

LIST OF FIGURES

1.15 INTERIM LIMITS

Limitations are imposed upon reactor core power and power distribution beyond previously established design bases consistent with interim bases for core cooling analysis established by the AEC in 1971 and bases for the effects of densification established in November 1972. Interim power of the reactor core is limited to the values determined in accordance with specification 3.2. Interim power in MWt equals N x 2200, where N is determined in accordance with Section 6.c. of specification 3.2. The fuel residence time for cycle 1 shall be limited to 10,000 effective full power hours (EFPH) under design operating conditions.

1.16 LOW POWER PHYSICS TESTS

Low power physics tests are tests below a nominal 5% of rated power which measure fundamental characteristics of the reactor core and related instrumentation.

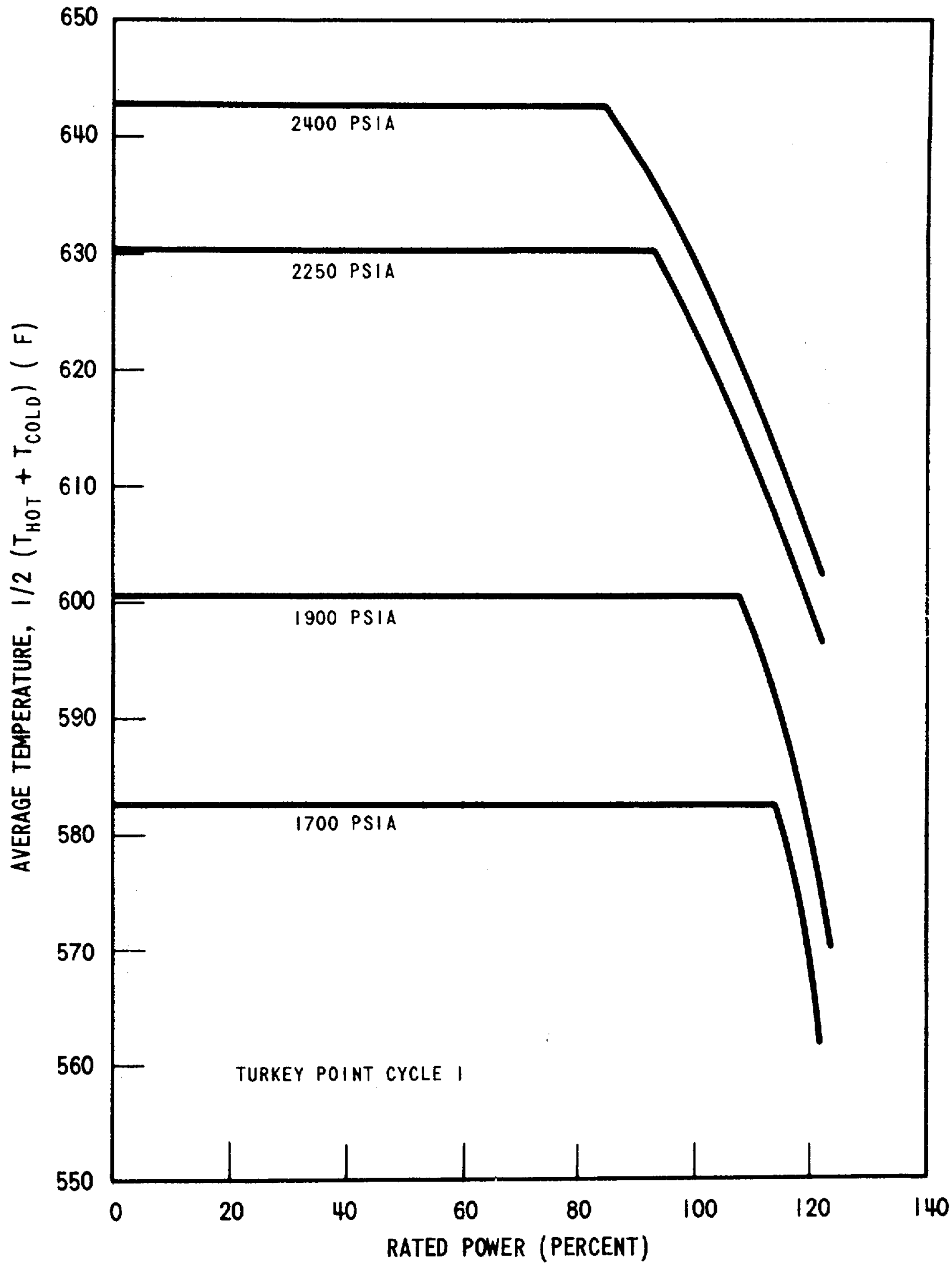

Figure 2.1-1. Reactor Core Thermal and Hydraulic Safety Limits, Three Loop Operation,

