Docket Nos. 50-280
and 50-281

JUN 1 8 1975

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Virginia Electric & Power Company
ATTN: Mr. Stanley Ragone
Senior Vice President
Post Office Box 26666
Richmond, Virginia 23261

Gentlemen:

The Commission has issued the enclosed Amendments No. 7 to Facility Licenses No. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. The amendments include Change No. 22 to your Technical Specifications for each license and are in response to your request dated April 15, 1975, as supplemented May 1 (Proprietary Information appended) May 20, June 6, June 9 and June 11, 1975.

The amendments revise provisions of the Technical Specifications related to the emergency core cooling system (ECCS). These revisions are based on the licensee's reevaluation of the ECCS performance and are consistent with the requirements of 10 CFR 50 Part 50.46.

The Commission's staff has evaluated the potential for environmental impact associated with operation of the facility in the proposed manner. From this evaluation, the staff has determined that there will be no change in effluent types or total amounts, no increase in authorized power level, and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Part 51, §51.5(c)(1) that no environmental impact statement need be prepared for this action. Copies of the related Negative Declaration and supporting Environmental Impact Appraisal are enclosed. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by:
Robert A. Purple

Robert A. Purple, Chief

(1)

Enclosures:
1. Amendment No. 7 to DPR-32
2. Amendment No. 7 to DPR-37
3. Negative Declaration
4. Environmental Impact Appraisal
5. Safety Evaluation
4. Federal Register Notice

cc w/enclosures:
See next page

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SVarga
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Virginia Electric & Power Company

JUN 1 8 1975

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

Docket Nos. 50-280
and 50-281

June 16, 1975

Virginia Electric & Power Company
ATTN: Mr. Stanley Ragone
Senior Vice President
Post Office Box 26666
Richmond, Virginia 23261

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The Commission has issued the enclosed Amendments No. 7 to Facility Licenses No. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. The amendments include Change No. 22 to your Technical Specifications for each license and are in response to your request dated April 15, 1975, as supplemented May 1 (Proprietary Information appended) May 20, June 6, June 9 and June 11, 1975.

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Sincerely,

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing

Enclosures:
See next page

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-32

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

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated April 15, 1975, as supplemented May 1 (Proprietary Information appended), May 20, June 6, June 9 and June 11, 1975, 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; and

   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.

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

"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 22."

3. This license amendment is effective ten days after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 22 to the
Technical Specifications

Date of Issuance: June 16, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

Revise Appendix A as follows:

Remove Pages	Insert Revised Pages
1.0-7	1.0-7
2.1-2	2.1-2
2.1-6	2.1-6
--	3.2-3a
3.3-1 through 3.3-8	3.3-1 through 3.3-9
3.12-1 through 3.12-18	3.12-1 through 3.12-22
--	Figure 3.12-8
4.10-1 through 4.10-3	4.10-1 through 4.10-3
5.3-2	5.3-2

L. Low Power Physics Tests

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

(Deleted) 22

4. The reactor thermal power level shall not exceed 118% of rated power.

B. 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 TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be limited to 26,000 effective full power hours (EFPH) for Cycles 1 and 2 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2.

22

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 operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially

to this limiting criterion. 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 calculation of this limit.

The fuel residence time is limited to 26,000 EFPH for Cycles 1 and 2 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff.

22

References

(1) FSAR Section 3.4

(2) FSAR Section 3.3

(3) FSAR Section 14.2

4. The manually operated valves CS-25, in Units 1 and 2, in the single flow path from the refueling water storage tank to the charging pumps may be closed on a monthly schedule to perform surveillance tests to verify that the valve position can be changed from fully opened to fully closed and returned to the fully open condition. the valves shall not remain in the closed position for more than 30 minutes. 22

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water tank contains not less than 350,000 gal. of borated water with a boron concentration of at least 2000 ppm.

2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 989 ft^3 of borated water with a boron concentration of at least 1950 ppm.

3. The boron injection tank and isolated portion of the inlet and outlet piping contains no less than 900 gallons of water with a boron

concentration equivalent to at least 11.5% to 13% weight boric acid solution at a temperature of at least 145°F.

4. Two channels of heat tracing shall be available for the flow paths.

5. Two charging pumps are operable.

6. Two low head safety injection pumps are operable.

7. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions are operable.

8. The Charging Pump Cooling Water Subsystem shall be operating as follows:

   a. Make-up water from the Component Cooling Water Subsystem shall be available.

   b. Two charging pump component cooling water pumps and two charging pump service water pumps shall be operable.

   c. Two charging pump intermediate seal coolers shall be operable.

9. During power operation the A.C. power shall be removed from the following motor operated valves with the valve in the open position:

Unit No. 1	Unit No. 2
MOV 1862	MOV 2862
MOV 1885C	MOV 2885C
MOV 1890C	MOV 2890C

10. During power operation the A.C. power shall be removed from the following motor operated valves with the valve in the closed position:

Unit No. 1	Unit No. 2
MOV 1869A	MOV 2869A
MOV 1869B	MOV 2869B
MOV 1890A	MOV 2890A
MOV 1890B	MOV 2890B
MOV 1860A	MOV 2860A
MOV 1860B	MOV 2860B

11. The accumulator discharge valves listed below in non-isolated loops shall be blocked open by de-energizing the valve motor operator when the reactor coolant system pressure is greater than 1000 psig.

Unit No. 1	Unit No. 2
MOV 1865A	MOV 2865A
MOV 1865B	MOV 2865B
MOV 1865C	MOV 2865C

12. Power operation with less than three loops in service is prohibited. The following loop isolation valves shall have AC power removed and be locked in open position during power operation.

Unit No. 1	Unit No. 2
MOV 1590	MOV 2590
MOV 1591	MOV 2591
MOV 1592	MOV 2592
MOV 1593	MOV 2593
MOV 1594	MOV 2594
MOV 1595	MOV 2595

B. The requirements of Specification 3.3-A may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of Specification 3.3-A within the time period specified, the reactor shall initially be placed in the hot shutdown condition. If the requirements of Specification 3.3-A are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition.

1. One accumulator may be isolated for a period not to exceed 4 hours.

2. Two charging pumps per unit may be out of service, provided immediate attention is directed to making repairs and one pump is restored to operable status within 24 hours. If one pump unit is out of service, the standby pump shall be tested before initiating maintenance and once every 8 hours to assure operability.

3. One low head safety injection pump per unit may be out of service, provided immediate attention is directed to making repairs and the pump is restored to operable status within 24 hours. The other low head safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump and shall be tested once every eight (8) hours thereafter, until both pumps are in an operable status or the reactor is shut down.

4. Any one valve in the Safety Injection System may be inoperable provided repairs are initiated immediately and are completed within

24 hours. Prior to initiating repairs, all automatic valves in the redundant system shall be tested to demonstrate operability.

5. One channel of heat tracing may be inoperable for a period not to exceed 24 hours, provided immediate attention is directed to making repairs.

6. One charging pump component cooling water pump or one charging pump service water pump may be out of service provided the pump is restored to operable status within 24 hours.

7. One charging pump intermediate seal cooler or other passive component may be out of service provided the system may still operate at 100 percent capacity and repairs are completed within 48 hours.

8. Power may be restored to any valve referenced in 3.3.A.9 and 3.3.A.10 for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours.

9. Power may be restored to any valve referenced in 3.3.A.11 for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing or maintenance is completed and power removed within 4 hours.

(DELETED)

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant. With this mode of startup the Safety Injection System is required to be operable as specified. During low power physics tests there is a negligible amount of energy stored in the system; therefore an accident comparable in severity to the Design Basis Accident is not possible, and the full capacity of the Safety Injection System is not required.

The operable status of the various systems and components is to be demonstrated by periodic tests, detailed in TS Section 4.1. A large fraction of these tests are performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. A single component being inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component and, in some cases are to be retested at intervals during the repair period. In some cases, i.e. charging pumps, additional components are installed to allow a component to be inoperable without affecting system redundancy. For those cases which are not so designed, if it develops that (a) the inoperable component is not repaired within the

specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. After 48 hours in the hot shutdown condition, if the malfunction(s) are not corrected the reactor will be placed in the cold shutdown condition, following normal shutdown and cooldown procedures.

The Specification requires prompt action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. Furthermore, the specified repair times do not apply to regularly scheduled maintenance of the Safety Injection System, which is normally to be performed during refueling shutdowns. The limiting times for repair are based on: estimates of the time required to diagnose and correct various postulated malfunctions using safe and proper procedures, the availability of tools, materials and equipment; health physics requirements and the extent to which other systems provide functional redundancy to the system under repair.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat production decreases as follows after initiating hot shutdown.

