February 4, 1994

Docket Nos. 50-280
and 50-281

DISTRIBUTION:
See next page

Mr. W. L. Stewart
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONTROL ROD URGENT ALARM FAILURE (TAC NOS. M86087 AND M86088)

The Commission has issued the enclosed Amendment No. 186 to Facility Operating License No. DPR-32 and Amendment No. 186 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated March 19, 1993, as supplemented December 9, 1993.

These amendments address plant operation with a control rod urgent alarm failure, a change in the control rod assembly partial movement surveillance test frequency, and proposed administrative changes.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
(Original Signed By)

Bart C. Buckley, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 186 to DPR-32
2. Amendment No. 186 to DPR-37
3. Safety Evaluation

cc w/enclosures: See next page

*Previously Concurred
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OFC	:LA:PDII-2	:PM:PDII-2	:D:PDII-2	:OGC	:	:
NAME	:ETana*	:BBuckley*	:HBerkow*	:JMoore*	:	:
DATE	:01/13/94	:01/13/94	:01/13/94	:01/24/94	:	:

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subject to change see note.

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NAME	:ETana	:BBuckley	:HBerkow	: JMoore	:	:
DATE	: 1/13/94	: 1/13/94	: 1/13/94	: 1/24/94	:	:

Mr. W. L. Stewart
Virginia Electric and Power Company

Surry Power Station

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DATED: February 4, 1994

AMENDMENT NO. 186 TO FACILITY OPERATING LICENSE NO. DPR-32 - SURRY UNIT 1
AMENDMENT NO. 186 TO FACILITY OPERATING LICENSE NO. DPR-37 - SURRY UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
E. Tana (2)
B. Buckley
OGC
D. Hagan, 3302 MNBB
G. Hill P-137 (4)
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFMB
M. Sinkule, R-II

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated March 19, 1993, as supplemented December 9, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 186 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 4, 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 186
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Virginia Electric and Power Company (the licensee) dated March 19, 1993, as supplemented December 9, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 186 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 4, 1994

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 186 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 186 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 3.12-1 thru 3.12-19	TS 3.12-1 thru 3.12-20
TS Table 3.12-1	TS 3.12-21
TS Figure 3.12-2	TS Figure 3.12-2
TS Figure 3.12-3	TS Figure 3.12-3
TS Figure 3.12-4A	- - -
TS Figure 3.12-4B	- - -
TS Figure 3.12-5	- - -
TS Figure 3.12-6	- - -
TS Figure 3.12-7	- - -
TS Figure 3.12-8	- - -
TS Figure 3.12-9	- - -
TS Figure 3.12-10	- - -
TS 4.1-9b	TS 4.1-9b

3.12 CONTROL ROD ASSEMBLIES AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rod assemblies and power distribution limits.

Objective

To ensure core subcriticality after a reactor trip, a limit on potential reactivity insertions from hypothetical control rod assembly ejection, and an acceptable core power distribution during power operation.

Specification

A. Control Bank Insertion Limits

1. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, the shutdown control rod assemblies shall be fully withdrawn. With a shutdown control rod assembly not fully withdrawn, within 1 hour either fully withdraw the assembly or declare the assembly inoperable and apply Specification 3.12.C.

2. Whenever the reactor is critical, except for physics tests and control rod assembly surveillance testing, the full length control banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A or 3.12-1B. With a control bank inserted beyond the limits shown in the applicable figure, restore the control rod assembly bank to within its limits within 2 hours, or reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED POWER which is allowed by the group position using TS Figures 3.12.1A or 1B, or place the reactor in HOT SHUTDOWN within 6 hours.

3. The limits shown on TS Figures 3.12-1A and 1B may be revised on the basis of physics calculations and physics data obtained during unit startup and subsequent operation, in accordance with the following:

a. The sequence of withdrawal of the control banks, when going from zero to 100% power, is A, B, C, D.

b. An overlap of control banks, consistent with physics calculations and physics data obtained during unit startup and subsequent operation, will be permitted.

c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation conditions, except for physics tests, 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 ($T_{avg} \geq 547°F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon or boron.

4. Whenever the reactor is subcritical, except for physics tests, the critical control rod assembly position, i.e., the control rod assembly position at which criticality would be achieved if the control rod assemblies were withdrawn in normal sequence with no other reactivity changes, shall not be lower than the insertion limit for zero power.

5. Insertion limits do not apply during physics tests or during periodic surveillance testing of control rod assemblies. However, the shutdown margin indicated above must be maintained except for the LOW POWER PHYSICS TEST to measure control and shutdown bank worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod assembly, expected to have the highest worth, inserted.