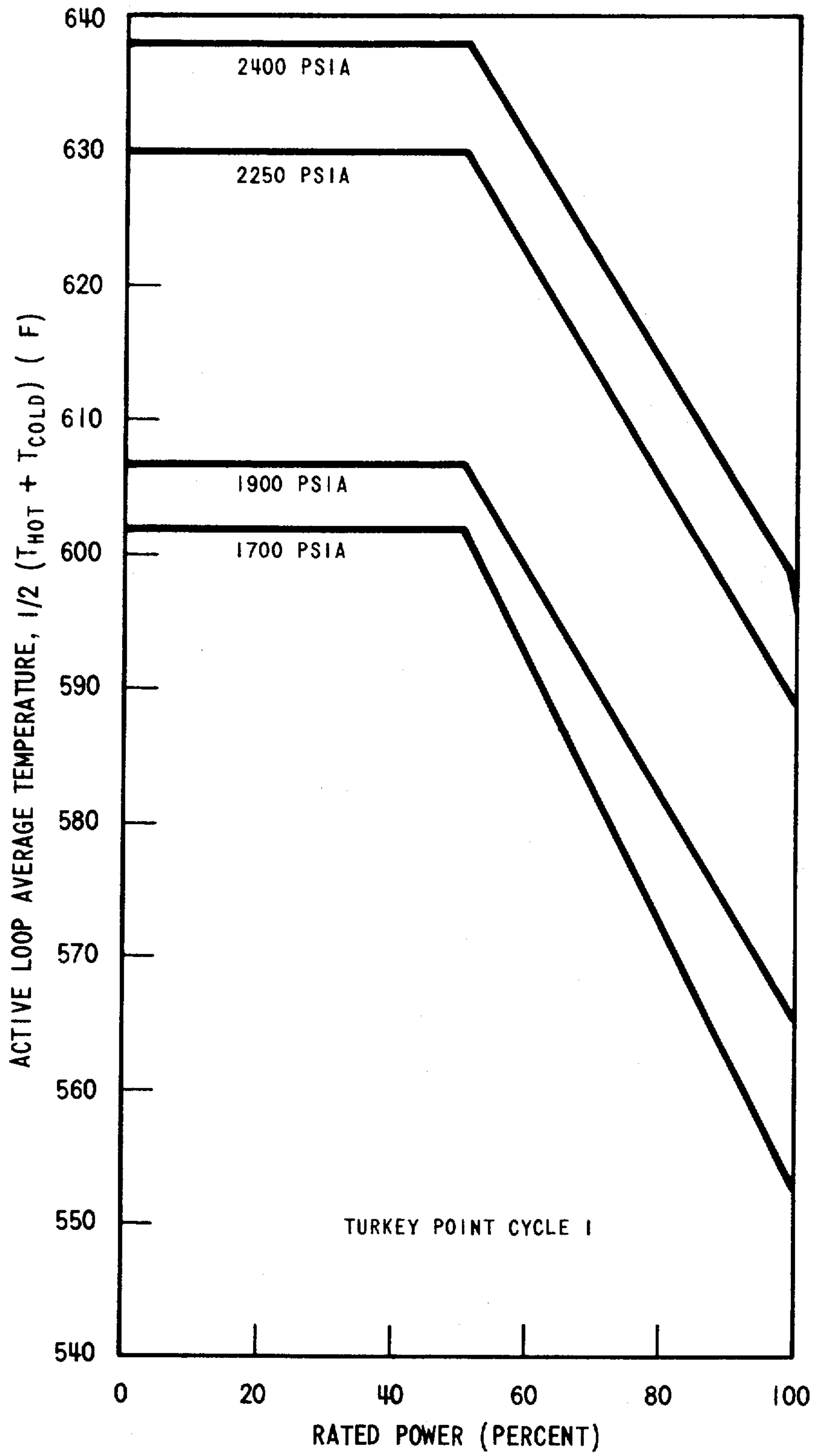

Figure 2. 1-2. Reactor Core Thermal and Hydraulic Safety Limits, Two Loop Operation

Reactor Coolant Temperature

Overtemperature ΔT $\leq \Delta T_o \left[K_1 - 0.0174(T-566.6) + 0.000976(P-1885) - f(\Delta I) \right]$

ΔT_o = Indicated ΔT at rated power, F

T = Average temperature, F

P = Pressurizer pressure, psig

$f(\Delta I)$ = 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, $f(\Delta) = 0$.