Time After Shutdown	Decay Heat, % of Rated Power
1 min.	3.7
30 min.	1.6

Time After Shutdown	Decay Heat, % of Rated Power
1 hour	1.3
8 hours	0.75
48 hours	0.48

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is reduced by orders of magnitude below the requirements for handling a postulated loss-of-coolant accident occurring during power operation. Placing and maintaining the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, allows access to some of the Safety Injection System components in order to effect repairs, and minimizes the exposure to thermal cycling.

Failure to complete repairs within 48 hours of going to hot shutdwon condition is considered indicative of unforeseen problems, i.e., possibly the need of major maintenance. In such a case the reactor is to be put into the cold shutdown condition.

The accumulators are able to accept leakage from the Reactor Coolant System without any effect on their availability. Allowable inleakage is based on the volume of water that can be added to the initial amount without exceeding the volume given in Specification 3.3.A.2. The maximum acceptable inleakage is 14 cubic feet per tank.

The accumulators (one for each loop) discharge into the cold legs of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motorized valve to isolate the accumulator during reactor start-up and shutdown to preclude the discharge of the contents of the accumulator when not required. These valves receive a signal to open when safety injection is initiated.

To assure that the accumulator valves satisfy the single failure criterion, they will be blocked open by de-energizing the valve motor operators when the reactor coolant pressure exceeds 1000 psig. The operating pressure of the Reactor Coolant System is 2235 psig and safety injection is initiated when this pressure drops to 600 psia. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal start-up operation to perform the actions required to de-energize the valve. This procedure will assure that there is an operable flow path from each accumulator to the Reactor Coolant System during power operation and that safety injection can be accomplished. 22

The removal of power from the valves listed in the specification will assure that the systems of which they are a part satisfy the single failure criterion. 22

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 exercises, the shutdown control rods shall be fully withdrawn.

2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A, 3.12-1B, 3.12-2, or 3.12-3 for three-loop operation and TS Figures 3.12-4A, 3.12-4B, 3.12-5, or 3.12-6 for two-loop operation.

3. The limits shown on TS Figures 3.12-1A through 3.12-6 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 controlling 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 exceed the applicable value shown on TS Figure 3.12-7 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 conditions ($T_{avg} \geq 547^{o}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod 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. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times. 22

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted and part length rods fully withdrawn.

B. <u>Power Distribution Limits</u>

1. At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits: 22

$F_Q(Z) \leq (2.10/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.20) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1 - P))$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q.

2. Prior to exceeding 75% 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 three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. 22

   b. The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

If either measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq 2.10 \times K(Z)$ and $F_{\Delta H}^N \leq 1.55$ within 24 hours, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

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 rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P 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 measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life.

4. Except during physics tests, during excore detector calibration and except as modified by 3.12.B.4.a., b., or c. below, the indicated axial flux difference shall be maintained within a +6 to -9% 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, the flux difference shall be returned to the target band immediately, or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

b. At a power level no greater than 90 percent of rated power,

(1) The indicated axial flux difference may deviate from its +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -18 percent and +11.5 percent at 90% power. For every 4 percent below 90% power, the permissible positive flux difference boundary is extended by 1 percent. For every 5 percent below 90% power, the permissible negative flux difference boundary is extended by 2 percent.

(2) If 3.12.B.4.b.(1) is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint shall be reduced to no greater than 55% power. 22

(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.

c. At a power level no greater than 50 percent of rated power,

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

) 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 two hours (cumulative) out of the preceding 24 hour period. One half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributed to the one hour cumulative maximum the flux difference maximum deviate from its target band at a power level less than or equal 90 percent of rated power.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in 3.12.B.4.a and the flux difference time limits in 3.12.B.4.b. 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.

5. The allowable quadrant to average power tilt is

$$T = 2.0 + 50\ (1.435/F_{xy} - 1) \leq 10\%$$

where F_{xy} is 1.435, or the value of the unrodded horizontal plane peaking factor appropriate to F_Q as determined by a movable incore detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2% if F_{xy} is 1.435 or a value up to 10% if the option to measured F_{xy} is in effect.

6. If the quadrant to average power tilt exceeds a value T% as selected in 3.12.B.5, except for physics and rod exercise testing, then: 22

   a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.12.B.1, or 22

   b. If the hot channel factors are not determined within two hours, the power and high neutron flux trip setpoint shall be reduced from rated power, 2% for each percent of quadrant tilt.

   c. If the quadrant to average power tilt exceeds ±10% except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

7. If after a further period of 24 hours, the power tilt in 3.12.B.6 above is not corrected to less than +T%: 22

   a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Nuclear Regulatory Commission. 22

b. If the design hot channel factors for rated power are exceeded and the power is greater than 10%, the Nuclear Regulatory Commission shall be notified and the nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values. 22

c. If the hot channel factors are not determined the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt. 22

C. Inoperable Control Rods

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism, or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod

drive mechanisms, i.e. programming circuitry, the provisions of 3.12.C.1 and 3.12.C.2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned to service within this specified period the reactor will be brought to hot shutdown conditions. 22

4. The provisions of 3.12.C.1 and 3.12.C.2 shall not apply during physics test in which the assemblies are intentionally misaligned. 22

5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.

6. If an inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in TS Figure 3.12-3 apply.

7. No insertion limit changes are required by an inoperable part-length rod.

8. If a full-length rod becomes inoperable and reactor operation is continued the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the unit power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis. 22

D. If the reactor is operating above 75% of rated power with one excore nuclear channel out of service, the core quadrant power balance shall be determined.

1. Once per day, and

2. After a change in power level greater than 10% or more than 30 inches of control rod motion.

The core quadrant power balance shall be determined by one of the following methods:

1. Movable detectors (at least two per quadrant)

2. Core exit thermocouples (at least four per quadrant).

E. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service then:

   a. For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion, of the non-indicating rod, exceeding 24 steps, whichever occurs first.

   b. During operation below 50% of rated power no special monitoring is required.

2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

F. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or full length control rod is more than 15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Section 3.12.B.1 within 8 hours, power shall be reduced so as not to exceed 75% of permitted power.

2. To increase power above 75% of rated power with a part-length or full length control rod more than 15 inches out of alignment with its bank an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 3.12.B.

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 groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition.

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 rod 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 require-

ment occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

22

The various control rod assemblies (shutdown banks, control banks A, B, C, and D and part-length rods) are each to be moved as a bank, that is, with all assemblies 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 Differential Transformer is approximately ±5% of span (±7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the

most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. 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 (part-length of 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 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.

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 21.1 kw/ft for Unit No. 1 and 20.4 kw/ft for Unit No. 2. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients. 22

In addition to the above, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS 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 on 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 tolerances 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. 22

$F_{\Delta H}^N$, 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.

CHANGE NO. 29

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

An upper bound envelope of 2.10 times the normalized peaking factor axial dependence of TS Figure 3.12-8 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control given in Section 3.12.B.4. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46. 22

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerances.

In the specified limit of $F^N_{\Delta H}$ there is an eight percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases do not necessarily affect 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 can be compensated for the F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system.

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. 22

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

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

2. Control rod banks are sequenced with overlapping banks as shown in Figures 3.12-1A, 3.12-1B and 3.12-2.

3. The full length and part 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 in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core. 22

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, these hot channel factors limits are met. In specification 3.12.B.1 F_Q is arbitrarily limited for $P \leq .5$ (except for physics tests).

The procedures for axial power distribution control referred to above 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 3.12.B.4 assure that the F_Q upper bound envelope of 2.10 times Figure 3.12-8 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

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 +6 to -9% 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. 22

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 possible during certain physics tests or during required, periodic, excore detector calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

22

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 +14.5 to -21 percent (+11.5 percent to -18 percent indicated) where for every 4 percent below rated power, the permissible positive flux difference boundary is extended

by 1 percent, and for every 5 percent below rated power, the permissible negative flux difference boundary is extended by 2 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 rods to produce the required indicated flux difference. 22

