

OCT 16 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Change No. 24
License No. DPR-23

Gentlemen:

By letters dated April 20, June 28, August 14, and September 4, 1973, you proposed changes to the Technical Specifications appended to License No. DPR-23, as amended, for Unit No. 2 at the H. B. Robinson Steam Electric Plant. The proposed change would revise a safety limit, limiting safety system settings, limiting conditions for operation, and surveillance which pertain to fuel densification. In support of this proposal, you submitted an analysis of the effects of fuel densification on Unit No. 2 (Westinghouse non-proprietary report WCAP-8115). The analysis follows the methods previously reviewed by the staff for Unit No. 2 at Point Beach, and takes into account expected fuel clad flattening in regions 2 and 3 of the Cycle 2 core. You have concluded from the analysis that full power operation to a fuel exposure of 7000 effective full power hours (EFPH) is justified with appropriate provisions made for reduced power peaking and for limiting steam generator leakage.

We have now completed our review of the fuel densification analysis and the proposed changes to the Technical Specifications. In order to assure safe operation of Robinson-2 with collapsed fuel rods you have:

- (1) Limited the clad temperature in collapsed sections of a fuel rod to less than 1800°F during a loss-of-coolant accident.
- (2) Adequately allowed for the large power spike that will result in rods adjacent to collapsed rods.
- (3) Included an additional 1.9% decrease in DNBR to account for increased pellet clad eccentricity and reduced fuel rod circumference and heat transfer area.
- (4) Included a 10% penalty applied at the point of minimum DNBR to conservatively account for possible contact of rods due to flattening and bowing.

- (5) Changed the reference axial power distribution for DNB analysis from a chopped cosine with a 1.79 peak to average power to one with a 1.55 peak to average power.

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- (6) For excore monitoring and at power levels up to 94.8% rated power, reduced the overall peaking factor F_Q from 2.83 to 2.60 to allow for local power peaking due to fuel densification and flattened cladding.
- (7) For incore monitoring and at power levels up to 100% rated power, reduced F_Q^T from 2.83 to 2.41 for fuel in regions 2 and 3 and to 2.56 for fuel in region 4.

On the basis of the above and our review of your report, we have determined that the effects of fuel densification have been adequately analyzed and that the plant can be operated at 100% of rated power with appropriate changes to the Technical Specifications.

Changes to the Technical Specifications have been made in Section 2.1 to limit fuel residence time to 7000 EPPH; to Section 2.3 to further reduce the overtemperature ΔT trip setpoint for positive axial offset; to Section 3.1.5 to limit primary to secondary steam generator leakage so that in the event of an overpower transient and failure of 12% of the flattened fuel rods the radiation exposure at the site boundary will not exceed the limits of 10 CFR Part 20; to Section 3.10 to incorporate revised full length and part length control rod insertion limits and power distribution limits; and to Section 4.11 to incorporate additional power distribution surveillance requirements. The positive limit on axial offset has been arbitrarily reduced from 9% to 3% pending receipt and evaluation of information which we requested by letter dated September 12, 1973, regarding the consequences of power peaking above the core midplane. This action has been discussed with your staff.

We conclude that the changes to the Technical Specifications described above do not involve a significant hazards consideration and that 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 appended to Facility Operating License No. DPR-23 are hereby changed as set forth by margin bars in Attachment A.

Interim conditions for operation required by our letter dated July 25, 1973, are hereby revoked.

Sincerely,

Original signed by
Robert J. Schemel for

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:

OF ~~see~~ next page

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OCT 16 1973

Enclosure:
Attachment A - Change No. 24 to
the Technical Specifications

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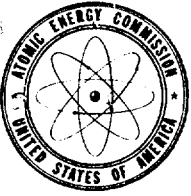
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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

October 16, 1973

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

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Sincerely,



Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure:
see next page

October 16, 1973

Enclosure:

Attachment A - Change No. 24 to
the Technical Specifications

cc w/enclosure:

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ATTACHMENT A

CHANGE NO. 24 TO THE TECHNICAL SPECIFICATIONS

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT, REACTOR CORE

Applicability:

Applies to the limiting combinations of thermal power, Reactor Coolant System pressure, coolant temperature, and flow when the reactor is critical.

Objective:

To maintain the integrity of the fuel cladding.

Specification:

- a. The combination of thermal power level, coolant pressure, and coolant temperature shall not exceed the limits shown in Figure 2.1-1 when full flow from three reactor coolant pumps exists and shall not exceed the limits shown in Figure 2.1-2 when the full flow from two reactor coolant pumps exists.
- b. When full flow from one reactor coolant pump exists, the thermal power level shall not exceed 20%, the coolant pressure shall remain between 1820 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 590°F.
- c. When natural circulation exists, the thermal power level shall not exceed 12%, the coolant pressure shall remain between 2135 psig and 2400 psig, and the Reactor Coolant System average temperature shall not exceed 620°F.
- d. The safety limit is exceeded if the combination of Reactor Coolant System average temperature and thermal power level is at any time above the appropriate pressure line in Figures 2.1-1 or 2.1-2 or if the thermal power level, coolant pressure, or Reactor Coolant System average temperature violates the limits specified above.
- e. The fuel residence time for Cycle 2 shall be presently limited to 7,000 effective full power hours (EFPH) under design operating conditions. The Licensee may propose to operate the core in excess of 7,000 EFPH by providing an analysis which includes the effect of further clad flattening or a change in operating conditions. Any such analysis, if proposed, shall be approved by the Regulatory staff prior to operation in excess of 7,000 EFPH.

Basis:

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under all operating conditions. This is accomplished by maintaining the

the self actuated safety valves on the steam generators. An arbitrary upper safety limit of 120% for thermal power is shown. The upper limit is below the damage limit of 1.7% for maximum clad strain which is reached at 123% thermal power with design hot channel factors.

The curves of Figure 2.1-2 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation) represent the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the DNB ratio is equal to 1.30 or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the DNB ratio reaches 1.30 and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. In order to completely specify limits at all power levels, arbitrary constant upper limits of average temperatures are shown for each pressure at powers lower than approximately 40%. The limits at low power as well as the limits based on the average enthalpy at the exit of the core are considerably more conservative than would be required if they were based upon a minimum DNB ratio of 1.30. The plant conditions required to violate these limits are precluded by the protection system and the self actuated safety valves on the steam generator. An upper limit of 70% for power is shown to completely bound the area where clad integrity is assured. This latter limit is arbitrary but cannot be reached due to the permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculations of the curves shown in Figures 2.1-1 and 2.1-2. However, the curves presently in the Technical Specifications have not been revised since analysis shows that they are adequate and conservative even including the effects of densification.