6. With a maximum of one control or shutdown bank inserted beyond the insertion limit specified in Specification 3.12.A.2 during control rod assembly testing pursuant to Specification 4.1, and immovable due to a failure of the Rod Control System, POWER OPERATION may continue* provided that:

   a. the affected bank insertion is limited to 18 steps below the insertion limit as measured by the group step counter demand position indicators,

   b. the affected bank is trippable,

   c. each control rod assembly is aligned to within ± 12 steps of its respective group step counter demand position indicator,

   d. The shutdown margin requirement of Specification 3.12.A.3.c is determined to be met at least every 12 hours thereafter, and

   e. the affected bank is restored to within the insertion limits of Specification 3.12.A within 72 hours.

   Otherwise place the unit in HOT SHUTDOWN within the next 6 hours.

B. Power Distribution Limits

1. At all times except during LOW POWER PHYSICS TESTS, the hot channel factors defined in the basis must meet the following limits:

   $F_Q(Z) \leq 2.32/P \times K(Z)$ for $P > 0.5$

   $F_Q(Z) \leq 4.64 \times K(Z)$ for $P \leq 0.5$

   $F^N_{\Delta H} \leq 1.56\,[1 + 0.3\,(1-P)]$ for three loop operation

   where P is the fraction of RATED POWER at which the core is operating, K(Z) is the function given in TS Figure 3.12-2, and Z is the core height location of F_Q.

* Provision for continued operation does not apply to Control Bank D inserted beyond the insertion limit.

2. Prior to exceeding 75% of RATED POWER following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

   a. The measurement of total peaking factor F_Q^{Meas} shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor $F_{\Delta H}^N$ shall be compared directly to the limit specified in Specification 3.12.B.1. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits of $F_Q(Z) \leq 2.32/P \times K(Z)$ and $F_{\Delta H}^N \leq 1.56$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced within the next 4 hours.

   b. The provisions of Specification 4.0.4 are not applicable.

3. The reference equilibrium indicated axial flux difference (called the target flux difference) at a given power level P_O is that indicated axial flux difference with the core in equilibrium xenon conditions (small or no oscillation) and the control rod assemblies more than 190 steps withdrawn. The target flux difference at any other power level P is equal to the target value at P_O multiplied by the ratio P/P_O. The target flux difference shall be measured at least once per equivalent full power quarter. The target flux difference must be updated during each effective full power month of operation either by actual measurements or by linear interpolation using the most recent value and the value predicted for the end of the cycle life The provisions of Specification 4.0.4 are not applicable.

4. Except as modified by Specifications 3.12.B.4.a, b, c, or d below the indicated axial flux difference shall be maintained within a ± 5% band about the target flux difference (defines the target band on axial flux difference).

a. At a power level greater than 90 percent of RATED POWER, if the indicated axial flux difference deviates from its target band, within 15 minutes either restore the indicated axial flux difference to within the target band or reduce the reactor power to less than 90 percent of RATED POWER.

b. At a power level less than or equal to 90 percent of RATED POWER,

(1) The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference is within the limits shown on TS Figure 3.12-3. One minute penalty is accumulated for each one minute of operation outside of the target band at power levels equal to or above 50% of RATED POWER.

(2) If Specification 3.12.B.4.b.(1) is violated, then the reactor power shall be reduced to less than 50% power within 30 minutes and the high neutron flux setpoint shall be reduced to less than or equal to 55% power within the next four hours.

(3) A power increase to a level greater than 90 percent of RATED POWER is contingent upon the indicated axial flux difference being within its target band.

(4) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to TS Table 4.1-1 provided the indicated axial flux difference is maintained within the limits of TS Figure 3.12-3. A total of 16 hours of operation may be accumulated with the axial flux difference outside of the target band during this testing without penalty deviation.

c. At a power level less than or equal to 50 percent of RATED POWER,

(1) The indicated axial flux difference may deviate from its target band.

(2) A power increase to a level greater than 50 percent of RATED POWER is contingent upon the indicated axial flux difference not being outside its target band for more than one hour accumulated penalty during the preceding 24-hour period. One half minute penalty is accumulated for each one minute of operation outside of the target band at power levels between 15% and 50% of RATED POWER.

d. The axial flux difference limits for Specifications 3.12.B.4.a, b, and c may be suspended during the performance of physics tests provided:

(1) The power level is maintained less than or equal to 85% of RATED POWER, and

(2) The limits of Specification 3.12.B.1 are maintained. The power level shall be determined to be less than or equal to 85% of RATED POWER at least once per hour during physics tests. Verification that the limits of Specification 3.12.B.1 are being met shall be demonstrated through in-core flux mapping at least once per 12 hours.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in Specification 3.12.B.4.a and the flux difference time limits in Specifications 3.12.B.4.b and c. If the alarms are out of service temporarily, the axial flux difference shall be logged and conformance to the limits assessed every hour for the first 24 hours and half-hourly thereafter. The indicated axial flux difference for each excore channel shall be monitored at least once per 7 days when the alarm is OPERABLE and at least once per hour for the. first 24 hours after restoring the alarm to OPERABLE status.