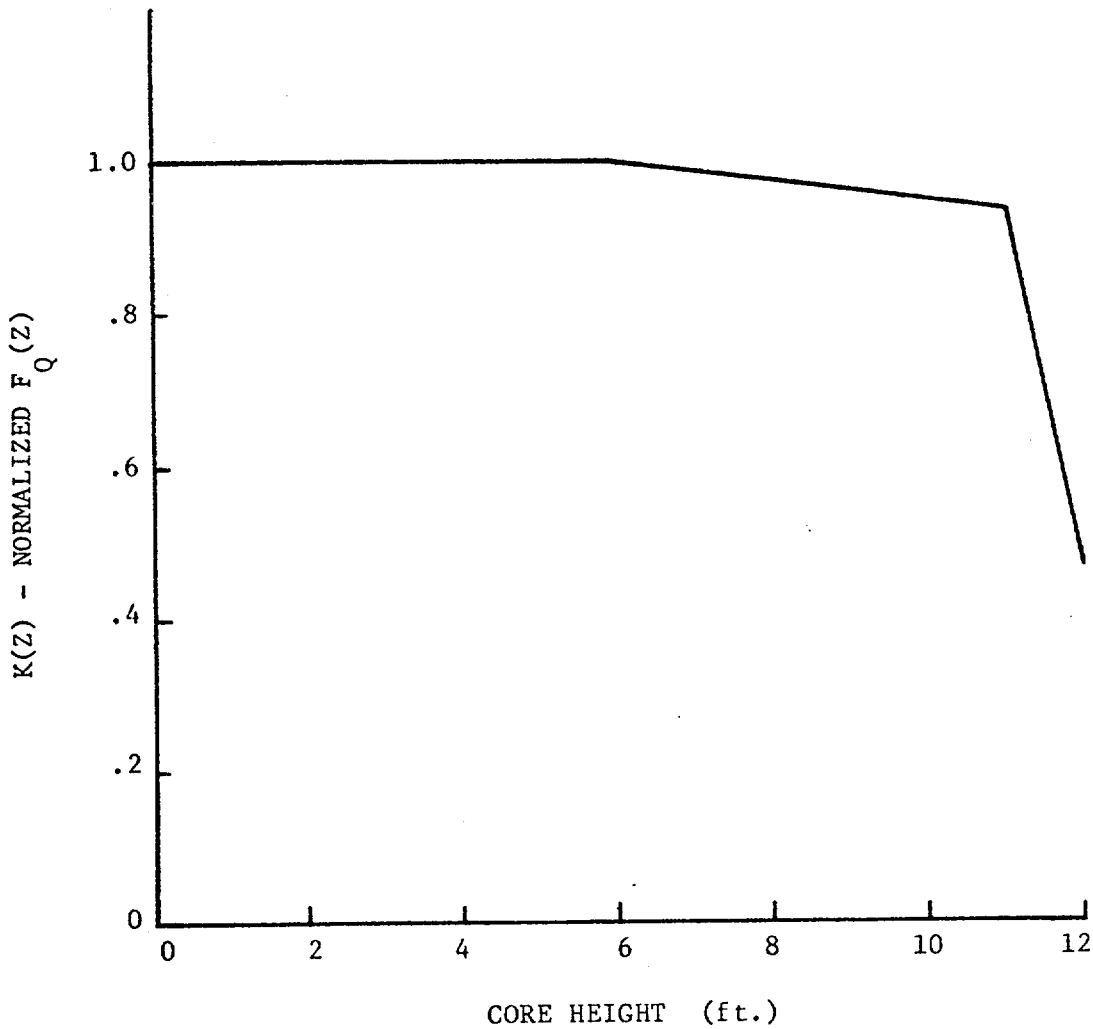
At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of an expansion of permissible quadrant tilt limits over tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 L).

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.

TS FIGURE 3.12-8

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE

SURRY POWER STATION
UNIT NOS. 1 AND 2



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4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, the hot channel factors, F_Q and $F^N_{\Delta H}$, shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed values of 22

$$F_Q(Z) \leq (2.10/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.20) \times K(Z) \text{ for } P \leq .5$$

$$F^N_{\Delta H} \leq 1.55\ (1 + 0.2\ (1 - P))$$

(where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q), an evaluation as to the cause of the anomaly shall be made. 22

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

PEAKING FACTORS

A thermal criterion in the reactor core design specifies that "no fuel melting during any anticipated normal operating condition" should occur. To meet the above criterion during a thermal overpower of 118% with additional margin for design uncertainties, a steady state maximum linear power is selected. This then is an upper linear power limit determined by the maximum central temperature of the hot pellet.

The peaking factor is a ratio taken between the maximum allowed linear power density in the reactor to the average value over the whole reactor. It is of course the average value that determines the operating power level. The peaking factor is a constraint which must be met to assure that the peak linear power density does not exceed the maximum allowed value.

During normal reactor operation, measured peaking factors should be significantly lower than design limits. As core burnup progresses, measured designed peaking factors are expected to decrease. A determination of F_Q and $F^N_{\Delta H}$ during each effective full power month of operation is adequate to ensure that core reactivity changes with burnup have not significantly altered peaking factors in an adverse direction. 22

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight per cent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex glass clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144 inch-length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36 inch-length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. The initial core and subsequent cores will meet the following performance criteria at all times during the operating lifetime.

   a. Hot channel factors:

$$F_Q(Z) \leq (2.10/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.20) \times K(Z) \text{ for } P \leq .5$$

$$F^N_{\Delta H} \leq 1.55\ (1 + 0.2\ (1 - P))$$

   where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q.

22

JUN 1 6 1975

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

VIRGINIA ELECTRIC & POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-37

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

   A. The application for amendment by Virginia Electric & Power Company (the licensee) dated April 15, 1975, as supplemented May 1 (Proprietary Information appended), May 20, June 6, June 9 and June 11, 1975, 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; and

   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.

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

"3.B Technical Specifications

The Technical Specifications contained in Appendix A, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 22."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Giambusso, Director
Division of Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Change No. 22 to the
Technical Specifications

Date of Issuance: June 16, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 7

CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Revise Appendix A as follows:

Remove Pages	Insert Revised Pages
1.0-7	1.0-7
2.1-2	2.1-2
2.1-6	2.1-6
--	3.2-3a
3.3-1 through 3.3-8	3.3-1 through 3.3-9
3.12-1 through 3.12-18	3.12-1 through 3.12-22
--	Figure 3.12-8
4.10-1 through 4.10-3	4.10-1 through 4.10-3
5.3-2	5.3-2

L. Low Power Physics Tests

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

(Deleted) 22

4. The reactor thermal power level shall not exceed 118% of rated power.

B. 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 TS Figures 2.1-1, 2.1-2 or 2.1-3; or the core thermal power exceeds 118% of rated power.

C. The fuel residence time shall be limited to 26,000 effective full power hours (EFPH) for Cycles 1 and 2 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2.

22

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 operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is very large and the clad surface temperature is only a few degrees Fahrenheit above the reactor coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed Departure From Nucleate Boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters; thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially

to this limiting criterion. 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 calculation of this limit.

The fuel residence time is limited to 26,000 EFPH for Cycles 1 and 2 of Unit 1 and to 17,000 EFPH for Cycles 1 and 2 of Unit 2 to assure no fuel clad flattening will occur in the cores without prior review by the Regulatory Staff.

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References

(1) FSAR Section 3.4

(2) FSAR Section 3.3

(3) FSAR Section 14.2

4. The manually operated valves CS-25, in Units 1 and 2, in the single flow path from the refueling water storage tank to the charging pumps may be closed on a monthly schedule to perform surveillance tests to verify that the valve position can be changed from fully opened to fully closed and returned to the fully open condition. the valves shall not remain in the closed position for more than 30 minutes. 22

3.3 SAFETY INJECTION SYSTEM

Applicability

Applies to the operating status of the Safety Injection System.

Objective

To define those limiting conditions for operation that are necessary to provide sufficient borated cooling water to remove decay heat from the core in emergency situations.

Specifications

A. A reactor shall not be made critical unless the following conditions are met:

1. The refueling water tank contains not less than 350,000 gal. of borated water with a boron concentration of at least 2000 ppm.

2. Each accumulator system is pressurized to at least 600 psia and contains a minimum of 975 ft^3 and a maximum of 989 ft^3 of borated water with a boron concentration of at least 1950 ppm. 22

3. The boron injection tank and isolated portion of the inlet and outlet piping contains no less than 900 gallons of water with a boron

concentration equivalent to at least 11.5% to 13% weight boric acid solution at a temperature of at least 145°F.

4. Two channels of heat tracing shall be available for the flow paths.

5. Two charging pumps are operable.

6. Two low head safety injection pumps are operable.

7. All valves, piping, and interlocks associated with the above components which are required to operate under accident conditions are operable.

8. The Charging Pump Cooling Water Subsystem shall be operating as follows:

   a. Make-up water from the Component Cooling Water Subsystem shall be available.

   b. Two charging pump component cooling water pumps and two charging pump service water pumps shall be operable.

   c. Two charging pump intermediate seal coolers shall be operable.

9. During power operation the A.C. power shall be removed from the following motor operated valves with the valve in the open position:

Unit No. 1	Unit No. 2
MOV 1862	MOV 2862
MOV 1885C	MOV 2885C
MOV 1890C	MOV 2890C

10. During power operation the A.C. power shall be removed from the following motor operated valves with the valve in the closed position:

Unit No. 1	Unit No. 2
MOV 1869A	MOV 2869A
MOV 1869B	MOV 2869B
MOV 1890A	MOV 2890A
MOV 1890B	MOV 2890B
MOV 1860A	MOV 2860A
MOV 1860B	MOV 2860B

11. The accumulator discharge valves listed below in non-isolated loops shall be blocked open by de-energizing the valve motor operator when the reactor coolant system pressure is greater than 1000 psig.