The limits specified for one loop operation and natural circulation result in DNB ratios greater than 1.30.

The specified limits are based on a $F_{\Delta H}^N$ of 1.55, a 1.55 cosine axial flux shape and a DNB analysis as described in Section 4.3 of the report Fuel Densification - H. B. Robinson Steam Electric Plant, Unit 2 (WCAP-8114), which includes the effects of fuel densification and fuel clad collapse.

These limiting hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. The control rod insertion limits are covered by Specification 3.10. Somewhat worse hot channel factors could occur at lower power levels because additional control rods are in the core. However, the control rod insertion limits dictated by Figure 3.10-1 ensure that the DNB ratio is always greater at part power than at full power.

The hot channel factors are also sufficiently large to account for the degree of malpositioning of part length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required.⁽²⁾ Rod withdrawal block and load runback occurs before reactor trip setpoints are reached.

The safety limit curves given in Figures 2.1-1 and 2.1-2 are for constant flow conditions. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 14.1 of the FSAR.

The reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.30⁽³⁾ based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.2°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and \pm 30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 45%.

To provide the Commission with added verification of the safety and reliability of pre-pressurized zircaloy clad nuclear fuel, a limited program of nondestruction fuel inspection will be conducted. The program shall consist of a visual inspection (e.g., underwater TV, periscope, and other) of the two lead burnup fuel assemblies during the second and third refueling outages. Any condition observed by this inspection which could lead to unacceptable fuel performance may be the object of an expanded effort. The visual inspection program and, if indicated, the expanded program will be conducted in addition to that being performed in the Saxton and Cabrera reactors. If another domestic plant which contains pre-pressurized fuel of the same design as that used for H. B. Robinson Unit No. 2 and reaches the second and third refueling outages first, and if a limited inspection program is or has been performed there, then the program may not have to be performed at H. B. Robinson Unit No. 2. However, such action requires approval of the AEC.

The fuel residence time for Cycle 2 is limited to 7,000 EFPH to assure no further fuel clad flattening without prior review by the Regulatory staff. Prior to 7,000 EFPH, the Licensee may provide the additional analyses required for operation beyond 7,000 EFPH.

References:

- (1) FSAR, Section 3.2.2
- (2) FSAR, Section 14.1.3
- (3) FSAR, Section 7.2.1

(d) Overtemperature ΔT

$$\leq \Delta T_0 [K_1 - K_2 (T - 574) + K_3 (P - 2235) - f(\Delta I)]$$

where:

ΔT_0 = Indicated T at rated power, °F

T = Average temperature, °F

P = Pressurizer pressure, psig

K_1 = 1.095

K_2 = 0.0107

K_3 = 0.000453

and $f(\Delta I)$ is 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 plant startup tests such that:

- (1) For $(q_t - q_b)$ within +3% and -17% where q_t and q_b are 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, $f(\Delta I) = 0$. For every 3.5% below 94.8% rated power level, the permissible positive flux difference range is extended by +1 percent. For every 2% below 94.8% rated power level, the permissible negative flux difference range is extended by -1 percent.
- (2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +3% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 3.5% of the value of ΔT at rated power.
- (3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17%, the ΔT trip setpoint shall be automatically reduced by 2.0% of the value of ΔT at rated power.

The overtemperature ΔT 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 piping transit delays from the core to the temperature detectors (about 4 seconds)⁽⁴⁾, 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, (2) is always below the core safety limit as shown in Figure 2.1-1 or 2.1-2. If axial peaks are greater than design, as indicated by difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced.⁽⁵⁾⁽⁶⁾

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 112% of design power density as discussed in Section 7.2.3 and 14.1.3 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.⁽²⁾

The overpower and overtemperature protection system setpoint have been revised to include effects of fuel densification on core safety limits. The revised setpoints in the Technical Specifications insure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figures 2.1-1 and 2.1-2.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis⁽⁷⁾. The under voltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 ft³ of water corresponds to 92% of span. The specified setpoint allows margin for instrument error⁽²⁾ and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

3.1.5 LEAKAGE

Specification:

- 3.1.5.1 If the primary system leakage exceeds 1 gpm and the source of leakage is not identified within 12 hours, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the source of leakage exceeds 1 gpm and is not identified within 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.
- 3.1.5.2 If the sources of leakage have been identified and it is evaluated that continued operation is safe, operation of the reactor with a total leakage rate not exceeding 10 gpm shall be permitted. If leakage exceeds 10 gpm, the reactor shall be placed in the hot shutdown condition within 12 hours utilizing normal operating procedures. If the leakage exceeds 10 gpm for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.
- 3.1.5.3 If the primary to secondary leakage in a steam generator exceeds the limit in Figure 3.1-4, the reactor shall be placed in the hot shutdown condition within 8 hours utilizing normal operating procedures. If the leakage exceeds this limit for 24 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Basis:

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System, and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage is a conservative limit on what is allowable before the guidelines of 10 CFR Part 20 would be exceeded. This is shown as follows: If the reactor coolant activity is $50/\bar{E}$ uCi/cc (\bar{E} = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.00 \times 10^{-5}$ sec/m³ is about the 10 CFR Part 20 guideline of 0.5 R/yr^(1,2).

With the limiting reactor coolant activity and assuming initiation of 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet

header would annunciate in the control room and initiate closure of the vent line from the surge tank in the Component Cooling System within about one-half minute⁽³⁾. In the case of failure of the closure of the vent line and resulting continuous discharge to the atmosphere via the component cooling surge tank vent, the resultant dose at the site boundary would be 0.50 R/yr, as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The 1 gpm leakage rate is within the range of detectable leakage and is well below the capacity of one coolant charging pump (77 gpm).

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Plant Operating Staff, will be documented in writing and approved by either the Plant Superintendent or Operating Supervisor. Under these conditions, an allowable leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm (or less) is also well within the capacity of one charging pump and makeup would be available even under the loss of offsite power condition.