5. The allowable QUADRANT POWER TILT is 2.0%.

6. If, except for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then:

   a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or

   b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

   c. If the QUADRANT POWER TILT exceeds ± 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

7. If, except for physics and control rod assembly surveillance testing, after a further period of 24 hours, the QUADRANT POWER TILT in Specification 3.12.B.5 above is not corrected to less than 2%:

   a. If the design hot channel factors for RATED POWER are not exceeded, an evaluation as to the cause of the discrepancy shall be made and a special report issued to the Nuclear Regulatory Commission.

   b. If the design hot channel factors for RATED POWER are exceeded and the power is greater than 10%, then the high neutron flux, Overpower ΔT, and Overtemperature ΔT trip setpoints shall be reduced 1% for each percent the hot channel factor exceeds the RATED POWER design values within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

c. If the hot channel factors are not determined, then the Overpower ΔT and Overtemperature ΔT trip setpoints shall be reduced by the equivalent of 2% power for every 1% QUADRANT POWER TILT within the next 4 hours, and the Nuclear Regulatory Commission shall be notified.

C. Control Rod Assemblies

1. To be considered OPERABLE during startup and POWER OPERATION each control rod assembly shall :
   1) be trippable,
   2) aligned within ± 24 steps of its group step demand position during the "Thermal Soak" period, as defined in Section 3.12.E.1.b, or ± 12 steps otherwise during power operation, and
   3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

2. To be considered OPERABLE during shutdown modes, each control rod assembly shall:
   1) be trippable,
   2) have its rod position indicator capable of verifying rod movement upon demand, and
   3) have a drop time of less than or equal to 2.4 seconds to dashpot entry.

3. Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:
   a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or

   b. the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that:
      1) either:

(a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or

(b) the remainder of the control rod assemblies in the group with the inoperable control rod assembly are aligned to within 12 steps of the inoperable rod within one (1) hour while maintaining the control rod assembly sequence and insertion limits of Figure 3.12-1A and B; the THERMAL POWER level shall be restricted pursuant to Specification 3.12.A during subsequent operation.

2) the shutdown margin requirement of Specification 3.12.A.3.c is determined to be met within one hour and at least once per 12 hours thereafter.

3) the hot channel factors are shown to be within the design limits of Specification 3.12.B.1 within 72 hours. Further, it shall be demonstrated that the value of Fxy(Z) used in the Constant Axial Offset Control analysis is still valid.

4) a reevaluation of each accident analysis of Table 3.12-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions.

5) If power has been reduced in accordance with Specification 3.12.C.3.b, power may be increased above 75% of RATED POWER provided that:

   (a) an analysis has been performed to determine the hot channel factors and the resulting allowable power level based on the limits of Specification 3.12.B.1, and

   (b) an evaluation of the effects of operating at the increased power level on the accident analyses of Table 3.12-1 has been completed.

4. With more than one inoperable control rod assembly, as defined in Specification 3.12.C.1, determine within 1 hour that the shutdown margin requirement of Specification 3.12.A.3.c is satisfied and be in HOT SHUTDOWN within 6 hours.

5. The provisions of Specifications 3.12.C.1 and 3.12.C.4 shall not apply during LOW POWER PHYSICS TESTS in which the control rod assemblies are intentionally misaligned.

D. QUADRANT POWER TILT

1. If the reactor is operating above 75% of RATED POWER with one excore nuclear channel out of service, the QUADRANT POWER TILT shall be determined:
   a. Once per day, and
   b. After a change in power level greater than 10% or more than 30 inches of control rod motion.

2. The QUADRANT POWER TILT shall be determined by one of the following methods:
   a. Movable detectors (at least two per quadrant)
   b. Core exit thermocouples (at least four per quadrant)

E. Rod Position Indication System

1. Rod position indication shall be provided as follows:

    a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

    b. From movement of control banks to achieve criticality up to 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 24 steps of their respective group step demand counter indications for a maximum of one hour out of twenty-four, and to within ± 12 steps otherwise. During the one-hour "Thermal Soak" period, the step demand counters shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps.

    c. In HOT, INTERMEDIATE, and COLD SHUTDOWN, the step demand counters shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps. The rod position indicators shall be available to verify control rod assembly movement upon demand.

2. If a rod position indicator channel is inoperable, then:

    a. For operation above 50% of RATED POWER, the position of the control rod assembly shall be checked indirectly using the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating control rod assembly exceeding 24 steps, or

    b. Reduce power to less than 50% of RATED POWER within 8 hours. During operations below 50% of RATED POWER, no special monitoring is required.

3. If more than one rod position indicator channel per group or two rod position indicator channels per bank are inoperable during control bank motion to achieve criticality or POWER OPERATION, then the unit shall be placed in HOT SHUTDOWN within 6 hours.

