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

January 6, 1988

Docket Nos. 50-280
and 50-281

Mr. W. L. Stewart
Vice President - Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CONTROL ROD ASSEMBLIES AND SURRY IMPROVED FUEL (TAC NOS. 63166, 63167, 65432, 65433, 65561 AND 65562)

The Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. DPR-32 and Amendment No. 116 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your three applications transmitted by letters dated: October 7, 1986, as supplemented June 8, 1987; April 1, 1987; and May 26, 1987.

These amendments revise Section 3.12 of the Surry Technical Specifications (TS) by revising the actions to be taken by the licensee while operating with an inoperable, misaligned or dropped control rod. Also, the fully withdrawn position of all rod cluster control assembly (RCCA) banks is redefined to minimize localized RCCA wear.

Finally, these amendments permit the operation of Surry Units 1 and 2 with 15 x 15 Surry Improved Fuel (SIF) Assemblies, in addition to the Westinghouse Low Parasitic 15 x 15 (LOPAR) Fuel Assemblies, during Cycle 10. The LOPAR fuel assemblies will eventually be replaced by the SIF assemblies.

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

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 116 to DPR-32
2. Amendment No. 116 to DPR-37
3. Safety Evaluation

cc: w/enclosures
See next page

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

Sincerely,

Original signed by

Chandu P. Patel, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-32

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

   A. The applications for amendment by Virginia Electric and Power Company (the licensee) dated October 7, 1986, as supplemented June 8, 1987; April 1, 1987; and May 26, 1987, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 116 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 6, 1988

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-37

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

   A. The applications for amendment by Virginia Electric and Power Company (the licensee) dated October 7, 1986, as supplemented June 8, 1987; April 1, 1987; and May 26, 1987, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 116 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 6, 1988

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 116 FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 116 FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 2.1-2	TS 2.1-2
TS 2.1-3	TS 2.1-3
TS 2.1-4	TS 2.1-4
TS 2.1-5	TS 2.1-5
TS 2.3-8	TS 2.3-8
TS 3.1-5a	TS 3.1-5a
TS 3.12-1	TS 3.12-1
TS 3.12-8	TS 3.12-8
TS 3.12-9	TS 3.12-9
TS 3.12-10	TS 3.12-10
TS 3.12-11	TS 3.12-11
TS 3.12-13	TS 3.12-13
TS 3.12-16	TS 3.12-16
- - -	TS Table 3.12-1
TS Figure 3.12-1A	TS Figure 3.12-1A
TS Figure 3.12-1B	TS Figure 3.12-1B
TS Figure 3.12-2	TS Figure 3.12-2
TS Figure 3.12-3	TS Figure 3.12-3
TS Figure 3.12-5	TS Figure 3.12-5
TS Figure 3.12-6	TS Figure 3.12-6

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 the rated power.

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, DNB has been correlated to thermal power, reactor coolant temperature and reactor coolant pressure which are observable parameters. This correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the DNB heat flux at a particular core location to the local heat flux, is indicative of the margin to DNB. The DNB basis is as follows: there must be at least a 95% probability with 95% confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is based on the entire applicable experimental data set to meet this statistical criterion.[1]

The curves of TS Figure 2.1-1 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (three loop operation) represent limits equal to, or more conservative than, the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which the calculated DNBR is not less than the design DNBR limit or the average enthalpy at the exit of the vessel is equal to the saturation value. The area where clad integrity is assured is below these lines. The temperature limits are considerably more conservative than would be required if they were based upon the design DNBR limit alone but are such that the plant conditions required to violate the limits are precluded by the self-actuated safety valves on the steam generators. The three loop operation safety limit curve allows for heat flux peaking effects due to fuel densification and applies to 100% of design flow. The effects of rod bowing are also considered in the DNBR analyses.