If leakage is to the containment, it may be identified by one or more of the following methods⁽⁴⁾:

- a. The containment air particulate monitor is sensitive to low leak rates. The rates of leakage to which the instrument is sensitive are 0.01 gpm to greater than 10 gpm, assuming corrosion product activity, and little or no fuel cladding leakage. Under these conditions, a coolant leak of 1 gpm is detectable within 1 minute.
- b. The containment radiogas monitor is less sensitive but can be used as a backup to the air particulate monitor. The sensitivity range of the instrument is from 10^{-3} to 10^{-6} uCi/cc assuming a constant background radioactivity in the containment atmosphere due to normal leakage of reactor coolant. With equilibrium fission product gaseous activity, a 1 gpm leak would double the background activity in less than 2 hours.
- c. The humidity detector provides a backup to (a) and (b). This instrumentation will be sensitive to incremental increases of leakage to the containment atmosphere on the order of 0.40 gpm per °F of dew point temperature increase.
- d. A leakage detection system which determines leakage losses from water and steam systems within the containment. This system collects and measures moisture condensed from the containment atmosphere

by cooling coils of the main recirculation units. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks, from the cooling coils themselves which are part of the containment boundary. Condensate flows from approximately 0.5 gpm to greater than 10 gpm can be detected and measured by this system. Condensate flow corresponding to coolant leakage of approximately 1 gpm can be detected within 10 minutes.

Leaks less than 1 gpm can be measured by periodic observation of the level changed in the condensate collection system.

If leakage is to another closed system, it will be detected by the plant radiation monitors and/or inventory control.

Steam generator tube leakage limits are based upon offsite dose considerations as limited by 10 CFR Part 20 in the event of a 112% overpower transient with the presence of collapsed rods.

The evaluation of the overpower transient assumed:

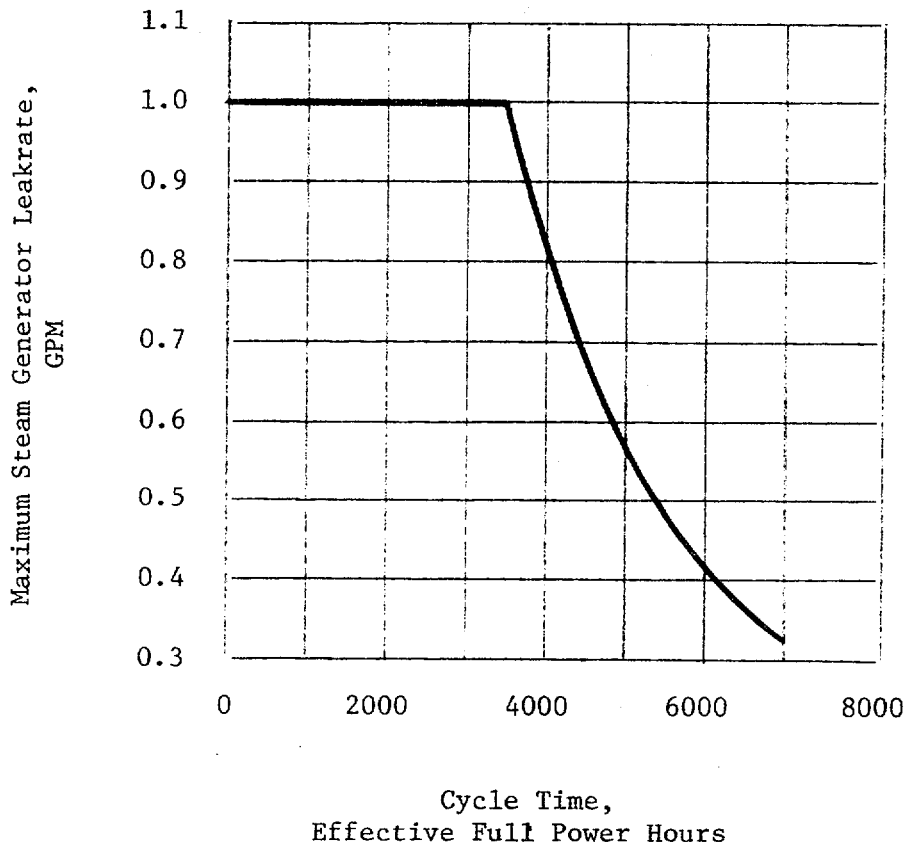
- a. Ten percent of the core iodine inventory is present in the fuel rod gaps.
- b. The overpower transient is assumed to fail 12% of all flattened rods in the core, and all iodine in the gaps of those rods is immediately released to the coolant.
- c. The coolant activity is assumed to leak to the secondary side at a constant rate as given in Figure 3.1-4.
- d. Iodine in the secondary system is released instantaneously.
- e. No activity is released after 2 hours.
- f. $X/Q = 1.0 \times 10^{-3} \text{ sec/m}^3$.
- g. The 2-hour site boundary dose limit is 1.5 Rem thyroid as per 10 CFR Part 20.

Operator action to start to place the reactor in the hot shutdown condition within 12 hours utilizing normal operating procedures provides adequate time for an orderly reduction of power. The hot shutdown condition allows personnel to enter the containment and inspect the pressure boundary for leaks. The 24 hours allowed prior to the operator starting to place the reactor in the cold shutdown condition utilizing normal operating procedures allows reasonable time to correct small deficiencies. If major repairs are needed, a cold shutdown condition would be in order.

References

- (1) FSAR, Section 2.7.3
- (2) FSAR, Table 9.2-5
- (3) FSAR, Section 11.2.3
- (4) FSAR, Section 4.2.7

STEAM GENERATOR LEAKRATE LIMIT



3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability:

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective:

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specifications:

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be no further inserted than the limits shown by the solid lines on Figure 3.10-1 for 3 loop or 2 loop operation.

3.10.1.4 After 50% 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 values as shown by the dotted lines on Figure 3.10-1.

3.10.1.5 Except for physics tests, if a part-length or full-length control rod is more than 15 inches out of alignment with its bank, then within two hours:

- a. Correct the situation, or
- b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1.1, or
- c. Limit power to 75% of rated power for 3 loop operation or 45% of rated power for 2 loop operation.

3.10.1.6 When the reactor is operated above 94.8% of rated power, control rods in Bank D shall be within 5 steps of the preselected position for the control rods at xenon equilibrium except:

- a. as necessary to control xenon oscillations or to obtain core power maps for a new preselected position of Bank D, or
- b. for time intervals up to two hours for control of transients other than xenon oscillations.

3.10.1.7 During physics and control rod exercises, the insertion limits need not be observed but the Figure 3.10-2 must be observed.