F. <u>DNB Parameters</u>

1. The following DNB related parameters shall be maintained within their limits during POWER OPERATION:

   - Reactor Coolant System T_{avg} ≤ 578.4°F
   - Pressurizer Pressure ≥ 2205* psig
   - Reactor Coolant System Total Flow Rate ≥ 273,000 gpm

   a. The Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

   b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED POWER within the next 4 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED POWER per minute or a THERMAL POWER step increase in excess of 10% of RATED POWER.

* ≥ 2105 psig for Unit 2 Cycle 12 operation at Reactor Coolant System nominal operation pressure of 2135 psig.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to COLD SHUTDOWN) are compensated for by changes in the soluble boron concentration. During POWER OPERATION, the shutdown control rod assemblies are fully withdrawn and control of power is by the control banks. A reactor trip occurring during POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly 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 control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit 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 the analyses of the hypothetical steam break accident. The control rod assembly insertion limits are based on end of core life conditions. The shutdown margin for the entire cycle length is established at 1.77% reactivity. Other accident analyses with the exception of the Chemical and Volume Control System malfunction analyses are based on 1% reactivity shutdown margin. Relative positions of control banks are determined by a specified control bank overlap. This overlap is based on the consideration of axial power shape control. The specified control rod assembly insertion limits have been established to limit the potential ejected control rod assembly worth in order to account for the effects of fuel densification. The various control rod assemblies (shutdown banks, control banks A, B, C, and D) are each to be moved as a bank; that is, with each assembly in the bank within one step (5/8 inch) of the bank position.

Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks, and a linear position indicator, Linear Variable Differential Transformer, which indicates the actual assembly position. The position indication accuracy of the Linear Variable Differential Transformer is approximately ±5% of span (±12 steps) under steady state conditions. The relative accuracy of the linear position indicator has been considered in establishing the maximum allowable deviation of a control rod assembly from its indicated group step demand position. In the event that the linear position indicator is not in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (full length control rod assembly 12 feet out of alignment with its bank), operation at 50% steady state power does not result in exceeding core limits.

The "Thermal Soak" allowance below 50% power, during which the Rod Position Indication System tolerance requirement is relaxed, provides time for the system to reach thermal equilibrium. A total of one hour in twenty-four is available for this allowance, which may be a continuous hour or may consist of discrete, shorter intervals. For such a short period of time, a misaligned control rod assembly does not pose an unacceptable risk. At these conditions, the rod position indicators should still be used to verify rod movement but not their exact location. The tolerance is tightened after one hour to ensure that the thermal overshoot does not conceal an actual control rod assembly misalignment.

The reliance upon the step demand counters at HOT and COLD SHUTDOWN shifts the monitoring of control rod assembly position from the Rod Position Indication System to the more reliable demand counters when Reactor Coolant System temperature is changing greatly but the core remains subcritical. The step demand counters also provide precise group demand positions during the thermal soak period.

The specified control rod assembly drop time is consistent with safety analyses that have been performed.

An inoperable control rod assembly imposes additional demands on the operators. The permissible number of inoperable control rod assemblies 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 control rod assemblies upon reactor trip.

In the event that a failure of the Rod Control System renders control rod assemblies immovable, provision is made for continued operation provided:

- the affected control rod assemblies remain trippable,
- the individual control rod assembly alignment limits are met.

In the event that a failure of the Rod Control System renders control rod assembly banks immovable during control rod assembly surveillance testing, provision is made for 72 hours of continued operation provided:

- the affected control rod assemblies remain trippable,
- the individual control rod assembly alignment limits are met,
- a maximum of one control or shutdown bank is inserted no more than 18 steps below the insertion limit, and
- the shutdown margin requirements are verified every 12 hours during the period the insertion limit is not met.

The 72 hour provision does not apply to Control Bank D since insertion of D bank below the insertion limit is not required for control rod assembly surveillance testing.

Checks are performed for each reload core to ensure that this minor bank insertion will not result in power distributions which violate the Departure from Nucleate Boiling (DNB) criterion for ANS Condition II transient (moderate frequency transients analyzed in Section 14.2 of the UFSAR) during the repair period or in a violation of the shutdown margin requirements of Specification of 3.12.A.3.c during the repair period.

The 72 hour period for a control rod assembly bank to be inserted below its limit restricts the likelihood of a more severe (i.e., ANS Condition III or IV) accident or transient condition.

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 fuel centerline temperature must not exceed 4700°F. Second, the minimum DNB Ratio (DNBR) in the core must not be less than the applicable design limit in normal operation or in short term transients.