For each percent that the magnitude of $(q_t - q_b)$ exceeds +10 percent, the Delta-T trip set point 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 set point shall be automatically reduced by 2 percent of its value at interim power.

K_1 (Three Loop Operation) = 1.120;
(Two Loop Operation) = 0.88

Overpower $\Delta T \leq \Delta T_o \left[1.09 - K_1 \frac{dT}{dt} - K_2 (T - T') - f(\Delta I) \right]$

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 = 0.00134 for T equal to or more than T'; 0 for T less than T'

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

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

Pressurizer

Low Pressurizer pressure - equal to or greater than 1715 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

f. Except for low power physics tests, the shutdown margin with allowance for a stuck control rod shall exceed the applicable value shown on Figure 3.2-2 under all steady-state operating conditions from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions (540 F) if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron concentration or part-length rod position.

g. During physics tests and control rod exercises, the insertion limits need not be met, but the required shutdown margin, Figure 3.2-2 must be maintained or exceeded.

2. MISALIGNED CONTROL ROD

If a part length or full length control rod is more than 15 inches out of alignment with its bank, and is not corrected within 8 hours power shall be reduced so as not to exceed 75% of interim power for 3 loop or 45% of interim power for two loop operation, unless the hot channel factors are shown to be no greater than allowed by Section 6a of Specification 3.2.

3. ROD DROP TIME

The drop time of each control rod shall be no greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

4. INOPERABLE CONTROL RODS

a. No more than one inoperable control rod shall be permitted during sustained power operation, except it shall not be permitted if the rod has a potential reactivity insertion upon ejection greater than 0.3% Δ k/k at rated power. Inoperable rod worth shall be determined within 4 weeks.

b. A control rod shall be considered inoperable if

(a) the rod cannot be moved by the CRDM, or

(b) the rod is misaligned from its bank by more than 15 inches, or

(c) the rod drop time is not met.

c. If a control rod cannot be moved by the drive mechanism, shutdown margin shall be increased by boron addition to compensate for the withdrawn worth of the inoperable rod.

5. CONTROL ROD POSITION INDICATION

If either the power range channel deviation alarm or the rod deviation monitor alarm are not operable rod positions shall be logged once per shift and after a load change greater than 10% of rated power. If both alarms are inoperable for two hours or more, the nuclear overpower trip shall be reset to 93% of interim power.

6. POWER DISTRIBUTION LIMITS

a. At all times the hot channel factors defined in the basis must meet the following limits:

$F_q^N \leq 2.50 [1 + 0.2 (1-P)]$ in the flux difference range +10 to -14 percent

$F_{\Delta H}^N \leq 1.55 [1 + 0.2 (1-P)]$

where P is the fraction of interim power at which the core is operating

$(P \leq 1.0)$

b. If peaking factors exceed the limits of Section 6a, the reactor power and high neutrol flux trip setpoint shall be reduced by 1 percent for every percent excess over $F_{\Delta H}^N$ or F_q^N, whichever is limiting. If the peaking factors cannot be corrected within 1 day, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

c. The permissible fraction of rated power, N, not to exceed the power levels given in Figure 3.2-3, at which the reactor can be operated shall be determined by

$$N = \frac{Q}{5.56 \times 1.02 \times 1.019 \times 1.007 \times M}$$

where $M = 2.58 \times \frac{F_{xy}}{1.435} [1 + 2 (T/100 - 0.02)]$;

Q = limiting local power from Figure 3.2-4;

F_{xy} is 1.435, or the value of the unrodded horizontal plane peaking factor appropriate to F_q as determined by a movable in-core detector map taken on at least a monthly basis; and

T is the percentage operating quadrant tilt limit, having a value of 2% if F_{xy} is 1.435 or a value up to 10% as selected by the operator if a measured F_{xy} value is used.

d. At interim power the indicated axial flux difference must be maintained within the range +10 percent to -14 percent.

e. For every 3.5 percent below interim power the permissible positive flux difference range is extended by +1 percent. For every 2 percent below interim power the permissible negative flux difference is extended by 1 percent.

f. Following initial loading and each subsequent reloading, a power distribution map, using the movable in-core detectors, shall be made to confirm that power distribution limits are met, in the full power configuration, before the reactor is operated above 75 percent of rated power.

g. For operation of the reactor above 75% of rated power;

(1) a full movable incore detector map shall be taken monthly. A full map is defined as surveillance of a minimum of 40 fuel assembly detector thimbles with at least 8 per quadrant.