3.10.2 Power Distribution Limits

3.10.2.1 Limiting Values

3.10.2.1.1 Power distribution limits are expressed as hot channel factors. Limiting values above 94.8% rated power are:

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

$$[F_j(z) S(z)]_{\max} = \frac{2.184/P}{\bar{R}_j(1 + 2\sigma)} \quad \text{for regions 2 and 3 fuel}$$

$$[F_j(z) S(z)]_{\max} = \frac{2.325/P}{\bar{R}_j(1 + 2\sigma)} \quad \text{for region 4 fuel}$$

where: (a) P is the fraction of rated power at which the core is operating ($P \leq 1.0$).

(b) \bar{R}_j , for thimble j, is determined from core power maps i and is by definition:

$$\bar{R}_j = 1/6 \sum_{i=1}^6 \frac{F_{qi}^N}{[F_{ij}(z) S(z)]_{\max}}$$

F_{qi}^N includes factors for uncertainty in measurement, engineering tolerances, and spike penalties.

(c) σ is the standard deviation associated with the determination of \bar{R} .

(d) S(z) is defined in Figure 3.10-3 and includes the variation of limiting kW/ft as a function of core height above the core midplane.

If measured peaking factors exceed these values, the maximum allowable reactor power level and the nuclear overpower trip setpoint shall be reduced in direct proportion to the amount which $F(z) S(z)$ or $F_{\Delta H}^N$

exceeds the limiting values, whichever is more restrictive. The measured value of $F(z) S(z)$ is permitted to remain in excess of the limiting value for a period of time not to exceed two hours, if the measured value is no greater than 4% in excess of the limit and if immediate corrective action is initiated to bring the measured value below the limit. If the measured value is greater than 4% in excess of the limit, the power shall be immediately reduced. If the $F_{\Delta H}^N$ or $F(z) S(z)$ cannot be reduced below the limiting values within twenty-four hours, the overpower ΔT and overtemperature ΔT trip setpoint shall be similarly reduced.

- 3.10.2.1.2 At all times the measured value of axial offset must either be between -17% and +3% or the overtemperature and overpower ΔT trip setpoints must be reduced as required in Specifications 2.3.1.2(d) and (e).
- 3.10.2.2 If the quadrant to average power tilt as defined in Section 1.9 exceeds 2% except for physics and rod exercise testing, then:
- a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.10.2.1 or
 - b. If the hot channel factors are not determined within two hours, the power shall be reduced from the maximum allowed power as defined in Specification 3.10.2.1, 2% for each percent of quadrant tilt greater than 2% and the trip settings in Specifications 2.3.1.2(a), (d), and (e) shall be reduced by a like amount.
 - c. If the quadrant to average power tilt exceeds $\pm 10\%$, except for physics tests, a power reduction to less than 50% of licensed power will be initiated immediately and the trip settings shall be reduced as in b above. Until the tilt is reduced to less than 10%, the reactor will be operated only for the purpose of determining the cause of the tilt.

3.10.3 Rod Drop Time

- 3.10.3.1 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.

3.10.4 Part Length Control Rod Banks

- 3.10.4.1 The eight (8) part length control rods shall be configured under administrative control into one of the following part length rod configurations.

- a. Four part length rods occupying core positions K-6, K-10, F-6, and F-10 shall constitute a part length control rod bank, hereafter designated bank P-1.
- b. Four part length rods occupying core positions P-8, H-2, H-14, and B-8 shall constitute a part length control bank, hereafter designated part length bank P-2.
- c. Combined Banks P-1 and P-2, hereafter designated Bank P-3.

3.10.4.2 The part length control rods will not be inserted. They will remain in the fully withdrawn position except for physics tests and for axial offset calibration which will be performed at 75% of permitted power or less.

3.10.5 Inoperable Full Length and Part Length Control Rods

3.10.5.1 A full length or part length control rod shall be deemed inoperable if (a) the rod is misaligned by more than 15 inches with its bank, (b) if the rod cannot be moved by its drive mechanism, or (c) if its rod drop time is not met in the case of a full length rod.

- 3.10.5.2 No more than one inoperable control rod shall be permitted during power operation.
- 3.10.5.3 If a full length control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that shutdown margin equal to or greater than shown on Figure 3.10-2 results.
- 3.10.6 Power Ramp Rate Limits
- 3.10.6.1 Should a power level less than 95% be maintained continuously for more than 100 hours but less than 24 days, the rate of power increase shall be limited to 10% per hour.
- 3.10.6.2 Should a power level less than 95% be maintained continuously for more than 24 days, the rate of power increase shall be limited to 3% per hour from 25% to 100% of full power.
- 3.10.7 Required Shutdown Margins
- 3.10.7.1 When the reactor is in the hot shutdown condition, the shutdown margin shall be at least that shown in Figure 3.10-2.
- 3.10.7.2 When the reactor is in the cold shutdown condition, the shutdown margin shall be at least 1% $\Delta k/k$.
- 3.10.7.3 When the reactor is in the refueling operation mode, the shutdown margin shall be at least 10% $\Delta k/k$.

Basis:

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of a hypothetical rod ejection and provide for acceptable nuclear peaking

factors. The solid lines shown in Figure 3.10-1 meet the shutdown requirement for the first 50% of second cycle. The end-of-cycle life limit is represented by the dotted lines. The end-of-cycle life limit may be determined on the basis of plant startup and operating data to provide a more realistic limit which will allow for more flexibility in plant operation and still assure compliance with the shutdown requirement. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin equivalent to 2.20% reactivity⁽³⁾ at end of life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin. The specified control rod insertion limits have been revised for Cycle 2 in order to meet the design basis criteria on (1) potential ejected control rod worth and peaking factor⁽⁴⁾, (2) radial power peaking factors, $F_{\Delta H}$, and (3) required shutdown margin.

The various control rod banks (shutdown banks, control banks, and part length rods) are each to be moved as a bank, that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position⁽²⁾. The 15-inch permissible misalignment provides an enforceable limit below which design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.2.2 are acceptable because complete rod misalignment (part-length or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power to 50% will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

Part length rod insertion has been limited to eliminate adverse power shapes (Section 3.10.4.2).

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 18.6 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 clad temperature to remain below that established by the Interim Policy Statement for LOCA. To aid in specifying the limits on power distribution the following

hot channel factors are defined. F_q^N , Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. $F_{\Delta H}^N$, 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.

For operation below 94.8% rated power, it has been determined by analysis that the design limits on peak local power density and on minimum DNBR at full power and LOCA are met provided:

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 3.10-1.
3. The control bank insertion limits are not violated.
4. Part length control rods are not inserted.
5. Axial offset limits are observed.