In addition to the above, the peak linear power density and the nuclear enthalpy rise hot channel factor must not exceed their limiting values which result from the large break loss of coolant accident analysis based on the Emergency Core Cooling System acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits of power distribution, the following hot channel factors are defined:

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerance on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux for non-statistical applications.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power for both loss of coolant accident and non-loss of coolant accident considerations.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and loss of coolant accident calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the Emergency Core Cooling System acceptance criteria as specified in 10 CFR 50.46 using the upper bound $F_Q(Z)$ times the hot channel factor normalized operating envelope given by TS Figure 3.12-2.

When an F_Q measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowance for measurement error for a full core map (greater than or equal to 38 thimbles, including a minimum of 2 thimbles per core quadrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of F_Q.

In the specified limit of $F^N_{\Delta H}$, there is a four percent allowance, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.56 [1 + 0.3 (1-P)]/1.04$. The 4% allowance is based on the considerations that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influence F_Q, can be compensated for by tighter axial control. An appropriate allowance for measurement uncertainty for $F^N_{\Delta H}$ obtained from a full core map (≥38 thimbles, including a minimum of 2 detectors per core quadrant, monitored) taken with the movable incore detector flux mapping system has been incorporated in the statistical DNBR limit.

Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor $F^N_{\Delta H}$ limit will be met. These conditions are as follows:

1. Control rod assemblies in a single bank move together with no individual control rod assembly insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a control rod assembly misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control banks are sequenced with overlapping banks as shown in TS Figures 3.12-1A and 1B.
3. The full length control bank insertion limits are not violated.
4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference 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 normalized power between the top and the bottom halves of the core.

The permitted relaxation in $F^N_{\Delta H}$ with decreasing power level allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, this hot channel factor limit is met.

A recent evaluation of DNB test data obtained from experiments of fuel rod bowing in thimble cells has identified that the reduction in DNBR due to rod bowing in thimble cells is more than completely accommodated by existing thermal margins in the core design. Therefore, it is not necessary to continue to apply a rod bow penalty to $F^N_{\Delta H}$.

The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but expressed as axial offset it varies only with burnup.

The technical specifications on power distribution control given in Specification 3.12.B.4 together with the surveillance requirements given in Specification 3.12.B.2 assure that the Limiting Condition for Operation for the heat flux hot channel factor is met.

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with the full length rod control bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating, is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of ±5% ΔI are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not always possible during certain physics tests or during excore detector calibrations. Therefore, the specifications on power distribution control are less restrictive during physics tests and excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band. However, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range ±13.8 percent (±10.8 percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rod assemblies to produce the required indicated flux difference.

A 2% QUADRANT POWER 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 rod assembly and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rod assemblies producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The limits of the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The measurement of the Reactor Coolant System Total Flow Rate once per refueling cycle is adequate to detect flow degradation.

TABLE 3.12-1

ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE CONTROL ROD ASSEMBLY

Control Rod Assembly Insertion Characteristics

Control Rod Assembly Misalignment

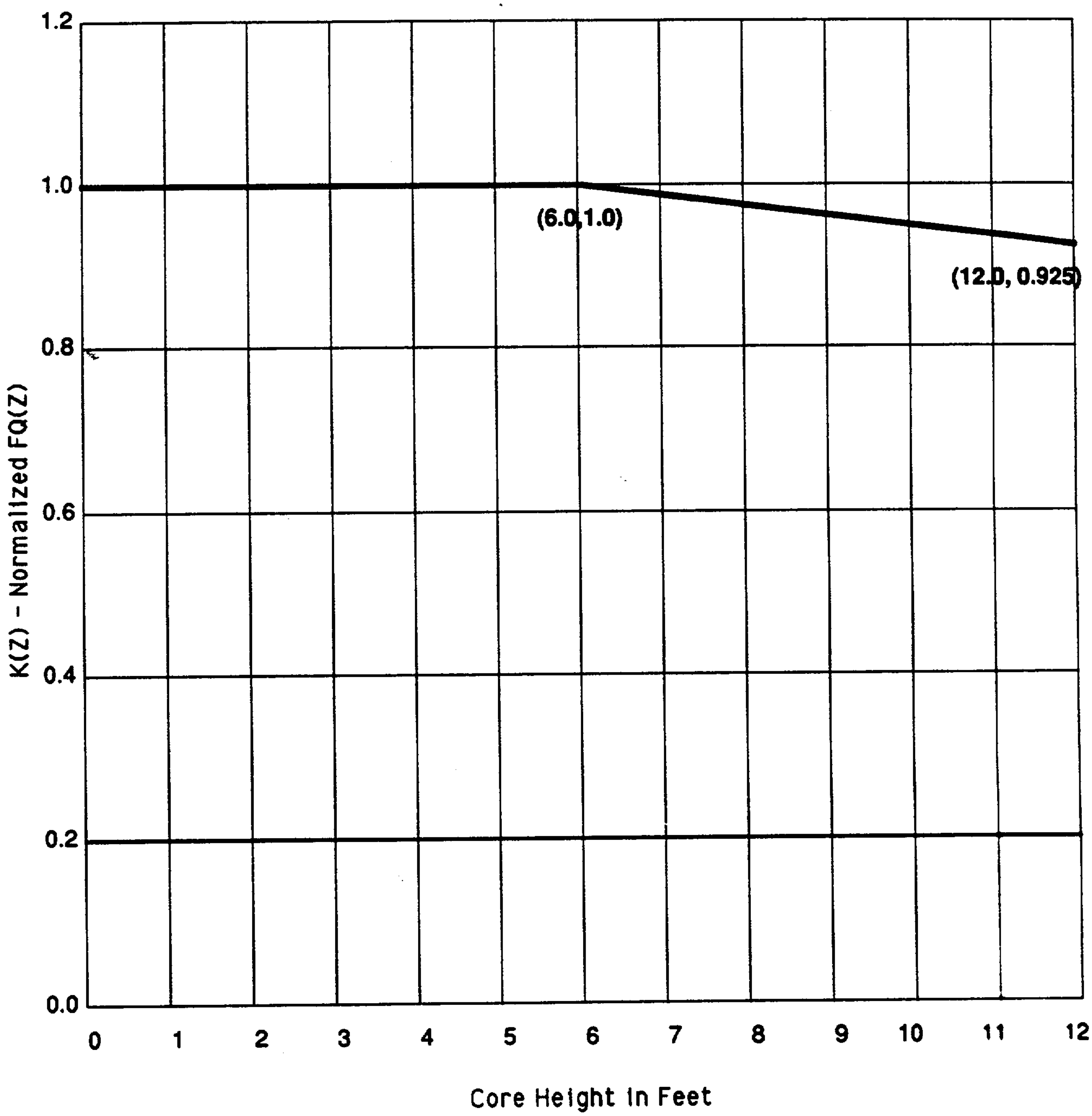
Large and Small Break Loss of Coolant Accidents

