Docket No. 50-250

3/30/73

Florida Power & Light Company
ATTN: Dr. James Coughlin
P. O. Box 3100
Miami, Florida 33101

Change No. 3
License No. DPR-31

Gentlemen:

By letter dated March 5, 1973, you submitted a report entitled "Fuel Densification Turkey Point Plant Unit No. 3." You stated that the analyses contained therein were performed consistent with our "Technical Report on Densification of Light Water Reactor Fuels" (November 14, 1972), and, further, that these analyses reflect the latest Westinghouse techniques employed in recently submitted analyses for Point Beach 2. Also, you show in your letter by a comparison of key fuel parameters that these analyses apply conservatively to Unit 4.

We have reviewed your report and have determined that the effects of fuel densification have been adequately analyzed. We note that the plant capability with respect to such effects is limited by the loss-of-coolant accident (LOCA), and that to meet the 2300°F maximum clad temperature limit, the maximum permitted reactor power level is 93% of rated power (2046 MWt).

Under your March 5, 1973 letter you also submitted proposed Technical Specification changes to reflect the results of the densification study. We have reviewed your proposal and have concluded that certain changes to the Technical Specifications for Turkey Point Units 3 and 4 are required for protection against the effects of fuel densification. Accordingly,

1. The maximum permitted reactor power level is 2046 MWt.
2. The fuel residence time for Cycle 1 is limited to 10,000 effective full power hours.
3. The overtemperature and overpower constant, f(ΔI), has been revised.
4. The power distribution limits have been revised and additional surveillance requirements have been incorporated.

We have determined that with these changes to the Technical Specifications, Turkey Point Unit 3 and Unit 4 (when licensed) can be operated safely.

We conclude the changes do not involve significant hazard considerations not described or implicit in the Final Safety Analysis Report 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 License No. DPR-31 are hereby changed as set forth in revised pages which are enclosed.

Sincerely,

/S/

R. C. DeYoung, Assistant Director
for Pressurized Water Reactors
Directorate of Licensing

Enclosure:
As stated

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

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bcc: J. R. Buchanan, ORNL
Thomas B. Abernathy, DTIE

I called J. Hendrie on 3/30/73 and he said OK to go without concurrence of BC, AD and him. RWK

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1972-466-983

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

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MAR 30 1973

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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 3 in November 1972. Interim power of the reactor core is limited to 2046 MWt. 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.

3.2 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability: Applies to the operation of the control rods and power distribution limits.

Objective: To ensure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification: 1. CONTROL ROD INSERTION LIMITS

a. Whenever the reactor is critical, except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

b. Whenever the reactor is critical, except for physics tests and control rod exercises, the control group rods shall be no further inserted than the limits shown by the solid lines on Figure 3.2-1 for three loop or two loop operation.

c. After 70% of the second and subsequent cycles as defined by burnup, the limits shall be adjusted as a linear function of burnup toward the end-of-core life as shown by the dotted lines on Figure 3.2-1.

d. The end-of-core life limit shown on Figure 3.2-1 may be revised on the basis of physics calculations and physics data obtained during startup and subsequent operation.

e. Part length rods shall not be permitted in the core except for low power physics tests and for axial offset calibration tests performed below 75% of rated power. 3

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. <u>MISALIGNED CONTROL ROD</u>

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 rated 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

3. <u>ROD DROP TIME</u>

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% $\Delta k/k$ 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: 3

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

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

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 neutron flux trip setpoint shall be reduced by 1 percent for every percent excess over $F^N_{\Delta H}$ or F^N_q, 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 0.93, at which the reactor can be operated shall be determined by 3

$$N = \frac{13.95}{5.56 \text{ X } 1.02 \text{ X } 1.019 \text{ X } 1.007 \text{ X } M}$$

where $M = 2.76 \text{ X } L \text{ X } \frac{F_{xy}}{1.435} \left[1 + 2\,(T/100 - 0.02)\right]$;

L = 1, or 0.95 when surveillance of the axial peaking factor F^N_Z, in Specification 6.g is in effect;

F_{xy} is 1.435, or the value of the unrodded horizontal plane peaking factor appropriate to Fq 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 the option to measure F_{xy} is in effect.