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 2.52 and allowed DNB shapes, including the effects of fuel densification, are not exceeded if the axial offset (flux difference) is maintained between -20% and +6%. The specified limits of -17% and +3% allow for a 3% error or more in the axial offset. Therefore, there is no requirement for measurement where the LOCA limited peak local rod power corresponds to an $F_q^N \leq 2.52$ at 94.8% rated power providing the flux difference is maintained between -20% and +6%.

For operation at power levels above 94.8% rated power, F_q^N must be reduced proportionately to limit the linear heat generation rate in regions 2 and 3 fuel to 14.2 kW/ft. Therefore, at rated power F_q^N must be equal to or less than 2.34. To provide confidence that F_q^N and the linear heat generation rate are not exceeded, monitoring of $F(z) S(z)$ which is defined as the value of the core axial peak to average power as determined by the product of the densification penalty factor $S(z)$ as a function of core height and the corresponding value of the average core axial point to average power $F(z)$ from the movable detector system is necessary. The equation in 3.10.2.1 includes allowance for two uncertainty factors, F_q^a and F_u^N , which are, respectively, the uncertainty associated with the analog equipment used to determine the product of $F(z) S(z)$ (1.02), and the nuclear uncertainty factor associated with the movable detector system in producing a full core map (1.05). Because region 4 fuel can be operated

for a much longer time before it is likely to collapse, the LOCA requirement for it is less severe. At rated power, F_q^N for region 4 fuel must be 2.49 or less which corresponds to 15.1 kW/ft. F_q^a and F_u^N remain unchanged.

In the specified limit of F_q^N there is a 5% allowance for uncertainties⁽¹⁾ which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured $F_q^N \leq 2.34/1.05$ for regions 2 and 3, for example, at rated power even on a worst case basis. When a measurement is taken experimental error must be allowed for and 5% is the appropriate allowance for a full core representative 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 in Figure 3.10-3 (at the appropriate core location) to account for fuel densification effects before comparison with the limiting value above.

In the specified limit of $F_{\Delta H}^N$ there is an 8% allowance for design prediction uncertainties, which means that normal operation of the core is expected to result in $F_{\Delta H}^N \leq 1.55/1.08$ at rated power. The uncertainty to be associated with a measurement of $F_{\Delta H}^N$ by the movable incore system on the other hand is 3.65% which means that the normal operation of the core shall result in a measured $F_{\Delta H}^N \leq 1.55/1.0365$ at rated power. The logic behind the larger design uncertainty in this case is that (a) abnormal perturbation in the radial power shape (e.g., rod misalignment) affect $F_{\Delta H}^N$ is in most cases without necessarily affecting F_q^N through movement of part length rods and can limit it to the desired value, (b) while the operator has some control over F_q^N through F_z^N by motion of control rods, he has no direct control over $F_{\Delta H}^N$, and (c) an error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_q^N by tighter axial control, but compensation for $F_{\Delta H}^N$ is less readily available.

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.

The specified rod drop time is consistent with safety analyses that have been performed⁽¹⁾.

An inoperable rod imposes additional demands on the operator. The permissible number of inoperable control rods is limited to one in order to limit the magnitude of the operating burden, but such a failure would not prevent dropping of the operable rods upon reactor trip.

Analyses indicate the 10% and 3% per hour rates of power increase are slow enough to relax strains which may be induced by axial pellet expansion at the pinched down portion of the clad during power increases following moderate and prolonged periods of operations below 95%.

Clad stress analysis has also shown that a 1 kW/ft rapid power increment increase should not cause failure in the flattened section. This is based on a clad axial stress of 7000 psi per kW/ft rod average power increase which in turn gives rise to a bending stress of 30,000 psi per kW/ft average rod power change at the flattened section.

An overpower accident transient, such as a rod bank withdrawal, can cause rapid power increments in excess of 1 kW/ft before the high power trip terminates the transient. The average power rod (5.56 kW/ft) will have approximately a 1 kW/ft increment during this power transient. Fuel rods with flattened sections mostly have power levels in excess of the average power rod. Therefore, it is conservatively assumed that all flattened fuel rods will exhibit cladding defects during an accidental overpower transient.

References

- (1) FSAR, Section 14 and WCAP-8114
- (2) FSAR, Section 7.3
- (3) WCAP-8115, Section 6.7
- (4) WCAP-8115, Section 6.5

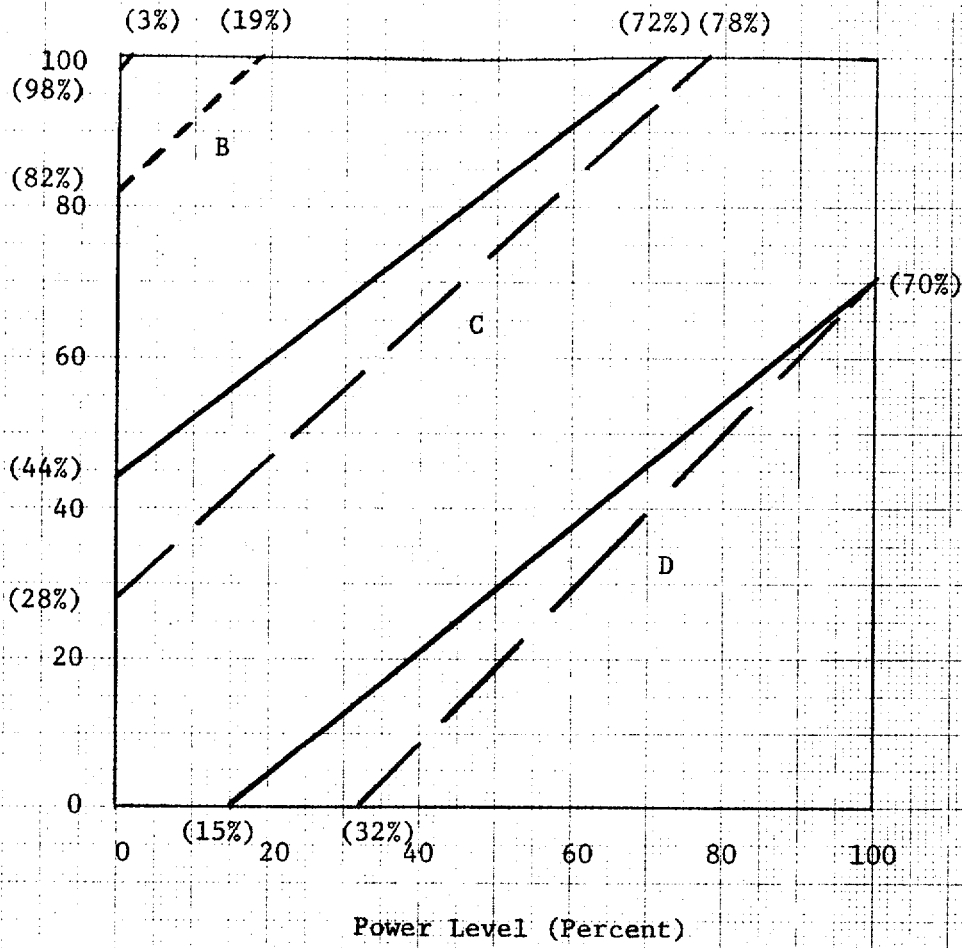
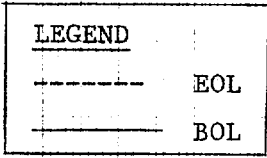