Single Reactor Coolant Pump Locked Rotor

Major Secondary Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing
(Control Rod Assembly Ejection)

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE



AXIAL FLUX DIFFERENCE LIMITS
AS A FUNCTION OF RATED POWER
SURRY POWER STATION

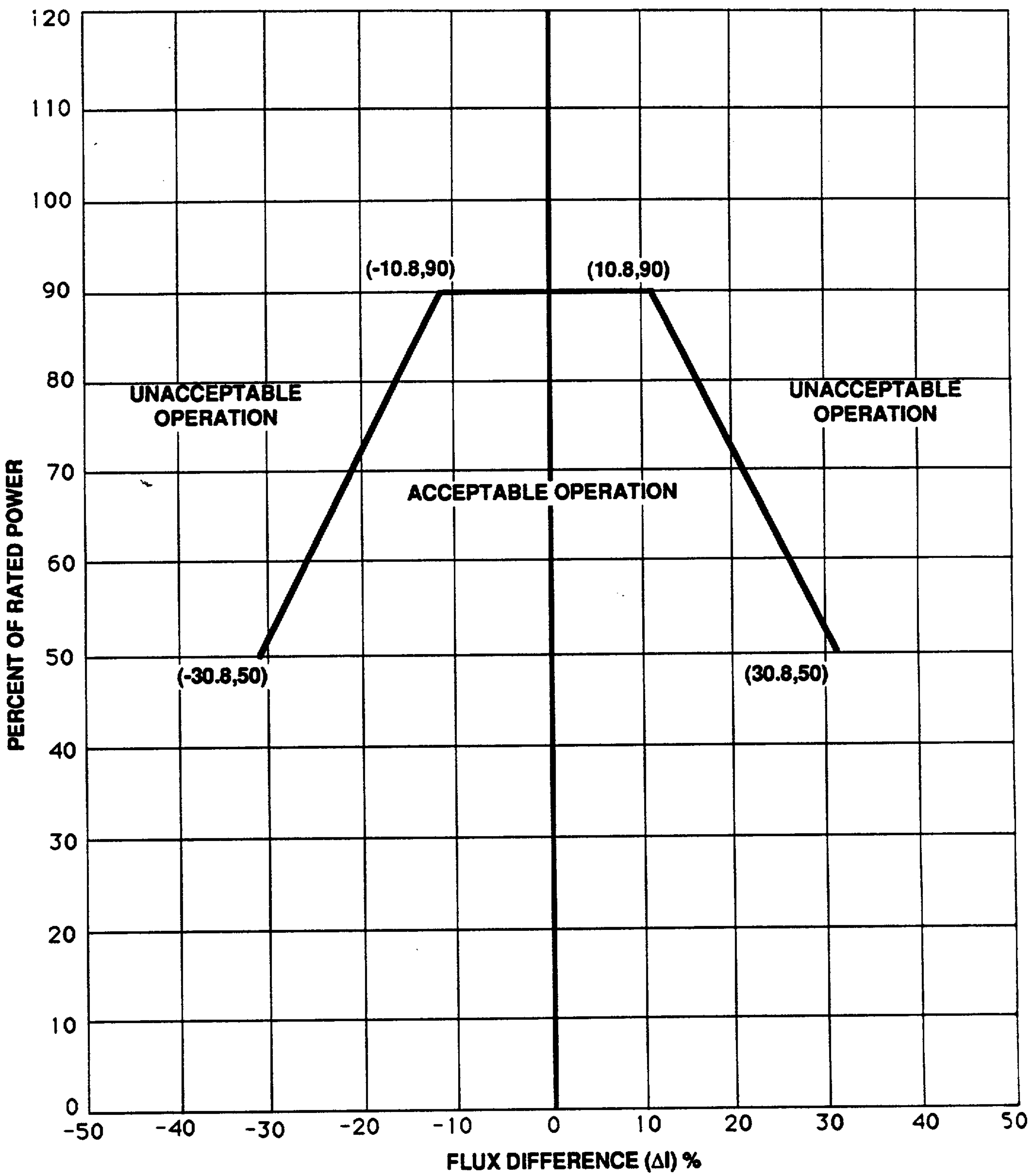


TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1.	Control Rod Assemblies	Rod drop times of all full length rods at hot conditions	Prior to reactor criticality: a. For all rods following each removal of the reactor vessel head b. For specially affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and c. Each refueling shutdown.	7
2.	Control Rod Assemblies	Partial movement of all rods	Monthly	7
3.	Refueling Water Chemical Addition Tank	Functional	Each refueling shutdown	6
4.	Pressurizer Safety Valves	Setpoint	Per TS 4.0.3	4
5.	Main Steam Safety Valves	Setpoint	Per TS 4.0.3	10
6.	Containment Isolation Trip	* Functional	Each refueling shutdown	5
7.	Refueling System Interlocks	* Functional	Prior to refueling	9.12
8.	Service Water System	* Functional	Each refueling shutdown	9.9
9.	Fire Protection Pump and Power Supply	Functional	Monthly	9.10
10.	Primary System Leakage	* Evaluate	Daily	4
11.	Diesel Fuel Supply	* Fuel Inventory	5 days/week	8.5
12.	Boric Acid Piping Heat Tracing Circuits	* Operational	Monthly	9.1
13.	Main Steam Line Trip Valves	Functional (Full Closure)	Before each startup (TS 4.7) The provisions of Specification 4.0.4 are not applicable	10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TECHNICAL SPECIFICATION CHANGE ADDRESSING OPERATION WITH A ROD

URGENT FAILURE ALARM

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATIONS UNITS 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated March 19, 1993, Virginia Electric and Power Company requested changes to the Technical Specifications (TS) for the Surry Power Station, Units 1 and 2. These changes address operation with a control rod urgent failure condition including limited operation with one