(2) A partial movable incore detector map must be taken 10 to 17 days after the full map. A partial map is defined as surveillance of a minimum of 20 fuel assembly detector thimbles with at least 4 per quadrant.

(3) Two traverses with the movable incore detectors in appropriate alternate thimbles shall be taken during each calendar week.

h. If the quadrant to average power tilt exceeds a value T% as selected in specification 6.c, except for physics and rod exercise testing, then:

1) The hot channel factors shall be determined within 2 hours and the power level and trips adjusted to meet the requirements of Section 6a and b, or

2) If the hot channel factors are not determined within two hours, the power shall be reduced from interim power 2% for each percent of quadrant tilt.

3) If the quadrant to average power tilt exceeds $\pm$10%, except for physics tests, the power level and high neutron flux trip setpoint will be reduced from interim power, 2% for each percent of quadrant tilt.

i. If after a further period of 24 hours, the power tilt in 2 above is not corrected to less than $\pm$ T%, and

1) If design hot channel factors for interim power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Atomic Energy Commission.

2) If the design hot channel factors for interim power are exceeded and the power is greater than 10% - The Atomic Energy Commission shall be notified and the nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values.

3) If the hot channel factors are not determined, the Atomic Energy Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

7. IN-CORE INSTRUMENTATION

a. A minimum of 16 thimbles, at least 2 per quadrant, and the necessary associated detectors shall be operable during the check and calibration of nuclear instrumentation ion chambers.

b. Power shall be limited to 90% of interim power for 3 loop or 50% of interim power for 2 loop operation if the requirements in Section 7.a are not met.