Unit No. 1	Unit No. 2
MOV 1865A	MOV 2865A
MOV 1865B	MOV 2865B
MOV 1865C	MOV 2865C

12. Power operation with less than three loops in service is prohibited. The following loop isolation valves shall have AC power removed and be locked in open position during power operation.

Unit No. 1	Unit No. 2
MOV 1590	MOV 2590
MOV 1591	MOV 2591
MOV 1592	MOV 2592
MOV 1593	MOV 2593
MOV 1594	MOV 2594
MOV 1595	MOV 2595

B. The requirements of Specification 3.3-A may be modified to allow one of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of Specification 3.3-A within the time period specified, the reactor shall initially be placed in the hot shutdown condition. If the requirements of Specification 3.3-A are not satisfied within an additional 48 hours the reactor shall be placed in the cold shutdown condition.

1. One accumulator may be isolated for a period not to exceed 4 hours.

2. Two charging pumps per unit may be out of service, provided immediate attention is directed to making repairs and one pump is restored to operable status within 24 hours. If one pump unit is out of service, the standby pump shall be tested before initiating maintenance and once every 8 hours to assure operability.

3. One low head safety injection pump per unit may be out of service, provided immediate attention is directed to making repairs and the pump is restored to operable status within 24 hours. The other low head safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump and shall be tested once every eight (8) hours thereafter, until both pumps are in an operable status or the reactor is shut down.

4. Any one valve in the Safety Injection System may be inoperable provided repairs are initiated immediately and are completed within

24 hours. Prior to initiating repairs, all automatic valves in the redundant system shall be tested to demonstrate operability.

5. One channel of heat tracing may be inoperable for a period not to exceed 24 hours, provided immediate attention is directed to making repairs.

6. One charging pump component cooling water pump or one charging pump service water pump may be out of service provided the pump is restored to operable status within 24 hours.

7. One charging pump intermediate seal cooler or other passive component may be out of service provided the system may still operate at 100 percent capacity and repairs are completed within 48 hours.

8. Power may be restored to any valve referenced in 3.3.A.9 and 3.3.A.10 for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours.

9. Power may be restored to any valve referenced in 3.3.A.11 for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing or maintenance is completed and power removed within 4 hours.

(DELETED)

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant. With this mode of startup the Safety Injection System is required to be operable as specified. During low power physics tests there is a negligible amount of energy stored in the system; therefore an accident comparable in severity to the Design Basis Accident is not possible, and the full capacity of the Safety Injection System is not required.

The operable status of the various systems and components is to be demonstrated by periodic tests, detailed in TS Section 4.1. A large fraction of these tests are performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. A single component being inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) are to be tested prior to initiating repair of the inoperable component and, in some cases are to be retested at intervals during the repair period. In some cases, i.e. charging pumps, additional components are installed to allow a component to be inoperable without affecting system redundancy. For those cases which are not so designed, if it develops that (a) the inoperable component is not repaired within the

specified allowable time period, or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. After 48 hours in the hot shutdown condition, if the malfunction(s) are not corrected the reactor will be placed in the cold shutdown condition, following normal shutdown and cooldown procedures.

The Specification requires prompt action to effect repairs of an inoperable component, and therefore in most cases repairs will be completed in less than the specified allowable repair times. Furthermore, the specified repair times do not apply to regularly scheduled maintenance of the Safety Injection System, which is normally to be performed during refueling shutdowns. The limiting times for repair are based on: estimates of the time required to diagnose and correct various postulated malfunctions using safe and proper procedures, the availability of tools, materials and equipment; health physics requirements and the extent to which other systems provide functional redundancy to the system under repair.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat production decreases as follows after initiating hot shutdown.

Time After Shutdown	Decay Heat, % of Rated Power
1 min.	3.7
30 min.	1.6

Time After Shutdown	Decay Heat, % of Rated Power
1 hour	1.3
8 hours	0.75
48 hours	0.48

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is reduced by orders of magnitude below the requirements for handling a postulated loss-of-coolant accident occurring during power operation. Placing and maintaining the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, allows access to some of the Safety Injection System components in order to effect repairs, and minimizes the exposure to thermal cycling.

Failure to complete repairs within 48 hours of going to hot shutdwon condition is considered indicative of unforeseen problems, i.e., possibly the need of major maintenance. In such a case the reactor is to be put into the cold shutdown condition.

The accumulators are able to accept leakage from the Reactor Coolant System without any effect on their availability. Allowable inleakage is based on the volume of water that can be added to the initial amount without exceeding the volume given in Specification 3.3.A.2. The maximum acceptable inleakage is 14 cubic feet per tank.

The accumulators (one for each loop) discharge into the cold legs of the reactor coolant piping when Reactor Coolant System pressure decreases below accumulator pressure, thus assuring rapid core cooling for large breaks. The line from each accumulator is provided with a motorized valve to isolate the accumulator during reactor start-up and shutdown to preclude the discharge of the contents of the accumulator when not required. These valves receive a signal to open when safety injection is initiated.

To assure that the accumulator valves satisfy the single failure criterion, they will be blocked open by de-energizing the valve motor operators when the reactor coolant pressure exceeds 1000 psig. The operating pressure of the Reactor Coolant System is 2235 psig and safety injection is initiated when this pressure drops to 600 psia. De-energizing the motor operator when the pressure exceeds 1000 psig allows sufficient time during normal start-up operation to perform the actions required to de-energize the valve. This procedure will assure that there is an operable flow path from each accumulator to the Reactor Coolant System during power operation and that safety injection can be accomplished. 22

The removal of power from the valves listed in the specification will assure that the systems of which they are a part satisfy the single failure criterion. 22

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 exercises, the shutdown control rods shall be fully withdrawn.

2. Whenever the reactor is critical, except for physics tests and control rod assembly exercises, the full length control rod banks shall be inserted no further than the appropriate limit determined by core burnup shown on TS Figures 3.12-1A, 3.12-1B, 3.12-2, or 3.12-3 for three-loop operation and TS Figures 3.12-4A, 3.12-4B, 3.12-5, or 3.12-6 for two-loop operation.

3. The limits shown on TS Figures 3.12-1A through 3.12-6 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 controlling 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 exceed the applicable value shown on TS Figure 3.12-7 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 conditions ($T_{avg} \geq 547^{o}F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, boron, or part-length rod position.

4. Whenever the reactor is subcritical, except for physics tests, the critical rod position, i.e., the rod 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. Operation with part length rods shall be restricted such that except during physics tests, the part length rod banks are withdrawn from the core at all times. 22

6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated in TS Figure 3.12-7 must be maintained except for the low power physics test to measure control rod worth and shutdown margin. For this test the reactor may be critical with all but one full length control rod, expected to have the highest worth, inserted and part length rods fully withdrawn.

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: 22

$F_Q(Z) \leq (2.10/P) \times K(Z)$ for $P > .5$

$F_Q(Z) \leq (4.20) \times K(Z)$ for $P \leq .5$

$F^N_{\Delta H} \leq 1.55\ (1 + 0.2(1 - P))$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q.

2. Prior to exceeding 75% 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 three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error. 22

   b. The measurement of enthalpy rise hot channel factor, $F_{\Delta H}^N$, shall be increased by four percent to account for measurement error.

   If either measured hot channel factor exceeds its limit specified under 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits $F_Q \leq 2.10 \times K(Z)$ and $F_{\Delta H}^N \leq 1.55$ within 24 hours, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

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 rods more than 190 steps withdrawn. The target flux difference at any other power level, P, is equal to the target value of P 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 measurement, or by linear interpolation using the most recent value and the value predicted for the end of the cycle life. 22

4. Except during physics tests, during excore detector calibration and except as modified by 3.12.B.4.a., b., or c. below, the indicated axial flux difference shall be maintained within a +6 to -9% 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, the flux difference shall be returned to the target band immediately, or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

b. At a power level no greater than 90 percent of rated power,

(1) The indicated axial flux difference may deviate from its +6 to -9% target band for a maximum of one hour (cumulative) in any 24 hour period provided the flux difference does not exceed an envelope bounded by -18 percent and +11.5 percent at 90% power. For every 4 percent below 90% power, the permissible positive flux difference boundary is extended by 1 percent. For every 5 percent below 90% power, the permissible negative flux difference boundary is extended by 2 percent.