control or shutdown bank inserted slightly below its insertion limit. A letter dated December 9, 1993, provided clarification of operation in the urgent failure condition. The December 9, 1993, submittal did not expand the scope of the original application and did not change the proposed no significant hazards determination.

Changes involving definition of actions and time limits for certain Limiting Conditions of Operation where none are currently defined are also added. In addition, administrative changes to provide consistency and readability are proposed. Finally, the control rod assembly partial movement surveillance test frequency is changed from biweekly to monthly.

The TS require periodic testing of each control and shutdown control rod assembly bank in the core during power operation to ensure that the control rod assemblies are trippable. This testing requires partial movement of each control rod assembly not fully inserted into the core. This is typically done at or near full power, one bank at a time. Current procedures call for sequential insertion and withdrawal of 18 steps for the bank being tested. Special test exceptions allow the rods to be inserted beyond their insertion limits for this test. The length of the test is not prescribed.

On several occasions, the Surry Power Station has experienced control rod urgent failure alarms during the control rod assembly surveillance testing. This alarm is indicative of an internal failure in the rod control equipment that has affected the ability of the system to move control rod assemblies. These failures have a number of causes and may take some time to diagnose. These failures in no way impact the trippability of the control rod assemblies.

With an urgent failure alarm, the present TS provide 2 hours for troubleshooting and repair and, if unsuccessful, the unit must be brought to hot shutdown in 6 hours. The proposed changes would allow up to 72 hours for troubleshooting and repair if the rod assembly exceeds the insertion limit.