Reactor Coolant Temperature

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

ΔT_o = Indicated ΔT at rated power

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 ofinterim power, $f(\Delta I) = 0$. 3

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) = 1.095;

(Two Loop Operation) = 0.88

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. 3

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

g. If the L factor used in Section 6.c is 0.95 axial surveillance of F_Z^N shall consist of:

   1) Two traverses with the movable in-core detectors in appropriate alternate pairs of channels shall be taken every eight hours, or at a frequency of 0. 30, 60, 120, 180, 240, 360 and 480 minutes following accumulated control rod motion of five steps except for control rod exercises limited to within 10 steps from the top of the core. From the traverses, determination that $F_Z^N \leq 1.63/S(Z)$ is the densification penalty factor as a function of core height shown in Fig. 3.2-3. This allows 4% measurement error.

   2) If a traverse indicates $F_Z^N > 1.63/S(Z)$, one of the following must be done as soon as practicable, but not exceeding two hours after the traverse:

a. Correct the condition and verify $F_Z^N \leq 1.63/S(Z)$ with two traverses.

b. If the measured F_z^N exceeds 1.63/S(Z), the reactor power shall be reduced by 1 percent for every percent excess over 1.63/S(Z).

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 3
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.

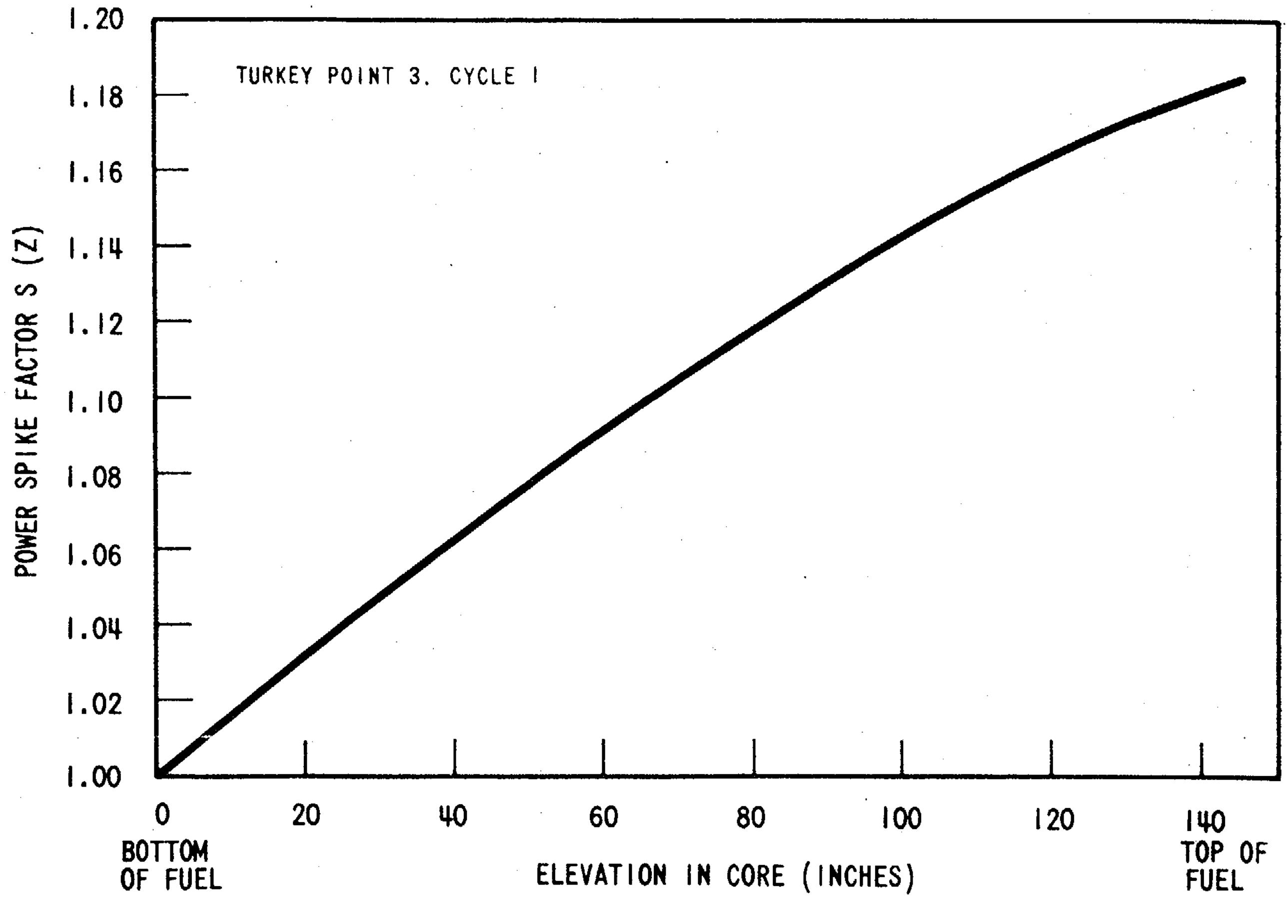

Figure 3.2-3. Power Spike Factor versus Elevation 3/2/73

6117-2

The f(ΔI) function in the Overpower ΔT and Overtemperature ΔT protection system setpoints includes effects of fuel densification on core safety limits. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and DNBR of 1.30 3
will not be violated.[10]

Pressurizer

The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[6]

The high pressurizer pressure reactor trip is set below the set pressure of the pressurizer safety valves and limits the reactor operating pressure range. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified set point allows margin for instrument error [3] and transient level overshoot before the reactor trips.

Reactor Coolant Flow

The low flow reactor trip protects the core against DNB in the event of loss of one or more reactor coolant pumps. The set point specified is consistent with the value used in the accident analysis.[7] The low frequency and under voltage reactor trips protect against a decrease in flow. The specified set points assure a reactor trip signal before the low flow trip point is reached. The underfrequency trip set point preserves the coastdown energy of the reactor coolant pumps, in case of a system frequency decrease, so DNB does not occur.

Steam Generators