(2) If 3.12.B.4.b.(1) is violated then the reactor power shall be reduced to no greater than 50% power and the high neutron flux setpoint shall be reduced to no greater than 55% power. 22

(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.

c. At a power level no greater than 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 two hours (cumulative) out of the preceding 24 hour period. One half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributed to the one hour cumulative maximum the flux difference maximum deviate from its target band at a power level less than or equal 90 percent of rated power.

Alarms shall normally be used to indicate the deviations from the axial flux difference requirements in 3.12.B.4.a and the flux difference time limits in 3.12.B.4.b. 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. 22

5. The allowable quadrant to average power tilt is

$$T = 2.0 + 50\ (1.435/F_{xy} - 1) \leq 10\%$$

where F_{xy} is 1.435, or the value of the unrodded horizontal plane peaking factor appropriate to F_Q as determined by a movable incore detector map taken on at least a monthly basis; and T is the percentage operating quadrant tilt limit, having a value of 2% if F_{xy} is 1.435 or a value up to 10% if the option to measured F_{xy} is in effect.

6. If the quadrant to average power tilt exceeds a value T% as selected in 3.12.B.5, except for physics and rod exercise testing, then: 22

   a. The hot channel factors shall be determined within 2 hours and the power level adjusted to meet the specification of 3.12.B.1, or 22

   b. If the hot channel factors are not determined within two hours, the power and high neutron flux trip setpoint shall be reduced from rated power, 2% for each percent of quadrant tilt.

   c. If the quadrant to average power tilt exceeds ±10% except for physics tests, the power level and high neutron flux trip setpoint will be reduced from rated power, 2% for each percent of quadrant tilt.

7. If after a further period of 24 hours, the power tilt in 3.12.B.6 above is not corrected to less than +T%: 22

   a. If design hot channel factors for rated power are not exceeded, an evaluation as to the cause of the discrepancy shall be made and reported as an abnormal occurrence to the Nuclear Regulatory Commission. 22

b. If the design hot channel factors for rated power are exceeded and the power is greater than 10%, the Nuclear Regulatory Commission shall be notified and the nuclear overpower, overpower ΔT and overtemperature ΔT trips shall be reduced one percent for each percent the hot channel factor exceeds the rated power design values. 22

c. If the hot channel factors are not determined the Nuclear Regulatory Commission shall be notified and the overpower ΔT and overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt. 22

C. Inoperable Control Rods

1. A control rod assembly shall be considered inoperable if the assembly cannot be moved by the drive mechanism, or the assembly remains misaligned from its bank by more than 15 inches. A full-length control rod shall be considered inoperable if its rod drop time is greater than 1.8 seconds to dashpot entry.

2. No more than one inoperable control rod assembly shall be permitted when the reactor is critical.

3. If more than one control rod assembly in a given bank is out of service because of a single failure external to the individual rod

drive mechanisms, i.e. programming circuitry, the provisions of 3.12.C.1 and 3.12.C.2 shall not apply and the reactor may remain critical for a period not to exceed two hours provided immediate attention is directed toward making the necessary repairs. In the event the affected assemblies cannot be returned to service within this specified period the reactor will be brought to hot shutdown conditions. 22

4. The provisions of 3.12.C.1 and 3.12.C.2 shall not apply during physics test in which the assemblies are intentionally misaligned. 22

5. If an inoperable full-length rod is located below the 200 step level and is capable of being tripped, or if the full-length rod is located below the 30 step level whether or not it is capable of being tripped, then the insertion limits in TS Figure 3.12-2 apply.

6. If an inoperable full-length rod cannot be located, or if the inoperable full-length rod is located above the 30 step level and cannot be tripped, then the insertion limits in TS Figure 3.12-3 apply.

7. No insertion limit changes are required by an inoperable part-length rod.

8. If a full-length rod becomes inoperable and reactor operation is continued the potential ejected rod worth and associated transient power distribution peaking factors shall be determined by analysis within 30 days. The analysis shall include due allowance for non-uniform fuel depletion in the neighborhood of the inoperable rod. If the analysis results in a more limiting hypothetical transient than the cases reported in the safety analysis, the unit power level shall be reduced to an analytically determined part power level which is consistent with the safety analysis. 22

D. If the reactor is operating above 75% of rated power with one excore nuclear channel out of service, the core quadrant power balance shall be determined.

1. Once per day, and

2. After a change in power level greater than 10% or more than 30 inches of control rod motion.

The core quadrant power balance shall be determined by one of the following methods:

1. Movable detectors (at least two per quadrant)

2. Core exit thermocouples (at least four per quadrant).

E. Inoperable Rod Position Indicator Channels

1. If a rod position indicator channel is out of service then:

   a. For operation between 50% and 100% of rated power, the position of the RCC shall be checked indirectly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to motion, of the non-indicating rod, exceeding 24 steps, whichever occurs first.

   b. During operation below 50% of rated power no special monitoring is required.

2. Not more than one rod position indicator (RPI) channel per group nor two RPI channels per bank shall be permitted to be inoperable at any time.

F. Misaligned or Dropped Control Rod

1. If the Rod Position Indicator Channel is functional and the associated part length or full length control rod is more than 15 inches out of alignment with its bank and cannot be realigned, then unless the hot channel factors are shown to be within design limits as specified in Section 3.12.B.1 within 8 hours, power shall be reduced so as not to exceed 75% of permitted power.

2. To increase power above 75% of rated power with a part-length or full length control rod more than 15 inches out of alignment with its bank an analysis shall first be made to determine the hot channel factors and the resulting allowable power level based on Section 3.12.B.

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 groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition.

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 rod 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 require-

ment occurs at end of core life and is based on the value used in the analysis of the hypothetical steam break accident. The rod insertion limits are based on end of core life conditions. Early in core life, less shutdown margin is required, and TS Figure 3.12-7 shows the shutdown margin equivalent to 1.77% reactivity at end-of-life with respect to an uncontrolled cooldown. All other accident analyses are based on 1% reactivity shutdown margin.

Relative positions of control rod banks are determined by a specified control rod bank overlap. This overlap is based on the consideration of axial power shape control.

The specified control rod insertion limits have been revised to limit the potential ejected rod worth in order to account for the effects of fuel densification.

22

The various control rod assemblies (shutdown banks, control banks A, B, C, and D and part-length rods) are each to be moved as a bank, that is, with all assemblies 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 Differential Transformer is approximately ±5% of span (±7.5 inches) under steady state conditions. The relative accuracy of the linear position indicator is such that, with the

most adverse errors, an alarm is actuated if any two assemblies within a bank deviate by more than 14 inches. 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 (part-length of 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 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.

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 21.1 kw/ft for Unit No. 1 and 20.4 kw/ft for Unit No. 2. Second, the minimum DNBR in the core must not be less than 1.30 in normal operation or in short term transients. 22

In addition to the above, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS 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 on 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 tolerances 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. 22

$F_{\Delta H}^N$, 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.

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

An upper bound envelope of 2.10 times the normalized peaking factor axial dependence of TS Figure 3.12-8 has been determined from extensive analyses considering all operating maneuvers consistent with the technical specifications on power distribution control given in Section 3.12.B.4. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46. 22

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerances.

In the specified limit of $F^N_{\Delta H}$ there is an eight percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.55/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g. rod misalignment) affect $F^N_{\Delta H}$, in most cases do not necessarily affect 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 can be compensated for the F_Q by tighter axial control, but compensation for $F^N_{\Delta H}$ is taken, experimental error must be allowed for and four percent is the appropriate allowance for a full core map ($\geq$ 40 thimbles monitored) taken with the movable incore detector flux mapping system.

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. 22

For normal operation, it is not necessary to measure these quantities. Instead it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

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

2. Control rod banks are sequenced with overlapping banks as shown in Figures 3.12-1A, 3.12-1B and 3.12-2.

3. The full length and part 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 in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core. 22

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, these hot channel factors limits are met. In specification 3.12.B.1 F_Q is arbitrarily limited for $P \leq .5$ (except for physics tests).

The procedures for axial power distribution control referred to above 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 3.12.B.4 assure that the F_Q upper bound envelope of 2.10 times Figure 3.12-8 is not exceeded and xenon distributions are not developed which at a later time, would cause greater local power peaking even though the flux difference is then within the limits specified by the procedure.

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 +6 to -9% 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. 22

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 possible during certain physics tests or during required, periodic, excore detector calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore detector calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations.