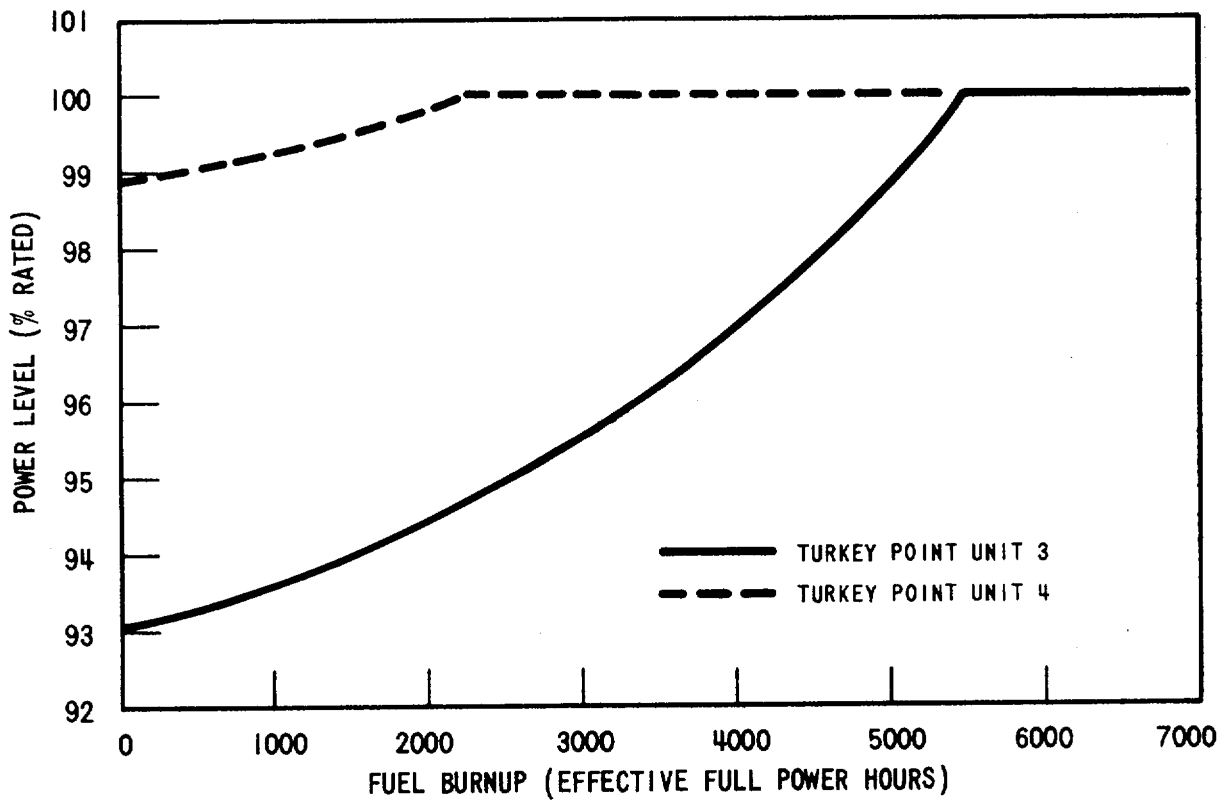

Figure 3.2-3 Maximum Allowable Operating Power Level versus Fuel Burnup

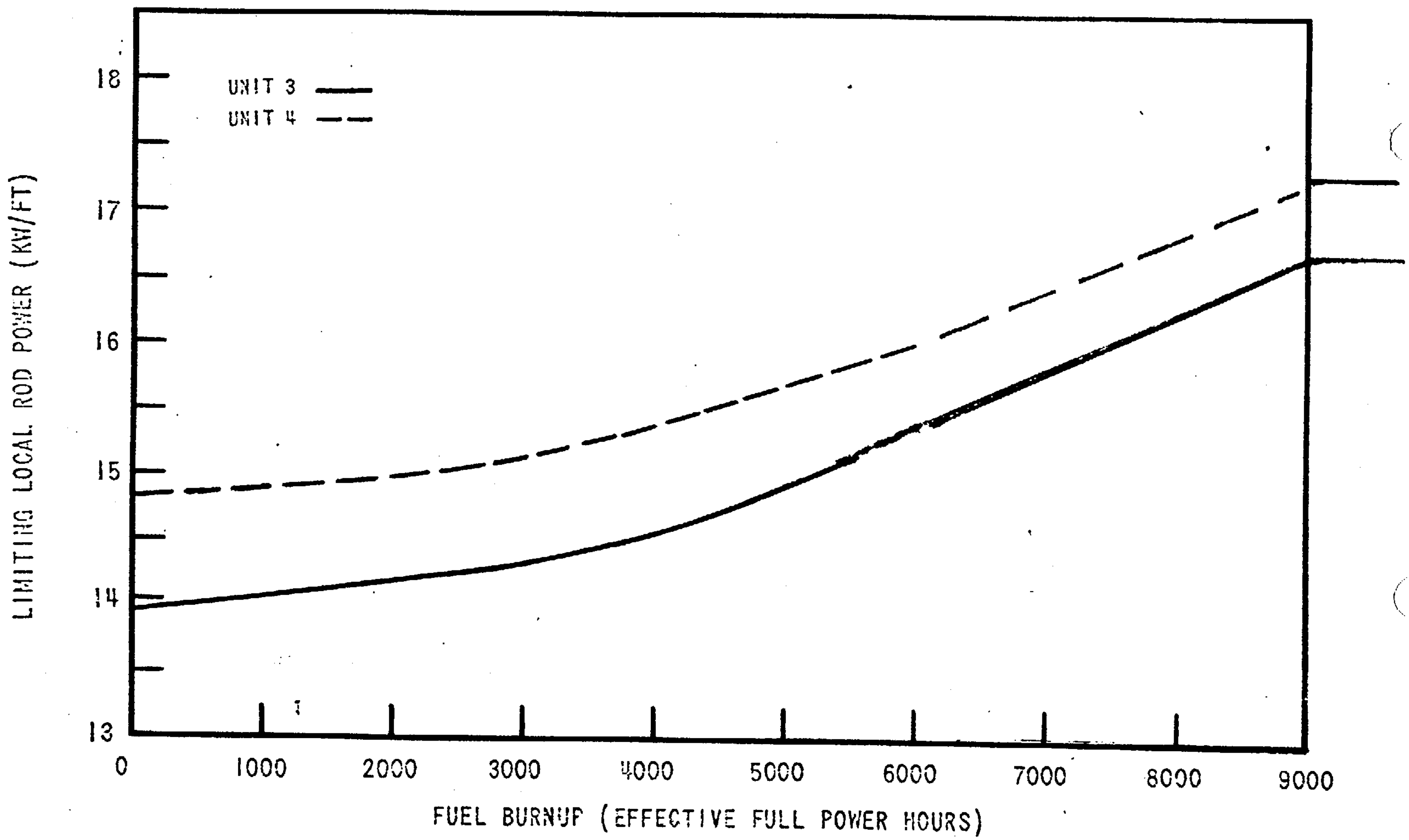

Figure 3.2-4. Maximum Allowable Local KW/FT

B2.3 BASES FOR LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Nuclear Flux