FIGURE 3.10-1

CONTROL GROUP INSERTION LIMITS FOR THREE LOOP OR TWO LOOP OPERATION

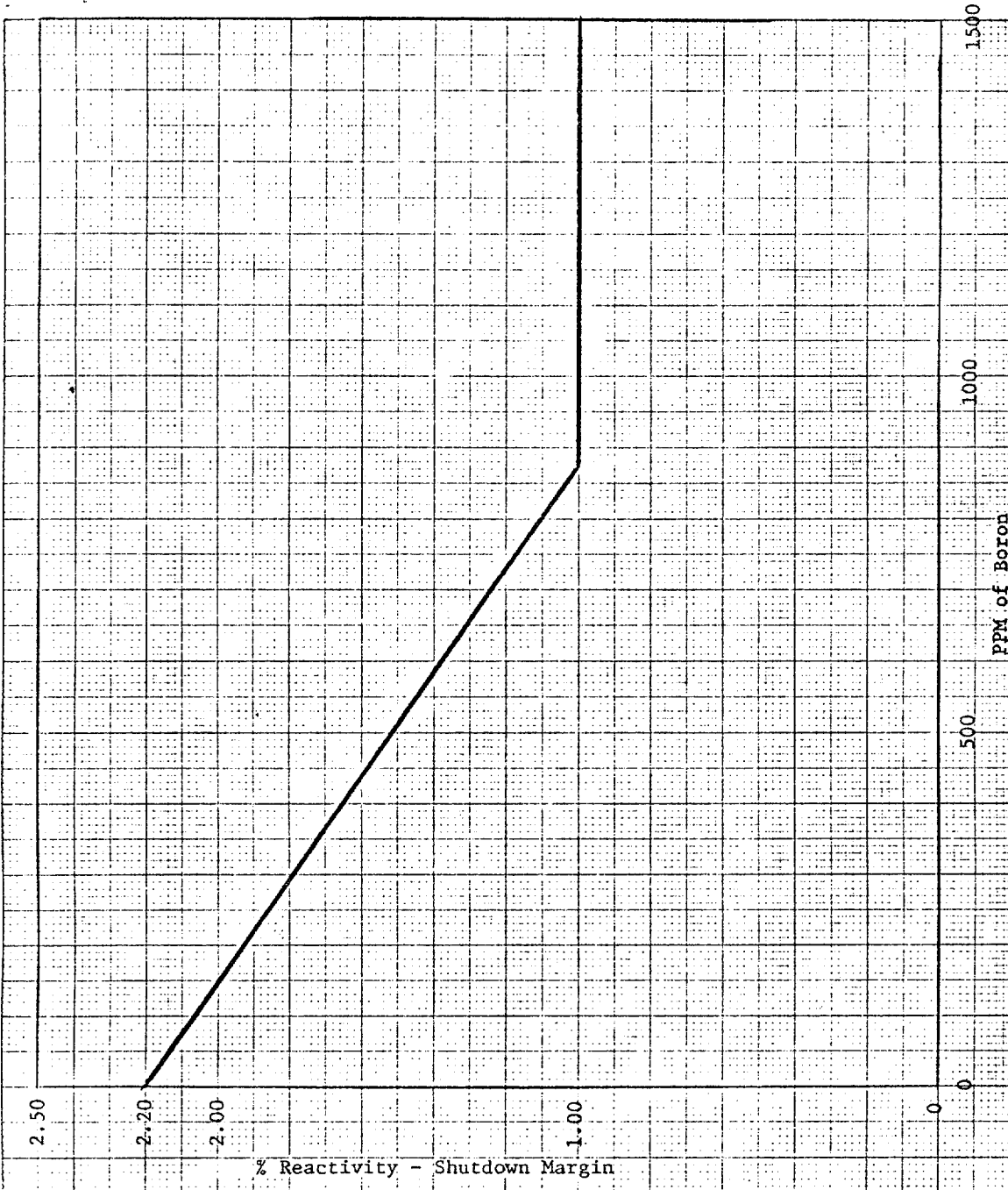


FIGURE 3.10-2

REQUIRED SHUTDOWN MARGIN VS REACTOR COOLANT BORON CONCENTRATION

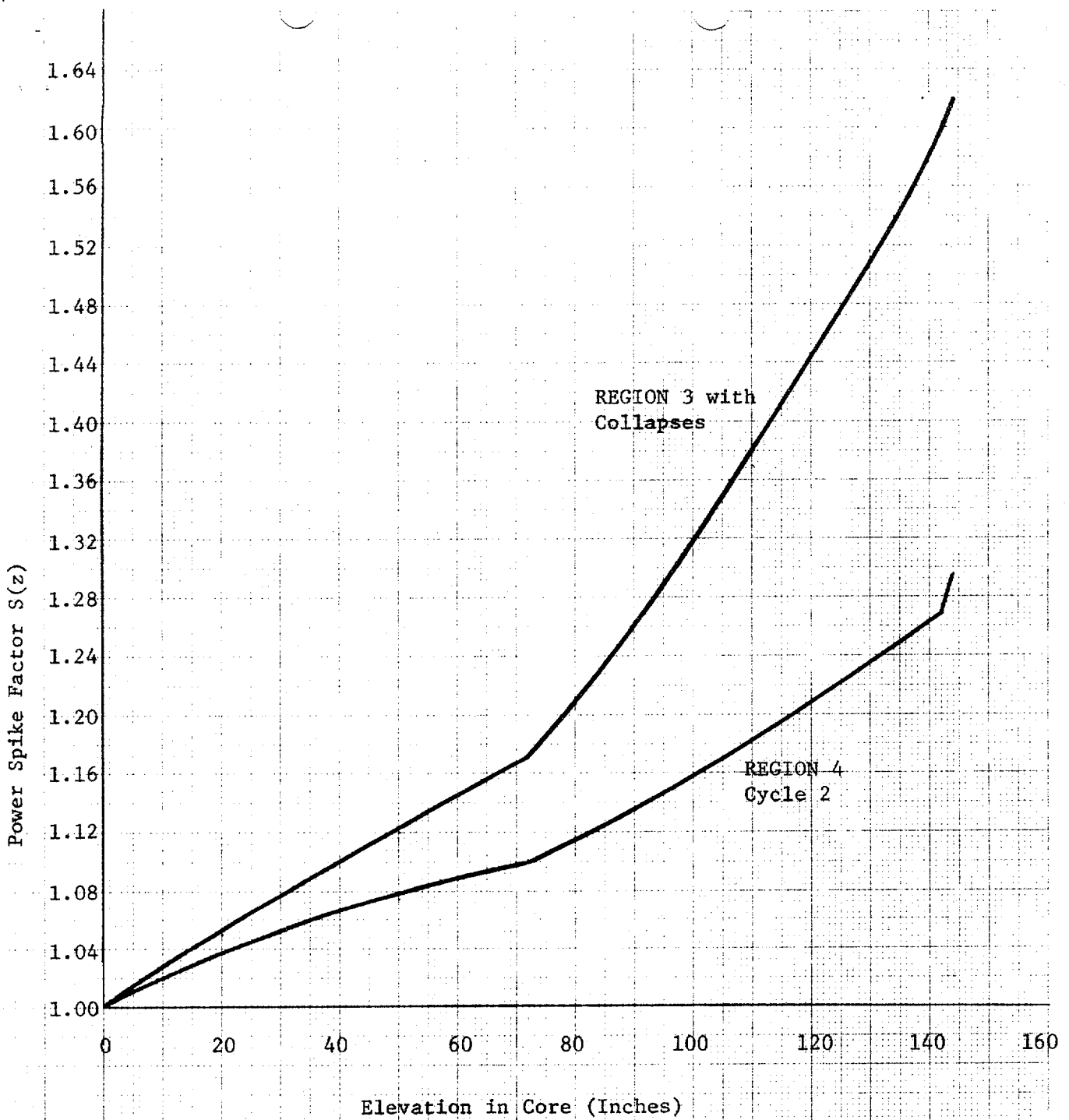


FIGURE 3.10-3

POWER SPIKE PENALTY FACTOR FOR CYCLE 2 INCLUDING EFFECT OF VARIABLE SPECIFIC POWER LIMIT

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability:

Applies to the operability of the movable detector instrumentation system.

Objective:

To specify functional requirements on the use of the in-core instrumentation systems, for the recalibration of the excore symmetrical off-set detection system.

- 3.11.1 A minimum of 16 total accessible thimbles and at least 2 per quadrant and sufficient movable in-core detectors shall be operable during recalibration of the excore symmetrical off-set detection system.
- 3.11.2 Power shall be limited to 90% of rated power for 3 loop or 40% of rated power for 2 loop operation if recalibration requirements for excore symmetrical off-set detection system identified in Table 4.1-1 are not met.

Basis:

The Movable In-Core Instrumentation System ⁽¹⁾ has five drives, five detectors, and 46 thimbles in the core. Each detector can be routed to twenty or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

To calibrate the excore detectors system, it is only necessary that the Movable In-Core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

TABLE 4.1-2

FREQUENCIES FOR SAMPLING TESTS

	<u>Check</u>	<u>Frequency</u>	<u>Maximum Time Between Tests</u>
1. Reactor Coolant Samples	Gross Activity (1)	5 day/week	3 days
	Radiochemical (2)	Monthly	45 days
	\bar{E} Determination	Semiannually (3)	30 weeks
	Tritium Activity	Weekly	10 days
	Cl & O ₂	5 day/week	3 days
2. Reactor Coolant Boron	Boron concentration	Twice/week	5 days
3. Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days
4. Boric Acid Tank	Boron concentration	Twice/week	5 days
5. Boron Injection Tank	Boron concentration	Weekly (6)	10 days
6. Spray Additive Tank	NaOH concentration	Monthly	45 days
7. Accumulator	Boron concentration	Monthly	45 days
8. Spent Fuel Pit	Boron concentration	Prior to Refueling	NA*
9. Secondary Coolant	Iodine-131	Weekly (5)	10 days
10. Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly (4)	10 days
11. Steam Generator Samples	Primary to secondary tube leakage	5 days/week	3 days

-
- (1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of uCi/cc.
- (2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.
- (3) \bar{E} determination will be started when the gross activity analysis indicates ≥ 10 uCi/cc and will be redetermined if the primary coolant gross radioactivity changes by more than 10 uCi/cc in accordance with Specification 3.1.4.