22

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 +14.5 to -21 percent (+11.5 percent to -18 percent indicated) where for every 4 percent below rated power, the permissible positive flux difference boundary is extended

by 1 percent, and for every 5 percent below rated power, the permissible negative flux difference boundary is extended by 2 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 rods to produce the required indicated flux difference. 22

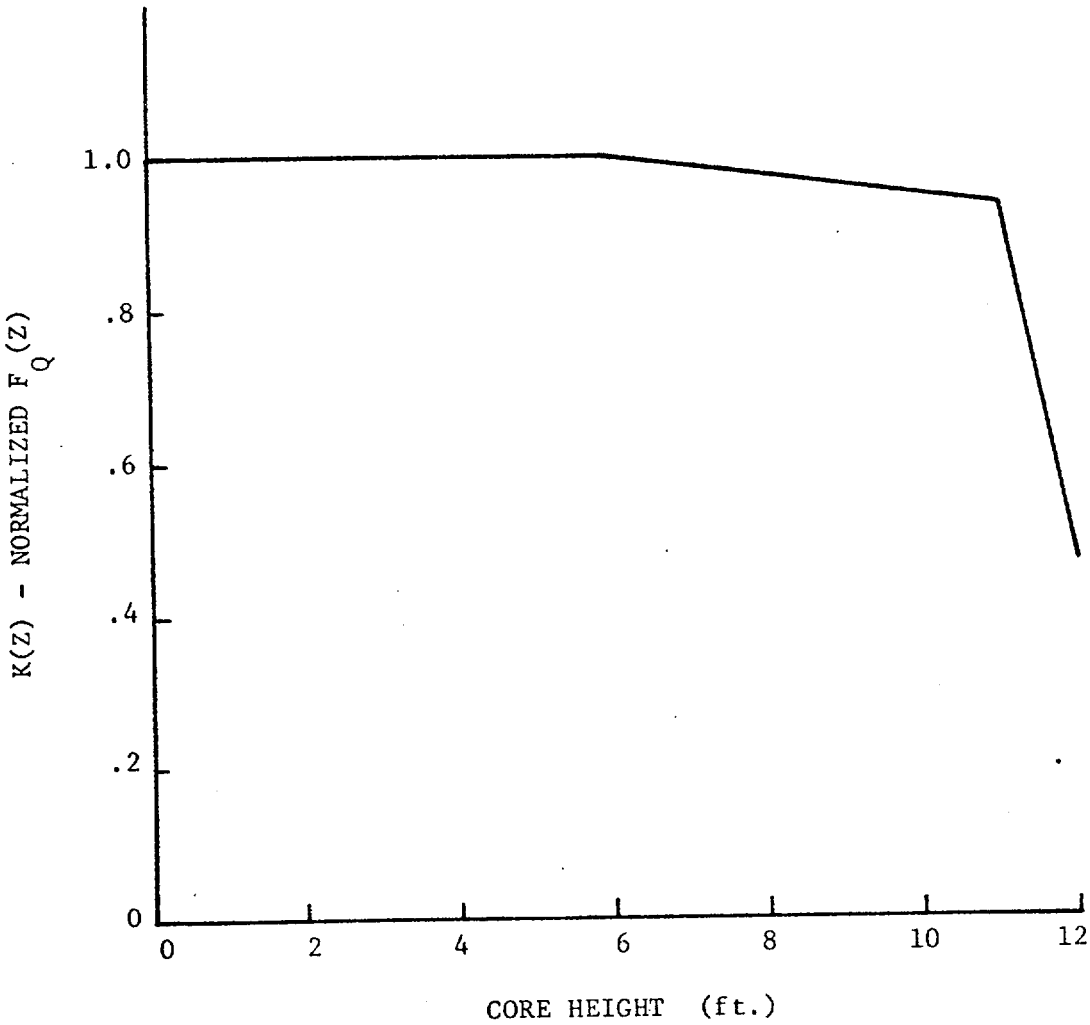
At the option of the operator, credit may be taken for measured decreases in the unrodded horizontal plane peaking factor, F_{xy}. This credit may take the form of an expansion of permissible quadrant tilt limits over tilt limits over the 2% value, up to a value of 10%, at which point specified power reductions are prudent. Monthly surveillance of F_{xy} bounds the quantity because it decreases with burnup. (WCAP-7912 L).

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.

TS FIGURE 3.12-8

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE

SURRY POWER STATION
UNIT NOS. 1 AND 2



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4.10 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of applicable reactivity anomalies within the reactor.

Specification

A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be compared monthly with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission per Section 6.6 of these Specifications.

B. During periods of power operation at greater than 10% of power, the hot channel factors, F_Q and $F^N_{\Delta H}$, shall be determined during each effective full power month of operation using data from limited core maps. If these factors exceed values of 22

$$F_Q(Z) \leq (2.10/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.20) \times K(Z) \text{ for } P \leq .5$$

$$F^N_{\Delta H} \leq 1.55\ (1 + 0.2\ (1 - P))$$

(where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q), an evaluation as to the cause of the anomaly shall be made. 22

Basis

BORON CONCENTRATION

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod assembly groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration, and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a shutdown margin of at least 1% with the most reactive control rod assembly in the fully withdrawn position is always maintained.

PEAKING FACTORS

A thermal criterion in the reactor core design specifies that "no fuel melting during any anticipated normal operating condition" should occur. To meet the above criterion during a thermal overpower of 118% with additional margin for design uncertainties, a steady state maximum linear power is selected. This then is an upper linear power limit determined by the maximum central temperature of the hot pellet.

The peaking factor is a ratio taken between the maximum allowed linear power density in the reactor to the average value over the whole reactor. It is of course the average value that determines the operating power level. The peaking factor is a constraint which must be met to assure that the peak linear power density does not exceed the maximum allowed value.

During normal reactor operation, measured peaking factors should be significantly lower than design limits. As core burnup progresses, measured designed peaking factors are expected to decrease. A determination of F_Q and $F^N_{\Delta H}$ during each effective full power month of operation is adequate to ensure that core reactivity changes with burnup have not significantly altered peaking factors in an adverse direction. 22

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3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight per cent of U-235.

4. Burnable poison rods are incorporated in the initial core. There are 816 poison rods in the form of 12 rod clusters, which are located in vacant control rod assembly guide thimbles. The burnable poison rods consist of pyrex glass clad with stainless steel.

5. There are 48 full-length control rod assemblies and 5 part-length control rod assemblies in the reactor core. The full-length control rod assemblies contain a 144 inch-length of silver-indium-cadmium alloy clad with stainless steel. The part-length control rod assemblies contain a 36 inch-length of silver-indium-cadmium alloy with the remainder of the stainless steel sheath filled with Al_2O_3.

6. The initial core and subsequent cores will meet the following performance criteria at all times during the operating lifetime.

   a. Hot channel factors:

$$F_Q(Z) \leq (2.10/P) \times K(Z) \text{ for } P > .5$$

$$F_Q(Z) \leq (4.20) \times K(Z) \text{ for } P \leq .5$$

$$F^N_{\Delta H} \leq 1.55\ (1 + 0.2\ (1 - P))$$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in Figure 3.12-8, and Z is the core height location of F_Q.

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NEGATIVE DECLARATION

REGARDING PROPOSED CHANGES TO THE

TECHNICAL SPECIFICATIONS OF LICENSE DPR-32 AND DPR-37

SURRY POWER STATION UNITS 1 AND 2

DOCKET NOS.: 50-280 AND 50-281

The Nuclear Regulatory Commission (the Commission) has considered the issuance of changes to the Technical Specifications of Facility Operating License Nos. DPR-32 and DPR-37. These changes would authorize the Virginia Electric and Power Company (VEPCO) (the licensee) to operate the Surry Power Station Units 1 and 2 (located in Surry County, Virginia) with changes to the limiting conditions for operation resulting from application of the Acceptance Criteria for Emergency Core Cooling System (ECCS). This change is being made in conjunction with a partial reactor refueling for core cycle 2 of Unit 2.

The U. S. Nuclear Regulatory Commission, Division of Reactor Licensing, has prepared an environmental impact appraisal for the proposed changes to the Technical Specifications of License Nos. DPR-32 and DPR-37, Surry Units 1 and 2, described above. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no environmental impact attributable to the proposed action other than that which has already been predicted and described in the Commission's Final Environmental Statements for Surry Units 1 and 2 published in May and June 1972,

respectively. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room, 1717 H Street, W. W., Washington, D. C., and at the Swem Library, College of William & Mary, Williamsburg, Virginia, 23185.

Dated at Rockville, Maryland, this 15 day of May 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Fred J. Clark, Jr.

Fred J. Clark, Jr., Acting Chief
Environmental Projects Branch 2
Division of Reactor Licensing

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

ENVIRONMENTAL IMPACT APPRASIAL BY DIVISION OF REACTOR LICENSING

SUPPORTING AMENDMENT NOS.: 7 TO DPR-32 AND DPR-37

CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNITS 1 AND 2

ENVIRONMENTAL IMPACT APPRAISAL

1. Description of Proposed Action

By letters dated March 12 and April 15, 1975, Virginia Electric and Power Company (VEPCO) submitted proposed changes to the Technical Specifications Appendix A to Licenses DPR-32 and DPR-37. The proposed changes resulted from the application of the Acceptance Criteria for Emergency Core Cooling System (ECCS) in conjunction with authorization for a partial reactor first core refueling for the second cycle of operation for Unit 2 only. The Commission staff has reviewed this matter and the conclusions are set forth below.