The low-low steam generator water level reactor trip assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting of the auxiliary feedwater system.[8]

Reactor Trip Interlocks

Specified reactor trips are by passed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed set points above which these trips are made functional assures their availability in the power range where needed.

An automatic reactor trip will occur if any pump is lost above 55% power which will prevent the minimum value of the DNBR from going below 1.30 during normal and anticipated transient operations when only two loops are in service,[9] and the over-temperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

Reset of reactor trip interlocks will be done under strict administrative control.

REFERENCES

(1) FSAR 14.1.1
(2) FSAR, 14.1.2
(3) FSAR 14.1
(4) FSAR 7.2, 7.3
(5) FSAR 3.2.1
(6) FSAR 14.3.1
(7) FSAR 14.1.6
(8) FSAR 14.1.11
(9) FSAR 14.1.9
(10) WCAP-8074

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, as defined in Section 1.0 of these Technical Specifications is 2046 MW thermal. Interim power is limited to 93% of rated power (2200 MWt) 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) would permit higher power, but have not been allowed in this specification. 3

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 dividied 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.

$F^N_{\Delta H}$, 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 3
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_q^N 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. In order to take credit for operation at the bounding value of the correlation in the permitted range of the axial offset surveillance of the axial peaking factor, F_Z^N is specified. Otherwise, the specification leads to a 5% penalty in power, unless the measurement of F_{xy} provides compensating benefit.

For operation at a fraction, P, of 93% of interim power the design limits
are met, provided, 3

$F_q^N \leq 2.50\ [1 + 0.2\ (1-P)]$ 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 factors 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 uncertainities[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 movable incore detector flux mapping system.

The measured value of F_q^N must be additionally corrected by including a penalty as shown on Figure 3.2-3 (at the appropriate core location) to account for fuel densification effects before comparison with the limiting value above

In the specified limit of F_q^N there is a 8 percent allowance for uncer- 3
tainties 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. Montly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 1).

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[1] WCAP-8074

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 3
do not extend to the unrodded plane, where the maximum F_q occurs.