The curves of TS Figure 2.1-2 and 2.1-3 which show the allowable power level decreasing with increasing temperature at selected pressures for constant flow (two loop operation), represent limits equal to, or more conservative, than the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the calculated DNBR is equal to the design DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the calculated DNBR reaches the design DNBR limit and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figures 2.1-1 through 2.1-3 are based on a $F^N_{\Delta H}$ of 1.55, a 1.55 cosine axial flux shape and a DNB analysis procedure including the fuel densification power spiking[(4)] as part of the generic margin to accommodate rod bowing.[(5)(6)] TS Figure 2.1-1 is also valid for the following limit of the enthalpy rise hot channel factor: $F^N_{\Delta H} = 1.55 \ (1 + 0.3 \ (1-P))$ where P is the fraction of rated power. TS Figures 2.1-2 and 2.1-3 include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figures 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit(3) based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 per cent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

The fuel overpower design limit is 118% of rated power. The over-power limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin will prevent the minimum value of the DNBR from going below the applicable design limit during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for three-loop operation. During two-loop operation with the loop stop valves in the inactive loop open, and the overtemperature ΔT trip setpoint is adjusted to the value specified for this condition, a reactor trip at 60% power will prevent the minimum value of DNBR from going below the applicable design limit during normal operational transients and anticipated transients when only two loops are in operation. During two-loop operation with the inactive loop stop valves closed and the overtemperature ΔT trip setpoint is adjusted to the value specified for this condition, a reactor trip at 65% power will prevent the minimum DNBR from going below the applicable design limit during normal operational transients and anticipated transients.

Although not necessary for core protection, other reactor trips provide additional protection. The steam/feedwater flow mismatch which is coincident with a low steam generator water level is designed for and provides protection from a sudden loss of the reactor's heat sink. Upon the actuation of the safety injection circuitry, the reactor is tripped to decrease the severity of the accident condition. Upon turbine trip, at greater than 10% power, the reactor is tripped to reduce the severity of the ensuing transient.

References

(1) FSAR Section 14.2.1

(2) FSAR Section 14.2

(3) FSAR Section 14.5

(4) FSAR Section 7.2

(5) FSAR Section 3.2.2

(6) FSAR Section 14.2.9

(7) FSAR Section 7.2

b. With one Reactor Vessel Head vent path inoperable; startup and/or power operation may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of both isolation valves in the inoperable vent path.

c. With two Reactor Vessel Head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuator of all isolation valves in the inoperable vent paths, and restore at least one of the vent paths to operable status within 30 days or be in hot shutdown within 6 hours and in cold shutdown within the following 30 hours.

Basis

Specification 3.1.A-1 requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling flow in the event of a loss of reactor coolant flow accident. This provided flow will maintain the DNBR above the applicable design limit.[1] Heat transfer analyses also show that reactor heat equivalent to approximately 10% of rated power can be removed with natural circulation; however, the plant is not designed for critical operation with natural circulation or one loop operation and will not be operated under these conditions.

When the boron concentration of the Reactor Coolant System is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the reactor coolant system volume in approximately one half hour.

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 or 3.12-1B for three-loop operation and TS Figures 3.12-4A or 3.12-4B 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 cal-

ΔT and Overtemperature ΔT trip settings shall be reduced by the equivalent of 2% power for every 1% quadrant to average power tilt.

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 group step demand position by more than ±12 steps. Additionally, a full-length control rod shall be considered inoperable if its rod drop time is greater than 2.4 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 mechanism, (i.e. programming circuitry), the provisions of Specifications 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.

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

5. Power operation may continue with one rod inoperable provided that within one hour either:

   a. the rod is no longer inoperable as defined in Specification 3.12.C.1, or

b. the rod is declared inoperable and the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. Operation at power may then continue provided that:

1) either:

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

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

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

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

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

6. If power has been reduced in accordance with Specification 3.12.C.5.b, power may be increased above 75% power provided that:
   a) an analysis has been performed to determine the hot channel factors and the resulting allowable power level based on the limits of Specification 3.12.B.1, and
   b) an evaluation of the effects of operating at the increased power level on the accident analyses of Table 3.12-1 has been completed.

D. Core Quadrant Power Balance:

1. 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:
   a. Once per day, and
   b. After a change in power level greater than 10% or more than 30 inches of control rod motion.

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

E. Rod Position Indicator Channels

1. The rod position indication system shall be operable and capable of determining the control rod positions within ±12 steps.