VEPCO is presently licensed to operate Surry Power Station Units 1 and 2, located in the State of Virginia, Surry County, at power levels up to 2441 megawatts thermal (MWt). The proposed change to incorporate the ECCS Acceptance Criteria does not result in an increase or decrease in power levels of either unit. The restrictions on heat generation rates will require careful control of fuel operating history. However, there should be no reduction on total fuel burnup resulting from the revised ECCS evaluation methods. Since neither power level nor fuel burnup is affected by the action, the action does not affect the benefits of electric power production considered for the captioned facility in the Commission's Final Environmental Statements (FES) for Surry Power Station, Docket Nos. 50-280 and 50-281, dated May and June 1972, respectively.

2. Environmental Impacts of Proposed Action

Potential environmental impacts associated with the proposed action are those which may be associated with incorporation of the ECCS Acceptance Criteria and utilization of nuclear fuel for this facility.

It is particularly noted that in the absence of any significant change in power levels, there will be no change in cooling water requirements and consequently no increase in environmental impact from radioactive effluents and thermal effluents for normal operation or post-accident conditions which in turn could not lead to significant increases in radiation doses or thermal stress to the public or to biota in the environment.

For normal operating conditions, no environmental impact other than as described in the Commission's Final Environmental Statements (FES) for Surry Power Station Units 1 and 2, Docket Nos. 50-280 and 50-281, dated May and June 1972, respectively can be predicted for the proposed action. The Commission's calculated releases for radioactive effluents, both gaseous and liquid, are based on expected release rates to the environment and are quantified on the basis of the total quantity of nuclear fuel within the reactor. The estimates of radionuclides and release rates will not be affected by the proposed action, and since the total quantity of nuclear fuel is unchanged, no increase in the calculated release of radioactive effluents is predicted. Consequently, no increases in radiation doses to man or other biota are predicted.

3. Conclusion and Basis for Negative Declaration

On the basis of the foregoing analysis, it is concluded that there will be no environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for Surry Power Station Units 1 and 2. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

DATE: May 15, 1975

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENTS NO. 7 TO LICENSES NO. DPR-32 AND DPR-37

CHANGE NO. 22 TO TECHNICAL SPECIFICATIONS

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By letter dated April 15, 1975, as supplemented May 1 (with proprietary information appended) May 20, June 6, June 9, and June 11, 1975, Virginia Electric and Power Company requested changes to the Technical Specifications appended to Facility Operating Licenses DPR-32 and DPR-37 for the Surry Power Station Units 1 and 2. The purpose of the request is to revise portions of the Technical Specifications related to the emergency core cooling system (ECCS).

Discussion

These revisions are based on the licensee's reevaluation of the ECCS performance and are consistent with the requirements of 10 CFR Part 50, Section 50.46. In addition, the enclosed revision to Technical Specification 2.1 removes a restriction, placed in OL Amendment No. 6, Change No. 21 dated June 10, 1975, on power operations at the level required for low power physics tests only. Since Change No. 21, authorizing low power physics test, was issued prior to completion of our evaluation of ECCS performance for facility power operation, Change No. 21 included a restriction limiting power operation to the levels required for low power physics testing only. Now that our evaluation of ECCS performance has been completed, this limitation is no longer necessary.

Background on ECCS Evaluation

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License (1) implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms with the provisions of 10 CFR Part 50, Section 50.46. The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendment as may be necessary to implement the evaluation results. As required by our Order of December 27, 1974, Virginia Electric Power Corporation (the licensee) has submitted an ECCS reevaluation and related Technical Specifications. The reevaluation

and Technical Specifications, which are applicable to the Surry reactors for the refueled core (Cycle 2), were submitted in a letter dated June 6, 1975, and made use of the Westinghouse March 15, 1975 model which reflects the reactor vessel reflood rate applicable to the Surry Units 1 and 2. The Licensee submitted an Appendix K reanalysis on April 11, 1975, utilizing the December 25, 1974 version of the Westinghouse evaluation model. Previous to this date, the staff and Westinghouse had concluded their development of a refinement of that model which reflected a refinement in the steam cooling model. The staff had requested the Licensee to reanalyze using the March 15, 1975 version of the Westinghouse evaluation model and that analysis was submitted for staff review June 6, 1975.

ECCS Reanalysis

The background of the staff review of the Westinghouse ECCS models and their application to Surry Nuclear Power Plants, Units 1 and 2 is described in the staff SER for this facility dated December 27, 1974 (the December 27, 1974 SER) issued in connection with the Order. The bases for acceptance of the principal portions of the evaluation model are set forth in the staff's Status Report of October 1974 (2) and the Supplement to the Status Report of November 1974 (3) which are referenced in the December 27, 1974 SER. The December 27, 1974 SER also describes the various changes required in the earlier Westinghouse evaluation model. Together, the December 27, 1974 SER and the Status Report and its Supplement describe an acceptable ECCS evaluation model and the basis for the staff's acceptance of the model. The Surry ECCS evaluation which is covered by this safety evaluation properly conforms to the accepted model. The June 6, 1975 submittal contained: (1) analyses of sufficient break sizes and locations to verify that the worst break condition had been considered and (2) documentation, by reference to submitted Westinghouse Topical Reports, of the ECCS model modifications described in our December 27, 1974 SER.

The analyses submitted June 6, 1974, identified the worst break size as the 0.4 double-ended cold leg guillotine with a calculated peak clad temperature of 2090°F, well below the acceptable limit of 2200°F as sepcified in 10 CFR 50.46(b). In addition, the calculated maximum local metal/water reaction of 5.60% and total core wide metal/water reaction of less than 0.3% were well below the allowable limits of 17% and 1%, respectively.

Our review of plant-specific assumptions regarding the Surry analysis addressed the areas of minimum containment pressure, long term core cooling with respect to potential boron precipitation concerns, and the single failure criterion.

A. ECCS Containment Pressure Evaluation

The ECCS containment pressure calculations for the Surry Plant were done using the Westinghouse ECCS evaluation model. We reviewed Westinghouse's model and published a Status Report on October 15, 1974(2), which was amended November 13, 1974(3). We concluded that Westinghouse's containment pressure model was acceptable for ECCS evaluation and required that justification of the plant-dependent input parameters used in the analysis be submitted for our review of each plant.

This information was submitted for the Surry Plant by letter dated December 31, 1974[4]. Vepco has reevaluated the containment net-free volume, and passive heat sinks, and operation of the containment heat removal systems with regard to the conservatism for ECCS analysis. This evaluation was based on measurements within the containment and from as-built drawings to which additional margin was added. The containment heat removal systems were assumed to operate at their maximum capacities and minimum operational values for the spray water and service water temperatures were assumed.

We have concluded that the plant-dependent information used for the ECCS containment pressure analysis for Surry was conservative and therefore the calculated containment pressures are in accordance with Appendix K to 10 CFR Part 50 of the Commission's regulations.

B. Single Failure Criterion

Appendix K to 10 CFR Part 50 to the Commission's regulations requires that the combination of ECCS subsystems to be assumed operative shall be those available after the most damaging single failure of ECCS equipment has occurred. The worst single failure which would minimize the ECCS available to cool the core and provide maximum containment cooling was identified by Westinghouse as the loss of a low pressure ECCS pump. We concluded in Reference 2 that the application of the single failure criterion was to be confirmed during subsequent plant reviews.

A review of the Surry piping and instrumentation diagrams indicated that the spurious actuation of specific motor operated valves could affect the appropriate single failure assumptions. We identified the following motor operated valves which did not satisfy the single failure criterion.

MOV #	Location
862	LPSI Pump Suction from RWST
885C	LPSI Pump Recirc. line to RSWT
890C	LPSI Pump Discharge to cold legs
869A	Charging Pump Discharge to Hot Legs
869B	Charging Pump Discharge to Hot Legs
865A, B & C	Accumulator Isolation Valves
890A & B	LPSI Pump Discharge to Hot Legs
860A & B	LPSI Pump Suction from Containment Sump

The licensee has reviewed the consequences of these spurious failures and has proposed the following actions:

1-During power operation A.C. power will be removed from MOV 890C. This valve will be in its open position in accordance with Technical Specifications 3.3.A.9 and 3.3.B.8.

