

REGULATORY DOCKET FILE COPY

JULY 25 1979

Docket Nos. 50-280
and 50-281

Mr. W. L. Proffitt
Senior Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Proffitt:

The Commission has issued the enclosed Amendment Nos. 50 and 49 to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. These amendments are in response to your application dated February 26, 1979.

The Amendments authorize the removal of all part-length control rods from both reactors.

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

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Sincerely,

Original signed by
M. Grotenhuis

for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

*Construct
CEO*

Enclosures:

1. Amendment No. 50 to DPR-32
2. Amendment No. 49 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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No legal objection to issue of amendment & notice. SER not reviewed.

TACS 11507 and 11508

OFFICE	DOR:ORB3	DOR:ORB1	DOR:ORB3	DOR:ORB1	OELD
SURNAME	JD [Signature]	DNeighbors	P. Kreutzer	A. Schwencer	CUTCHER
DATE	07/11/79	07/11/79	07/17/79	07/25/79	07/19/79



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

Docket Nos. 50-280
and 50-281

July 25, 1979

Mr. W. L. Proffitt
Senior Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

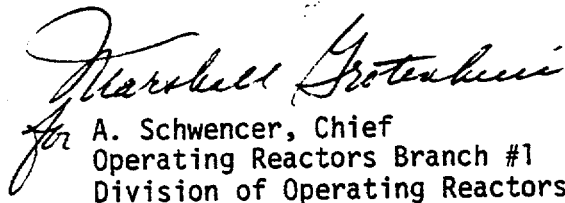
Dear Mr. Proffitt:

The Commission has issued the enclosed Amendment Nos. 50 and 49 to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Units 1 and 2. These amendments are in response to your application dated February 26, 1979.

The amendments authorize the removal of all part-length control rods from both reactors.

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

Sincerely,


for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:

1. Amendment No. 50 to DPR-32
2. Amendment No. 49 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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

- 2 -

July 25, 1979

cc: Mr. Michael W. Maupin
Hunton and Williams
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Richmond, Virginia 23213

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Atlanta, Georgia 30303



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. 50
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated February 26, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and,
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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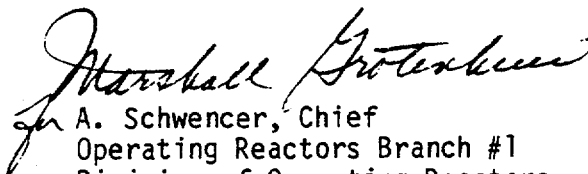
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION


for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: July 25, 1979



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. 49
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Virginia Electric and Power Company (the licensee) dated February 26, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and,
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

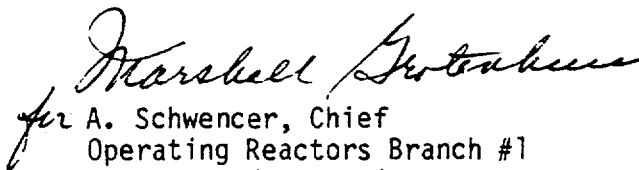
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49, 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.

FOR THE NUCLEAR REGULATORY COMMISSION


for A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the
Technical Specifications

Date of Issuance: July 25, 1979

ATTACHMENT TO LICENSE AMENDMENT NOS.50 AND 49
FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37
DOCKET NOS. 50-280 AND 50-281

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

<u>Remove</u>	<u>Insert</u>
3.12-2	3.12-2
3.12-9	3.12-9
3.12-10	3.12-10
3.12-11	3.12-11
3.12-12	3.12-12
3.12-13	3.12-13
3.12-16	3.12-16
5.3-2	5.3-2

culations and physics data obtained during Unit Startup and subsequent operation, will be permitted.

- c. The shutdown margin with allowance for a stuck control rod assembly shall be greater than or equal to 1.77% reactivity under all steady-state operation conditions, except for physics tests, from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions ($T_{avg} > 547^{\circ} F$) if all control rod assemblies were tripped, assuming that the highest worth control rod assembly remained fully withdrawn, and assuming no changes in xenon, or boron.
4. Whenever the reactor is subcritical, except for physics tests, the critical 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. Deleted
6. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. However, the shutdown margin indicated above 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.