2. If a rod position indicator channel is out of service, then:
   a. For operation above 50% of rated power, the position of the RCC shall be checked indirectly using core instrumentation (excore detectors and/or incore thermocouples and/or movable incore detectors) at least once per 8 hours and immediately after any motion of the non-indicating rod exceeding 24 steps, or

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

3. If more than one rod position (RPI) indicator channel per group or two RPI channels per bank are inoperable, then the requirements of Specification 3.0.1 will be followed.

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 acheiving 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 in service, the effects of malpositioned control rod assemblies are observable from nuclear and process information displayed in the Main Control Room and by core thermocouples and in-core movable detectors. Below 50% power, no special monitoring is required for malpositioned control rod assemblies with inoperable rod position indicators because, even with an unnoticed complete assembly misalignment (full length control rod assembly 12 feet out of alignment with its bank), operation at 50% steady state power does not result in exceeding core limits.

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

be compensated for by tighter axial control. Four percent is the appropriate allowance for measurement uncertainty for $F^N_{\Delta H}$ obtained from a full core map ($\geq$ 38 thimbles, including a minimum of 2 detectors per core quadrant, 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.

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

1. Control 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 TS Figures 3.12-1A, 3.12-1B.

3. The full length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference

TABLE 3.12-1
ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Large and Small Break Loss of Coolant Accidents