- (4) When iodine or particulate radioactivity levels exceed 10% of the limit in Specification 3.9.2.1, the sampling frequency shall be increased to a minimum of once each day.
- (5) When the iodine-131 activity exceeds 10% of the limit in Specification 3.4.2, the sampling frequency shall be increased to a minimum of once each day.
- (6) The boron concentration in the boron injection tank shall be checked immediately after any actuation of the safety injection system that might result in dilution of the boron concentration in the boron injection tank.

NA* - Not applicable

4.11 REACTOR CORE

Applicability:

Applies to surveillance of the reactor core.

Objective:

To insure the integrity of the fuel cladding.

Specification:

- 4.11.1 The power distribution shall be mapped monthly when the reactor is operated between 75% and 94.8% rated power. Before operating above 94.8% rated power, maps will be taken with the control rods in the D Bank at the preselected position, plus or minus five steps, that they will occupy when the reactor is operated at power levels greater than 94.8% rated power. The last six maps with this configuration will be used to determine the values of \bar{R} and σ . Above 94.8% rated power, the power distribution shall be mapped at least every 2 weeks. For a new preselected position of the control rods in the D Bank, maps taken within plus or minus five steps of the new preselected position must be used to determine new values of \bar{R} and σ . The new preselected position must be within 10 steps of the previous preselected position. The maximum power level attained during the mapping interval will determine the minimum mapping frequency. Each map will be based on flux traverses obtained from 36 or more of the 46 monitoring channels.
- 4.11.2 Axial surveillance of $F(z)$ $S(z)$ during Cycle 2 shall consist of:
- a. Traverses with the movable incore detectors in appropriate pairs of detector paths shall be taken every eight hours, or a frequency of approximately 0, 10, 30, 60, 120, 180, 240, 360, and 480 minutes following accumulated control rod motion in any one direction of five steps or more, exclusive of control rod movements within 15 steps from the top of the core. If the Axial Power Distribution Monitoring System is out of service, reactor operation at power can be continued for seven days provided that traverses are taken manually at equivalent frequencies. From the traverses, determination of $F(z)$ $S(z)$ shall be made and shown to result in an F_q^N less than the value specified in 3.10.2.1.1.
 - b. Surveillance limits on $F(z)$ $S(z)$ will be based on the most limiting core region, assuring that all F_q^N limits will be met.
- 4.11.3 The following criteria will be used for selecting the channels for measuring $F(z)$:

- a. For full core map, i , channels, j , are acceptable if:

$$[F_{ij}(z) S(z)]_{\max} \geq [\bar{F}_i(z) S(z)]_{\max}$$

The channel is not acceptable if it contains a Bank D control rod.

- b. For the latest full core map, i , channels, j , are acceptable if:

$$\left| R_{ij} - \bar{R}_j \right| \leq 2\sigma_{ji}$$

- 4.11.4 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% of rated power.

Basis:

The \bar{R} technique provides a means for using many of the monitoring thimbles to determine F_q^N without fully mapping the core. Frequent core maps assure that appropriate values of \bar{R} are being used for each thimble. Further, core mapping in the upper part of the regime where the F_q^N correlation to axial offset is used to provide assurance that fuel cladding is not jeopardized provides confidence that the correlation remains valid.

SEP 24 1973

Docket No. 50-261

Carolina Power & Light Company
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Change No. 23
License No. DPR-23

Gentlemen:

By letter dated June 15, 1973, you submitted an application for changes to the Technical Specifications appended to License No. DPR-23, as amended, for Unit No. 2 at the H. B. Robinson Steam Electric Plant. The proposed changes would eliminate surveillance requirements of: (1) power range symmetric off set in all conditions, except power operation; (2) protection logic circuits in the refueling and cold shutdown conditions; (3) reactor coolant system leakage in the cold shutdown and refueling conditions; (4) turbine steam stop, control, reheat stop, and intercept valves in all conditions, except power operations; and (5) portions of the safety injection system and the steam driven auxiliary feedwater pump in the refueling and cold shutdown conditions.

During our review of your application, we informed your staff that certain changes to your proposal were necessary to meet our regulatory requirements. These changes have been made. We have designated our action as Change No. 23.

We have reviewed your proposed change and conclude that none of the systems or components, exempted from testing under the specified conditions, are required for safe operation or shutdown of the plant under those conditions.

We conclude that Change No. 23 does not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. Pursuant to 10 CFR Part 50, Section 50.59, the Technical Specifications appended to License No. DPR-23 are changed as shown in Attachment A.

Sincerely,

Original signed by
Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosure: see next page

Enclosures:

Attachment A - Change No. 23 to
the Technical Specifications

cc w/enclosures:

George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
and Madden

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ATTACHMENT A

CHANGE NO. 23 TO THE TECHNICAL SPECIFICATIONS

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

1. Section 4.0 - Add the following after the first sentence of this paragraph:

"Performance of any surveillance test outlined in these specifications is not required when the system or component is out of service as permitted by the ~~Limiting~~ Conditions for Operation. Prior to returning the system to service, the specified calibration and testing surveillance shall be performed."

2. Table 4.1-1, Item No. 1, Note (3) - Change Note (3) to read:

"Upper and lower chambers for symmetric offset: monthly during power operations. When periods of reactor shutdown extend this interval beyond one month, the calibration shall be performed immediately following return to power."

3. Table 4.1-1, Item 27 - Under the column entitled "Test" add Note (1) as follows:

"(1) During hot shutdown and power operations. When periods of reactor cold shutdown and refueling extend this interval beyond one month, the test shall be performed prior to startup."

4. Table 4.1-3, Item No. 9 - Under the column entitled "Frequency" change the requirement to read:

"Daily when the reactor coolant system is above cold shutdown condition."

5. Table 4.1-3, Item No. 12 - Under the column entitled "Frequency" change the requirement to read:

"Monthly during power operation and prior to startup."

6. Section 4.5.2.1 - Add to the First sentence:

". . . when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed prior to heatup."

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7. Section 4.5.2.3 - Add to the first sentence:

". . . when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed prior to reactor heatup."

8. Section 4.5 Basis - Add the following to the next to the last paragraph:

"Monthly testing of the safety injection pumps, residual heat removal pumps, containment spray pumps and the boron injection tank isolation valves is not required when in the cold shutdown condition. These components are not required for plant safety when the reactor is in cold shutdown and testing during this condition will result in unnecessary wear on the equipment."

9. Section 4.8.2 - Add to the first sentence:

". . . when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed immediately following reactor heatup."

10. Section 4.8 Basis - Add the following sentence:

"Testing of the steam turbine auxiliary feedwater pump is not required during periods of cold shutdown when steam is not available. In this condition the pump is not required for plant safety."

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Docket No. 50-261

Carolina Power & Light Company
ATTN: E. E. Utley, Vice President
Bulk Power Supply Department
336 Fayetteville Street
Raleigh, North Carolina 27602

Change No. 23
License No. DPR-23

Gentlemen:

By letter dated June 15, 1973, you submitted an application for changes to the Technical Specifications appended to License No. DPR-23, as amended, for Unit No. 2 at the H. B. Robinson Steam Electric Plant. The proposed changes would eliminate surveillance requirements of: (1) power range symmetric off set in all conditions, except power operation; (2) protection logic circuits in the refueling and cold shutdown conditions; (3) reactor coolant system leakage in the cold shutdown and refueling conditions, turbine steam stop, control, reheat stop, and intercept valves in all conditions, except power operations; and (4) portions of the safety injection system and the steam driven auxiliary feedwater pump in the refueling and cold shutdown conditions.

During our review of your application, we informed your staff that certain changes to your proposal were necessary to meet our regulatory requirements. These changes have been made. We have designated our action as Change No. 23.

We have reviewed your proposed change and conclude that none of the systems or components are required for safe operation or shutdown of the plant under the conditions that they are exempted from testing.

We conclude that Change No. 23 does not present a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered. Pursuant to 10 CFR Part 50, Section 50.59, the Technical Specifications appended to License No. DPR-23 are changed as shown in Attachment A.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

OFFICE ►
Enclosure: see next page

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