2-During power operation A.C. power will be removed from MOV 869A & B and 890A & B. These valves will be in their closed position in accordance with Technical Specification 3.3.A.10 and 3.3.B.8.

3-When the reactor coolant system pressure exceeds 1000 pisg A.C. power will be removed from accumulator isolation valves MOV 865A, B, and C. These valves will be in their open position in accordance with Technical Specifications 3.3.A.11 and 3.3.B.9.

4-MOV 862 and MOV 885C are normally open valves which must be open during the injection phase following a LOCA and must be closed during the recirculation phase. MOV 860A and MOV 860B are normally closed valves which must be closed during the injection phase following a LOCA and must be open during the recirculation phase. To preclude a single failure which would result in a loss of capability to perform an intended safety function, the licensee has proposed to modify the piping and valve arrangment for the Surry plant (Units 1 & 2). The licensee will submit for staff review and approval the details of the proposed modifications within 30 days, along with a proposed schedule for implementation of these modifications. At that time the staff will review the proposed modifications and the schedule for implementation of the required modifications.

Until such time that the required modifications are completed, we have changed the technical specifications in accordance with the licensee's commitment of June 13, 1975, to implement the following interim procedures:

a) MOV 862 and MOV 885C which are normally open and MOV 860A and MOV 860B which are normally closed during plant operation will have A.C. power removed from the motor operators with the valves in their normal position.

b) In the event of a LOCA, operating personnel assigned specifically for the purpose to restore A.C. power to MOV 862, 885C, 860A, and 860B, will be dispatched immediately from the control room to the location of the circuit breakers. Power will be restored to these valves at the instruction of the control room following the actuation of the RWST low level alarm in the sequence specified in the emergency operating procedures. Communication systems must be provided between the control room and the assigned operating personnel. Once actuation of these valves is confirmed the assigned operating personnel will be instructed from the control room to remove A.C. power to the motor operators to prevent spurious motion of these valves during

the recirculation phase of operation.

c) In the event of a LOCA, operating personnel assigned specifically for the purpose to close the manual valve (CS-25) at the RWST will be dispatched immediately from the control room to the RWST. Following actuation of the RSWT low-level alarm at the instruction from the control room, valve CS-25 will be closed in the sequence specified in the emergency operating procedures. Communication systems will be provided between the control room and the assigned operating personnel. These personnel are in addition to those identified in procedure 4-b) above.

d) To provide assurance that the manual valve, CS-25, will be operable, the surveillance requirements for this valve will be changed to require manual surveillance on a monthly basis to verify that the valve position can be changed from fully opened to fully closed.

The staff has evaluated the time available for the operator to accomplish the necessary manual actions for changeover from injection to recirculation mode of operation. There are over 30 minutes available to perform the manual actuation operations involved in switchover which is well in excess of the time required to perform the actions.

C. Primary System

Operation with less than three primary coolant loops in service is not permitted, as loss-of-coolant accidents in this mode of operation have not yet been analyzed. The loop isolation valves listed below shall, during power operation, have the A.C. power removed from the motor operators with the valves locked in the open position in order to ensure primary system operation as stated above.

Unit 1	Unit 2
MOV 1590	2590
MOV 1591	2591
MOV 1592	2592
MOV 1593	2593
MOV 1594	2594
MOV 1595	2595

We conclude that the above procedures will prevent operation with less than three loops. These valves, which provide certain flexibility for the licensee, do not serve a functional consideration in assessment of facility safety.

D. Boric Acid Build-up During Long Term, Post LOCA Core Cooling

The licensee submitted the Surry Emergency operating procedures proposed for the long term post-LOCA core cooling period and indicated that these procedures would prevent excessive concentration of boron in the reactor vessel. The procedures were supported by a Westinghouse analyses[5]. We have reviewed the analyses and proposed procedures. We believe that the analyses are not sufficiently complete to justify the licensee's emergency procedures. They do demonstrate that the existing ECCS system can be operated in a manner that will prevent excessive boric acid concentration from occurring, provided certain of the proposed procedures are changed. We have required these changes on an interim basis until such time as the licensee has completed further analysis, and we have accordingly reviewed the analyses and modified the Technical Specifications. The procedural changes we have required at this time provide additional margin between the boron concentration and the solubility limit at the mode. Specifically, the licensee has committed[6] and the Technical Specifications provide for the modification of the long-term core cooling procedures to effect switchover occurring at 16 hours of cold leg injection instead of at 24 hours. Simultaneous hot and cold leg injection would be provided by the low head and safety injection pumps, respectively. The staff has found this procedure to be acceptable.

Evaluation Conclusions for ECCS

The staff has completed its review of the Surry Power Station Units 1 and 2 reanalysis and has concluded:

1) As long as the plant is operated within the proposed Technical Specification limits it will meet the criterion of 10 CFR 50.46.

2) The ECCS minimum containment pressure calculations were performed in accordance with Appendix K of 10 CFR Part 50.

3) The single failure criterion will be satisfied provided that the proposed locking out of power to the specified motor-operated valves as noted above is implemented in accordance with the required modifications to the Technical Specifications.

4) The alternate long-term core cooling procedures adopted by the licensee are acceptable to the staff. The implementation of these procedures prior to plant start-up is required to provide assurance that the ECCS can be operated in a manner which would prevent excessive boric acid concentration from occurring.

Power Distribution Monitoring

The licensee's loss-of-coolant accident analysis for Units 1 and 2 requires a core peaking factor of 2.10 for full power operation. The licensee has proposed technical specifications to implement constant axial offset control (CAOC) procedures to ensure that the core peaking factor will not exceed 2.10 in normal operation including load following maneuvers. The licensee has installed alarms in both Units 1 and 2 to warn against deviations from the CAOC procedures in the ranges of: (1) greater than the 90% power level and (2) between 50 and 90% power levels.

VEPCO submitted the results of a core physics analysis supporting the use of a +6 to -9% flux difference band for CAOC instead of the $\pm 5\%$ band in previous proposed technical specifications. This submittal satisfies the linear power density (kw/ft) requirements of the LOCA analysis.

In addition, the licensee submitted information on the preliminary results of an analysis to support a +8 to -12% flux difference target band, which are relevent to DNB evaluations. These show the relative performance of axial power shapes permitted by CAOC to the shape assumed for the DNB design limit, a 1.55 axial chopped cosine and an $F_{\Delta H}$ of 1.55 for 118% overpower (including cases initially at part power) and conditions simulating loss of flow. In all cases the power distributions permitted by CAOC are less limiting than the design power distribution. We conclude that the constant axial offset control monitoring procedures with an allowable flux difference band of +6 to -9% will ensure that the reactor peaking factor will not exceed 2.10 in normal reactor operation, as assumed in the LOCA analysis. The revised technical specification changes submitted by VEPCO in the June 6, 1975 letter implement the CAOC procedures, described above, and are acceptable.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: JUN 1 6 1975

References

1. "Order for Modification of License" letter sent to Virginia Electric & Power Company from Robert A. Purple, December 27, 1974.

2. "Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," October 15, 1974.

3. Supplement to the Status Report by the Directorate of Licensing in the Matter of Westinghouse Electric Company ECCS Evaluation Model Conformance to 10 CFR 50, Appendix K," November 13, 1974.

4. No.-326, letter to K. R. Goller from S. Ragone, Virginia Electric & Power Company, December 31, 1974.

5. CLC-NS-309, letter to T. M. Novak from C. L. Caso, Westinghouse Nuclear Energy Systems, April 1, 1975.

6. No.500-S, letter to K. R. Goller from C. M. Stallings, Virginia Electric & Power Company, June 6, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC & POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments No. 7 to Facility Operating Licenses No. DPR-32 and DPR-37 issued to Virginia Electric & Power Company (licensee) which revised Technical Specifications for operation of the Surry Power Station, Units 1 and 2, located in Surry County, Virginia. The amendment for Unit 2 is effective as of the date of issuance, and for Unit 1 within ten days after date of issuance.

The amendments revise the provisions of the Technical Specifications related to the emergency core cooling system (ECCS). These revisions are based on the licensee's reevaluation of the ECCS performance and are consistent with the requirements of 10 CFR 50 Part 50.46.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on May 1, 1975 (40 FR 19043). No request for a

hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated April 15, 1975, as supplemented May 1, May 20, June 6, June 9 and June 11, 1975, (2) Amendments No. 7 to Licenses No. DPR-32 and DPR-37, with Changes No. 22, (3) the Commission's related Safety Evaluation, and (4) the Commission's Negative Declaration dated May 15, 1975, which is being published concurrently with this notice, and associated Environmental Impact Appraisal. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Swem Library, College of William & Mary, Williamsburg, Virginia 23185.

A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 16th day of June 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Purple

Robert A. Purple, Chief
Operating Reactors Branch #1
Division of Reactor Licensing