Single Reactor Coolant Pump Locked Rotor

Major Secondary Pipe Rupture

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

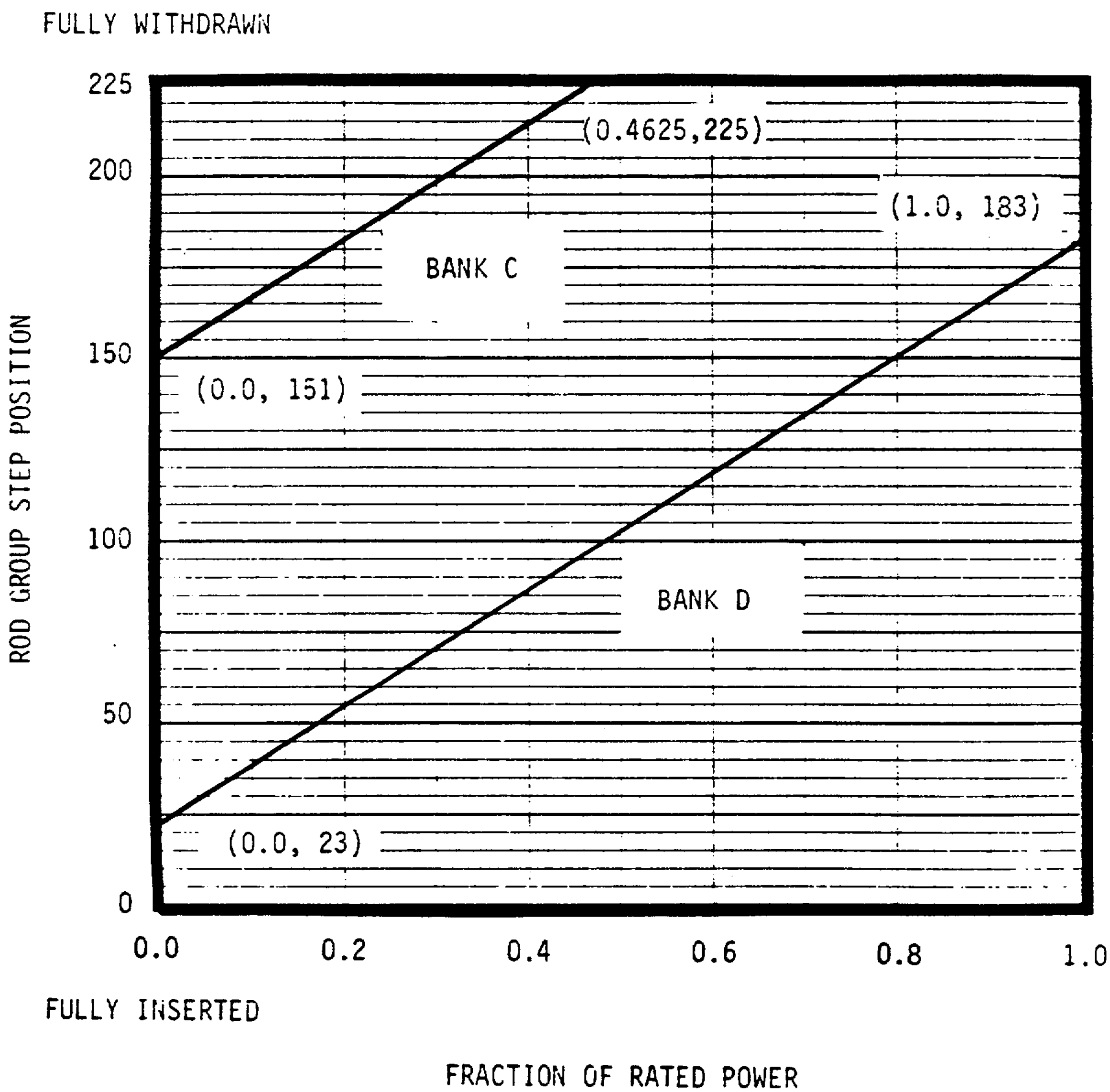

FIGURE 3.12-1A CONTROL BANK INSERTION LIMITS FOR 3-LOOP NORMAL OPERATION-UNIT 1

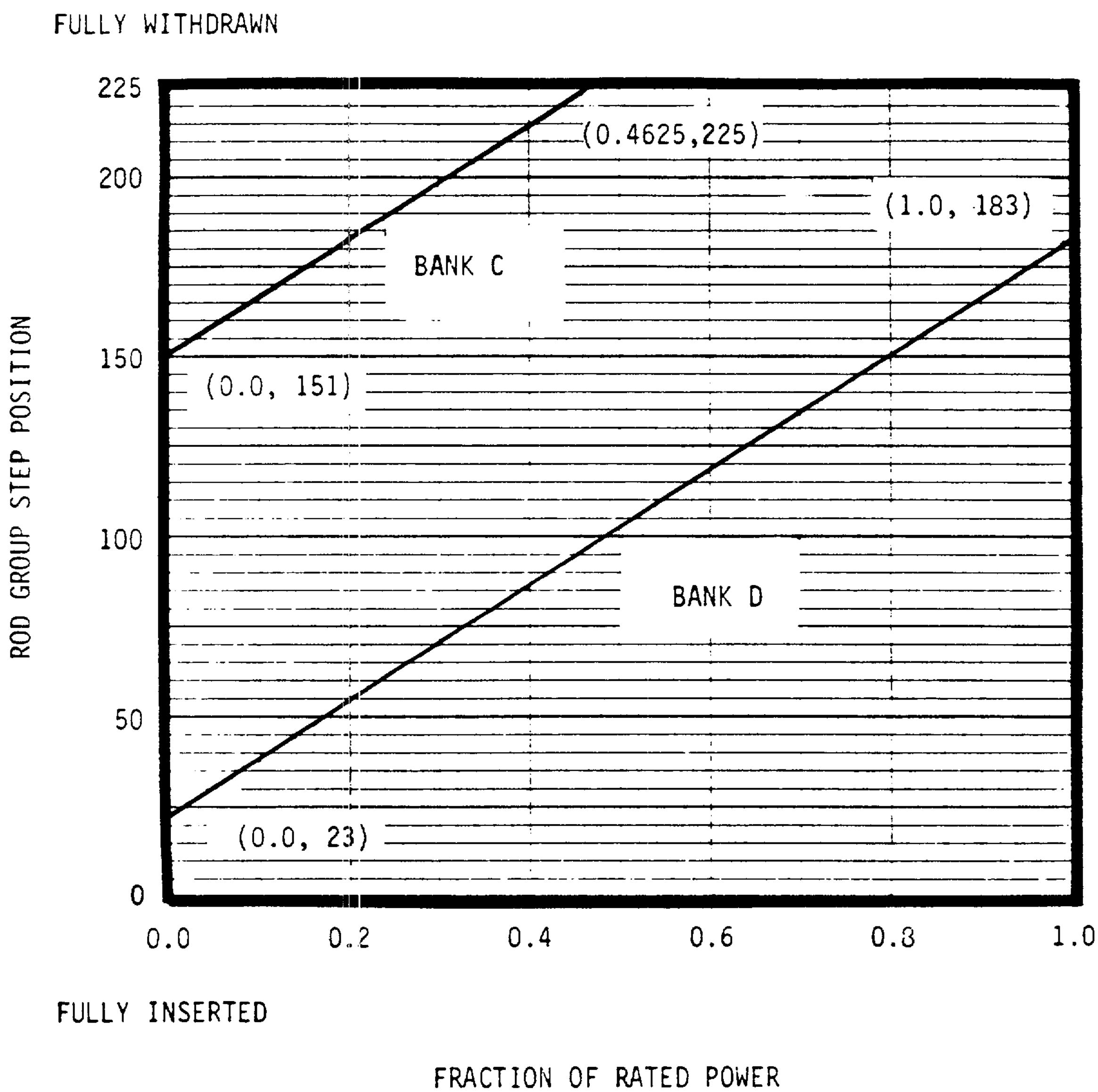

FIGURE 3.12-1B CONTROL BANK INSERTION LIMITS FOR NORMAL 3 LOOP OPERATION - UNIT 2

DELETE

DELETE

DELETE

DELETE

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

INTRODUCTION

By letters dated October 7, 1986, as supplemented June 8, 1987; April 1, 1987; and May 26, 1987, Virginia Electric and Power Company (the licensee) requested amendments to Facility Operating License Nos. DPR-32 and DPR-37, issued to the licensee for operation of the Surry Nuclear Power Station, Units 1 and 2, located in Surry County, Virginia.

By letter dated October 7, 1986, as supplemented June 8, 1987, the licensee proposed to revise Section 3.12 of the Surry Technical Specifications (TS) by revising the actions to be taken by the licensee while operating with an inoperable, misaligned or dropped control rod.

By letter dated April 1, 1987, the licensee requested to revise Figures 3.12-1A and 3.12-1B of the Surry TS, which govern the control rod insertion limit. The licensee proposed redefining the fully withdrawn position of all rod cluster control assembly (RCCA) banks to 225 steps, instead of the current 228 steps. The change will allow greater operational flexibility with regard to control rod bank positioning as a means of minimizing localized RCCA wear.