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.

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

- 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 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 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 requirement 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. The shutdown margin for the entire cycle length is established at 1.77% reactivity. All other accident analyses with the exception of the chemical and volume control system malfunction analysis 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.

The various control rod assemblies (shutdown banks, control banks A, B, C and D) are each to be moved as a bank, that is, with 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 (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 DNER in the core must not be less than 1.30 in normal operation or in short term transients.

In addition to the above, the peak linear power density, the nuclear enthalpy rise hot channel factor, and the hot assembly enthalpy rise factor must not exceed their limiting 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.

Amendment No. 50, Unit 1
Amendment No. 49, Unit 2

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor, $F_{\Delta H}^N$, 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, and 3.12-2.
3. The full length control bank insertion limits are not violated.

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4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux difference refers to the difference between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

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

Amendment No. 50 , Unit 1
Amendment No. 49 , Unit 2

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will not exceed 3.60 weight percent 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 clad with stainless steel.
5. There are 48 full-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.
6. Surry Unit 1, Cycle 4, Surry Unit 2, Cycle 3, and subsequent cores will meet the following criteria at all times during the operation lifetime.
 - a. Hot channel factor limits as specified in Section 3.12 shall be met.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 50 AND 49 TO

FACILITY OPERATING LICENSE NOS. DPR-32 AND DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

Introduction

By letter dated February 26, 1979, Virginia Electric and Power Company (the licensee) requested amendments to Facility Operating Licenses DPR-32 and 37 for Surry Units 1 and 2. The proposed amendments would revise the Technical Specifications to permit removal of the part length control rods. This has been approved for several other Westinghouse reactors.

Discussion and Evaluation

The five (5) part length control rods were initially installed to give the operator the ability to suppress xenon induced power oscillations in the axial direction.

The Technical Specifications, as now written, require that these part length rod cluster control assemblies (PLRCCAs) be withdrawn and excluded from the core at all times during reactor operations. The PLRCCAs are not needed, used or assumed to be available to achieve required reactor shutdown conditions. The proposed removal, therefore, will not cause any change in required reactivity characteristics, or safety margins at full power, low power or shutdown. To the contrary, removal will eliminate the potential for part length rods dropping into the core during operation. Such an event could cause abnormal flux distribution or reactor shutdown.

In addition, in order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCAs were to be performed, the licensee will install thimble plug assemblies in the spaces previously occupied by PLRCCAs. If found to be necessary during future cycles, the licensee may replace these thimble plug assemblies with either burnable poison rods, neutron source rods, or full length control rods.

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The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut. All components in the thimble plug assembly, except for the springs, are constructed from type 304 stainless steel.

The thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCAs have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Summary

Based on the considerations that 1) the PLRCCAs are not needed for reactor operation, 2) that removal of these assemblies will remove the chance for an abnormal flux distribution or reactor shutdown because of a dropped part length rod and 3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §1.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: July 25, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION
DOCKET NOS. 50-280 AND 50-281
VIRGINIA ELECTRIC AND POWER COMPANY
NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY
OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 50 and 49 to Facility Operating License Nos. DPR-32 and DPR-37 issued to Virginia Electric and Power Company, which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2 (the facilities) located in Surry County, Virginia. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to allow removal of all part-length control rods from both reactors.

The application for the amendments 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 in 10 CFR Chapter I, which are set forth in the license amendments.

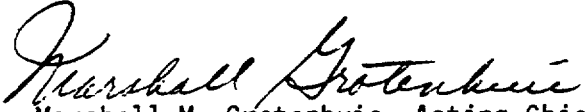
-2-

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendment dated February 21, 1978 (2) Amendment Nos.50 and 49 to License Nos. DPR-32 and DPR-37, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H. Street, N. W., Washington, D. C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 25th day of July.

FOR THE NUCLEAR REGULATORY COMMISSION


Marshall M. Grotenhuis, Acting Chief
Operating Reactors Branch #1
Division of Operating Reactors