2.0 TECHNICAL SPECIFICATION CHANGES

TS 3.12.A.6 supplements TS 3.12.A.5 by providing a limit on both time and insertion if a bank is immovable due to failures external to the control rod assembly drive mechanism. A maximum of one control or shutdown bank (with the exception of Control Bank D) may be inserted below its insertion limit for up to 72 hours during diagnosis and repair of the Rod Control System provided that:

1) the control or shutdown bank is inserted no more than 18 steps below the insertion limit as measured by the group step counter demand position indicators,

2) the affected bank is trippable,

3) each shutdown and control rod is aligned to within ± 12 steps of its respective group step counter demand position, and

4) the shutdown margin requirement of TS 3.12.A.3.c is determined to be met at least once per 12 hours.

TS 3.12.C.3 has been changed to treat control banks which cannot be moved by the Rod Control System as operable provided:

1) the affected banks are trippable, and

2) each control rod assembly in the affected bank is aligned to within ± 24 steps of its respective group step counter demand position during the "Thermal Soak" period and to within ± 12 steps otherwise.

TS Table 4.1-2A has been revised to change the frequency for the control rod assembly exercise test frequency from biweekly to monthly.

3.0 EVALUATION

The present TS 3.12.A.5 allows exemption from the insertion limits for physics testing and periodic exercise of individual control rod assemblies. The exemption for control rod assembly testing is necessary because insertion limits require shutdown banks and control banks A, B and C to be fully withdrawn for full power operation. TS 3.12.A.5 provides 2 hours for troubleshooting and repair, and if unsuccessful, the unit must be brought to hot shutdown in 6 hours. The 2-hour time limit does not allow sufficient time for diagnosis and repair and the licensee has had to request enforcement discretion in order to complete diagnosis and repair on several occassions.

The proposed TS 3.12.A.6 supplements TS 3.12.A.5 by defining a limit of both time and insertion if a bank is immovable due to failures external to the control rod assembly drive mechanism. A maximum of one control or shutdown bank(with the exception of Control Bank D) may exceed its insertion limits by no more than 18 steps for up to 72 hours for diagnosis and repair of the rod control system provided the bank is trippable and satisfaction of shutdown margin requirements is verified once per 12 hours. Concurrent control rod misalignment (misalignment of individual control rod assemblies from their group step counter demand position by more than ± 12 steps) is not allowed. Because of the misalignment constraints and the 18 step limit, the impact on core reactivity and power distribution is very small. In addition the shutdown margin is specifically reconfirmed every 12 hours and explicit analytical checks on the radial power distribution are performed as part of the reload safety evaluation process. Furthermore, if the affected bank is not restored to above the insertion limit within the allowed 72 hours, the unit must be placed in hot shutdown within the next 6 hours. This change will allow sufficient time for diagnosis and repairs while maintaining the safety function of the control rods, since the affected rods are still trippable. In addition, alignment must be maintained and shutdown margin will be checked.

Adding TS 3.12.A.6 is acceptable because:

1) all control and shutdown rod assemblies are trippable,
2) immovable rod assemblies exceed no more than 18 steps,
3) immovable rod assemblies are aligned,
4) shutdown margin is specifically reconfirmed every 12 hours,
5) explicit analytical checks of radial power distribution are performed as part of reload safety evaluation,
6) if rod assemblies are not restored to within insertion limits within 72 hours, the unit must be placed in hot shutdown within the next 6 hours.

The next proposed TS change deals with the definition of an operable control rod assembly. If more than one control rod assembly in a given bank is immovable due to a failure external to the control rod assembly drive mechanism but remains trippable, the current specification (TS 3.12.C.3) allows 2 hours to restore the affected control rod assemblies to operable status. The proposed change to TS 3.12.C defines control banks which cannot be moved as OPERABLE as long as they are trippable and each control rod assembly is aligned with the group step counter. While there is no time limit for correcting such a problem, the licensee has committed in a letter dated December 9, 1993 to take prompt corrective action to return the Control Rod Drive System to service and regain the normal plant control function provided by the control rods. This change in the definition of an OPERABLE control rod assembly is acceptable because rods which are trippable, above the insertion limits and within the analyzed alignment requirements are fully capable of performing their intended safety function, even if they cannot be moved by the Rod Control System.

Various changes pertaining to action times were proposed because the existing Surry Technical Specifications do not contain action statements for certain conditions, or contain action statements with no time limits. This creates the potential for unnecessary entry into TS 3.0.1 when the requirements of a limiting condition for operation are not met. These proposed changes are consistent with the Standard Technical Specifications and are, therefore, acceptable.

Several editorial changes were proposed to correct inconsistencies in capitalization of defined terms and operating mode names. Also, previously deleted figures are being removed and the remaining figures are being renumbered. The staff finds the proposed editorial changes acceptable.

Finally, the control rod assembly surveillance frequency is being changed from biweekly to monthly. A review of recent test experience (in excess of 4,000 individual control rod assemblies tested) revealed no instance of mechanically stuck control rod assemblies. This test experience supports the proposed relaxation. The staff finds this change acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and also changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (58 FR 28064). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: M. Chatterton

Date: February 4, 1994