By letter dated May 26, 1987, the licensee requested changes in the TS to support the planned fuel design change from the Westinghouse Low Parasitic (LOPAR) 15 x 15 Fuel Assembly to the 15 x 15 Surry Improved Fuel (SIF) Assembly during Cycle 10 for both Surry units.

DISCUSSION AND EVALUATION

Control Rod Insertion Limits

By letter dated October 7, 1986, as supplemented June 8, 1987, the licensee proposed to revise Section 3.12 of the Surry TS. The proposed revision would alter the manner in which inoperable rods are treated. Currently, when an inoperable rod is discovered, alternate insertion limits are invoked depending on the position of the inoperable rod and whether or not it is stuck. Operation may then continue indefinitely with the new limits. The limits are pre-calculated so that the limiting inoperable rod condition is covered.

The licensee has proposed to revise the actions required upon discovery of an inoperable rod so that they more nearly match those of the Westinghouse Standard Technical Specifications. The proposed TS would require that the rod be repaired within 1 hour, or that either the power be reduced to 75% of full power within the next hour or the bank be aligned to the position of the inoperable rod and the permitted power be determined by the bank position. Reduction in power for the misaligned rod is not required for 8 hours in the current TS. The proposed TS would also require that the shutdown margin be determined within 1 hour and at least once per 12 hours thereafter. This is an additional action not required in the current TS. The proposed TS are thus conservative with respect to the current ones, and are therefore acceptable.