The power range reactor trip low set point provides protection in the power range for a power excursion beginning from low power.[1]

The power range reactor trip high set point protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.[2]

Reactor Coolant Temperature

The overtemperature Delta-T reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to loop transit time from the core to the temperature detectors (about 4 seconds),[2] and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[3] is always below the core safety limit as shown on **Figure 2.1-1.** If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced.[4] [5]

The overpower Delta-T reactor trip prevents power density anywhere in the core from exceeding 118% of design power density, and includes corrections for axial power distribution, change in density and heat capacity of water with temperature, and dynamic compensation for loop transit time from the core to the loop temperature detectors. The specified set points meet this requirement and include allowance for instrument errors.[3]

Part length rod insertion has been eliminated for this cycle to eliminate potential adverse axial power shapes.

Two criteria have been chosen as a design basis for fuel performance related to fission gas release, pellet temperature and cladding mechanical properties. First the peak value of linear power density must not exceed 18.0 kw/ft. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the initial steady state conditions for the peak linear power for a loss of coolant accident must not exceed the values assumed in the accident evaluation. This limit is required in order for the maximum cladding temperature to remain below those limits established by the Interim Policy Statement for LOCA. Interim power is limited to the level given in Figure 3.2-3, to compensate for the effects of fuel densification (1). The effects of fuel densification are such as to increase fuel stored energy and cause local power spikes. The decrease in fuel temperatures and stored energy as a result of cladding creep down in reference (1) permits higher power at increased burnup.

To aid in specifying the limits on power distribution the following hot channel factors are defined.

F_q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods. F_q is the product of F_q^N and F_q^E.

F_q^E, Engineering Heat Flux Hot Channel Factor is defined as the allowance on heat flux required for manufacturing tolerances.

F_q^N is the Nuclear Hot Channel Factor describing the neutron flux distribution in the core.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

It has been determined by analysis that the design limits on peak local power density, on minimum DNBR and LOCA are met, provided:

$$F^N_q \leq 2.50 \text{ and } F^N_{\Delta H} \leq 1.55$$

These quantities are measurable although there is not normally a requirement to do so. Instead it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

2. Control rod banks are sequenced with overlapping banks as shown in Figure 3.2-1.

3. The control bank insertion limits are not violated.

4. Part length control rods are not inserted.

5. Axial power distribution guide lines, which are given in terms of flux difference control, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in power between the top and bottom halves of the core. Calculation of core peaking factors under a variety of operating conditions have been correlated with axial offset. The correlation shows that an F^N_q of 2.50 and allowed DNB shapes, including the effects of fuel densification, are not exceeded if the axial offset (flux difference) is maintained between +13 and -17 percent. The specified limits of +10 and -14 allow for a 3% error in the axial offset.

For operation at a fraction, P, of interim power the design limits are met, provided,

$$F_q^N \leq 2.50 \ [1 + 0.2 \ (1-P)]$$ in the indicated flux difference range of +10 to -14 percent,

and $F_{\Delta H}^N \leq 1.55 \ [1 + 0.2 \ (1-P)]$

The permitted relaxation allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 5 are observed, these hot channel factor limits are met.

For normal operation and anticipated transients the core is protected from exceeding 18.0 KW/ft locally, and from going below a minimum DNBR of 1.30, by automatic protection on power, flux difference, pressure and temperature. Only conditions 1 through 4, above, are mandatory since the flux difference is an explicit input to the protection system.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit to F_q^N there is a 5 percent allowance for uncertainties[1] which means that normal operation of the core within the defined conditions and procedures is expected to result in $F_q^N \leq 2.50/1.05$ even on a worst case basis. When a measurement is taken experimental error must be allowed for and 5 percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system.

In the specified limit of F_q^N there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.58/1.08$. The logic behind the larger uncertainty in this case is that (a) abnormal perturbations in the radial power shape (e.g., rod misalignment) affect $F_{\Delta H}^N$, in most cases without necessarily affecting F_q^N, and (b) an error in the predictions for radial power shape, which may be detected during startup physics tests can be compensated for F_q^N by tighter axial control limits, but compensation for $F_{\Delta H}^N$ is less readily available. Five percent is the appropriate allowance for a full core map taken with the movable in-core detector flux mapping system.

At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of a reduction in F_q or expansion of permissible quadrant tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup.

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rods and an error allowance. No increase in F_q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_q occurs.[2]

References:

(1) WCAP-8074
(2) WCAP-7912-1