The proposed TS would also require that the hot channel factors ($F_Q(Z)$ and $F^N_{\Delta H}$) be shown to be within the limits within 72 hours from the time of discovery of the inoperable rod. This action is required within 8 hours in the present TS for the purpose of determining the necessity for power reduction below 75% of full power. Considering other immediate actions required by the licensee, the staff finds the 72 hour time limit to be acceptable. However, the staff requested that confirmation (by measurement or calculation) be made within the same time frame, which is required for $F_Q(Z)$ and $F^N_{\Delta H}$ that the value of the axially dependent radial or planar peaking factor ($F_{xy}(Z)$) used in the constant axial offset control analysis is still valid. By letter dated June 8, 1987, the licensee submitted a revised proposal to change TS 3.12.C.5.b.3, which satisfied the staff's request.

The proposed TS would also require a reanalysis of the transients and accidents that are affected by the inoperable rod within 5 days to confirm that the previous analyses are valid. This is an expanded requirement from that in the present TS and is, therefore, acceptable.

In the present TS, the power may be increased to greater than 75% of full power after the determination of the acceptability of the hot channel factors. The proposed TS include an additional requirement that the effect of increased power operation on the accident analyses be determined prior to increasing the power. Thus, this change is acceptable.

Based on the above evaluation, the staff concludes that the proposed revisions to the Surry TS are acceptable.

Fully Withdrawn Control Rods

By letter dated April 1, 1987, the licensee proposed to redefine the fully withdrawn position of all RCCA banks to 225 steps, which would eliminate localized RCCA wear at the top of the control rods. The current fully withdrawn position is 228 steps. At 225 steps withdrawn, the RCCAs are only 0.31 inches into the active fuel region (228 steps withdrawn is above the active fuel region). Since the top region of the core has such low worth, the effect of the proposed change was expected to be minimal. To confirm this, neutron calculations were performed by the licensee.

The shutdown margin calculations showed a 0.003% $\Delta\rho$ decrease at beginning-of-life (BOL) and a 0.006% $\Delta\rho$ decrease at end-of-life (EOL) (recent cycles have excess shutdown margin of approximately 1.75% $\Delta\rho$ at BOL and 1.6% $\Delta\rho$ at EOL). The changes in the other parameters which would be affected are similarly minimal with respect to available margins.

Because the proposed change will result in slight insertion of the RCCAs into the active fuel region, the staff expects essentially negligible effects due to the proposed change. Therefore, the staff concludes that the proposed change is acceptable.

Surry Improved Fuel

By letter dated May 26, 1987, the licensee proposed amendments which support the planned fuel design change from the Westinghouse Low Parasitic (LOPAR) 15 x 15 Fuel Assembly to the 15 x 15 Surry Improved Fuel (SIF) Assembly.

The Surry units currently use the Westinghouse LOPAR fuel assemblies. In recent years, Westinghouse offered two advanced fuel designs, known as the Optimized Fuel Assemblies (OFA) and the VANTAGE 5 assemblies. The proposed SIF fuel is similar to the OFA fuel but includes some features of the VANTAGE 5 fuel.

The most significant differences between the LOPAR and SIF fuel assemblies are common to both the OFA and VANTAGE 5 assemblies. These include the use of Zircaloy grids instead of Inconel grids, smaller diameter thimble tubes, and three-leaf holddown springs instead of two-leaf springs. VANTAGE 5 features in the SIF design include slightly shorter nozzles, which result in slightly longer thimble tubes and fuel rods and a removable top nozzle.

In addition to use of the SIF fuel assemblies, the licensee has proposed to eliminate the use of thimble plugs in reload cores. These devices are used to inhibit flow in those assemblies which do not have either control rods or burnable poison assemblies. Their elimination results in a significant increase in core bypass flow and a slight increase in overall core flow.

Both the OFA and VANTAGE 5 fuel designs have been approved for use in Westinghouse reactors. The major effect on the neutronic behavior of the core from the use of these fuels is due to the increase in the drop time for the control rods. This increase is 0.6 seconds (from 1.8 to 2.4 seconds), due to the reduction in diameter of the thimble tubes. There is no change in the fuel rod design (over its fueled length), and core neutronics parameters are not affected by the change in assembly design.

There are small changes in the thermal-hydraulic parameters of the core, due primarily to the bulkier grid straps in the advanced designs. The thermal-hydraulic comparability of the LOPAR and the advanced designs has been confirmed by hydraulic tests. Since the SIF fuel is essentially the same as the OFA fuel, these tests also apply to that fuel. A more significant difference in the thermal-hydraulic performance results from the removal of the thimble plugs from the core. Westinghouse calculates that the increased core bypass flow results in a 2% loss of DNBR margin due to the 1.5% flow decrease in the fuel rod channels. Thimble plug removal also results in a reduction to the fuel assembly hydraulic loss coefficient; however, Westinghouse hydraulic tests show that the reduced fuel assembly loss coefficient results in a net reduction in the hydraulic lift force which more than compensate for the slight increase in core flow rate.

The licensee performed an evaluation to confirm that all safety criteria will be met when the SIF fuel is substituted for the LOPAR fuel. Care was taken in the analysis to choose parameters which would bound transition cores as well as a core fully loaded with SIF fuel.

The SIF design is mechanically similar to the approved OFA and VANTAGE 5 fuel, and is expected to have the same mechanical response. Also, when compared to LOPAR fuel, the SIF design is expected to have the same mechanical response except that it is expected to experience less bowing. The Zircaloy grids result in larger lifting forces, but these are offset by use of the three-leaf holddown springs. These springs have been used on OFA and VANTAGE 5 fuel and their use on SIF fuel is acceptable.

The shorter bottom and top nozzle designs result in longer fuel rods, since the overall envelope of the assemblies is not changed. The fuel pellet stack length is not changed and as a result, the upper plenum length is increased. This will permit increased fuel burnup. Analysis has been performed to show that fuel design criteria for fission gas pressure is not violated for batch average discharge burnups as high as 45,000 MWD/MTU.

The effects of mixed core loading and thimble plug removal on fuel and control rod wear have been evaluated. It was concluded that the resultant cross flow has a negligible effect on fuel rod wear. In fact, the removal of the thimble plugs has a beneficial effect on control rod wear.

The similarity between the LOPAR and SIF fuel with respect to fuel rod diameter, rod-to-rod spacing and cladding results in negligible differences in neutronics parameters between the two fuels. Cycle-to-cycle variations in these parameters are dictated by fuel management policy, rather than by the differences in fuel design. These differences are evaluated for each cycle as part of the cycle design using approved reload methodology.

The hydraulic compatibility of LOPAR and OFA fuel has been confirmed by tests conducted at the Westinghouse Fuel Test Systems facility. These tests are also valid for the SIF assembly. The W-3 correlation is currently used for the DNB analysis of the Surry plants and will continue to be used for LOPAR fuel in the mixed core cycles. The SIF assemblies will be analyzed with the THINC-I code using the WRB-1 CHF correlation. This procedure has been approved for the OFA and VANTAGE 5 fuel and is also acceptable for SIF fuel.

The 95/95 limit for the WRB-1 correlation is 1.17. However, a plant-specific margin of 20% has been added to arrive at a design limit of 1.46 for the SIF fuel. The 1.3 limit value for the W-3 correlation contains an 18 percent margin for the LOPAR fuel. These margins are sufficient to account for the rod bow penalty, the transition core penalty and the impact of the removal of the thimble plugs.

The only significant changes in core characteristics that affect transient and accident evaluations are the increased scram time and thimble plug removal. The first of these changes affects the "fast" accidents such as rod ejection, loss of flow, and locked rotor. The second affects the large-break LOCA analysis. The first three events were reanalyzed using approved methods and all safety criteria were shown to be met. A reevaluation of the large-break LOCA resulted in a peak clad temperature of 1979°F, including a 10°F transition core penalty. This meets the acceptance criterion of 2200°F and is acceptable.

Based on the above review, the staff concludes that the use of SIF fuel in Surry Units 1 and 2 is acceptable. Further, the staff has reviewed the proposed TS changes and concludes that they are consistent with analyses provided and are acceptable.

ENVIRONMENTAL CONSIDERATION

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

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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 6, 1988

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