

Docket File

MARCH 4 1980

Docket Nos. 50-280
and 50-281

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Mr. J. H. Ferguson
 Executive Vice President - Power
 Virginia Electric and Power Company
 Post Office Box 26666
 Richmond, Virginia 23261

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment Nos. *57* and *56* to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 15, 1978.

The amendments revise the Technical Specifications to specify new pressure-temperature limits for heatup and cooldown operations.

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

Sincerely,

Original Signed By

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

Enclosures:

1. Amendment Nos. *57* and *56* to DPR-32 and DPR-37
2. Safety Evaluation
3. Notice of Issuance

cc: w/enclosures
 See next page

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SURNAME	JDNeighbors	CS Parrish	ASchwencer	WPGammill	
DATE	02/ /80	02/ /80	02/ /80	02/ /80	02/ /80



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

March 4, 1980

Docket Nos. 50-280
and 50-281

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Ferguson:

The Commission has issued the enclosed Amendment Nos. 57 and 56 to Facility Operating License Nos. DPR-32 and DPR-37 for the Surry Power Station, Unit Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated December 15, 1978.

The amendments revise the Technical Specifications to specify new pressure-temperature limits for heatup and cooldown operations.

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

Sincerely,

A handwritten signature in cursive script, appearing to read "A. Schwencer".

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

Enclosures:

1. Amendment Nos. 57 and 56
to DPR-32 and DPR-37
2. Safety Evaluation
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cc: w/enclosures
See next page

Mr. J. H. Ferguson
Virginia Electric and Power Company - 2 -

March 4, 1980

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Richmond, Virginia 23213

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Surry, Virginia 23883

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Board of Supervisors of Surry County
Surry County Courthouse, Virginia 23683

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Council on the Environment
903 Ninth Street Office Building
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Mr. James R. Wittine
Commonwealth of Virginia
State Corporation Commission
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U. S. Environmental Protection Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building - 6th Floor
6th and Walnut Streets
Philadelphia, Pennsylvania 19106



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. 57
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 December 15, 1978, 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-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. 57, 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



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

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 4, 1980



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. 56
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 December 15, 1978, 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 amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 56, 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



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

Attachment:
Changes to
Technical Specifications

Date of Issuance: March 4, 1980

ATTACHMENT TO LICENSE AMENDMENT NOS. 57 AND 56
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 certain vertical lines indicating the area of change.

<u>Remove</u>	<u>Insert</u>
3.1-7	3.1-7
3.1-8	3.1-8
3.1-10	3.1-10
3.1-11	3.1-11
3.1-12	3.1-12
TS Fig. 3.1-1	TS Fig. 3.1-1
TS Fig. 3.1-2	TS Fig. 3.1-2
	TS Fig. 3.1-3
4.2-34	4.2-34

3. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr. and 200°F/hr., respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.
4. TS Figure 3.1-1 shall be updated periodically in accordance with the following procedures, before the calculated maximum exposure of the vessel exceeds the exposure for which TS Figure 3.1-1 applies. The curve based on 0.25% Cu weld in TS Figure 3.1-2 shall be used to predict the increase in transition temperature based on integrated power.
 - a. If measurements on the most recently examined irradiation specimen show that its data point is above the 1/4T (thickness) line of T.S. Figure 3.1-3 then a new line shall be constructed through the origin such that it is above all the applicable data points. Once T.S. Figure 3.1-3 is revised, T.S. Figure 3.1-1 must be updated, either by a temperature shift, as by T.S. 3.1.B.4c below, or by revising the applicable period (EFPY) to match the new transition temperature from TS Figure 3.1-2.
 - b. At or before the end of the integrated power period for which TS Figure 3.1-1 applies, the limit lines on the figure shall be updated for a new integrated power period as follows. The total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated neutron exposure. The predicted increase in transition temperature at the end of the new period shall then be obtained from TS Figure 3.1-2.

- c. The limit lines in TS Figure 3.1-1 shall be moved parallel to the temperature axis (horizontally) in the direction of increasing temperature a distance equivalent to the transition temperature increase obtained from TS Figure 3.1-2, less the increment used for the end of the present period.

Basis

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. (1) These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Section 4.1 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure are limited. The maximum plant heatup and cooldown rate of 100°F/hr. is consistent with the design number of cycles and satisfies stress limits for cyclic operation. (2)

The allowable pressure vs. temperature is based on a temperature scale relative to the RT_{NDT} . The RT_{NDT} is basically the drop weight NDTT of the material, as determined by ASTM E208. However, to assure that this value is conservative, and to guard against the possibility that material with low upper shelf toughness, or with a low rate of increase of toughness with temperature, is not properly evaluated, Charpy tests are also performed. If 35 mils lateral

available for the core region weld material but on the basis of actual drop weight data on many similar weld materials, plus the actual Charpy values on this material, the drop weight NDTT is estimated to be 0°F.

The RT_{NDT} for the first two years of operation included a conservative estimate of the shift in RT_{NDT} caused by radiation of 100°F. This added to the original RT_{NDT} of 0°F assumed for the welds, gave a reference RT_{NDT} of 100°F to be used for the first two years of operation, or until the radiation shift was estimated to be over 100°F.

In examining the data for the rest of the material in the vessel; as well as the properties for the other ferritic components of the reactor system, it is certain that all other materials initially had RT_{NDT} values significantly lower than 100°F.

Since the neutron spectra at the samples and vessel inside radius are identical, the measured $(RT)_{NDT}$ shift for a sample can be supplied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel is obtainable from the measured sample data by appropriate application of the calculated azimuthal neutron flux variation.

During cooldown and steady state, the thermal stress varies from tensile at the inner wall to compressive at the outer wall. The internal pressure superimposes a tensile stress on this thermal stress pattern, increasing the stress at the inside wall and relieving the stress at the outside wall. Therefore, the limiting stress always appears at the inside wall and the limit line has a

direct dependence on cooldown rate. For heatup, the thermal stress is reversed and the location of the limiting stress is a function of heatup rate. The 1/4T location is considered conservative since the enhanced metallurgical properties of the surface are not used for the determination of NDTT. The 1/4T location is used for cooldown and steady state and 3/4T location is used for heatup but the 1/4T location is the most restrictive so it will be the controlling curve. In addition, the limiting NDTT for the reactor vessel after operation is based on the NDTT shift due to irradiation. Since the fast neutron dose is highest at the inner surface, usage of the 1/4T NDTT criterion is conservative (FSAR Section 4). The 50°F/hr. heatup and cooldown line on TS Figure 3.1-1 bounds all limit lines for heatup and cooldown rates up to 50°F/hr. for indicated temperatures at or below 440°F, and 100°F/hr. above 440°F. TS Figure 3.1-1 is based on the Standard Review Plan as modified by measured irradiation sample temperature shifts and appropriate vessel attenuation factors and azimuthal neutron flux variations.

TS Figure 3.1-1 defines stress limitations only. For normal operation other inherent plant characteristics, e.g., pump parameter and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure ranges.

The heatup and cooldown rate of 100°F/hr. for the steam generator is consistent with the remainder of the Reactor Coolant System, as discussed in the first paragraph of the basis. The stresses are within acceptable limits for the anticipated usage.

Temperature requirements for the steam generator correspond with the measured NDT for the shell. The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

References:

- (1) FSAR, Section 4.1.5
- (2) ASME Boiler & Pressure Vessel Code, Section III, N-415
- (3) ASME Boiler & Pressure Vessel Code, Section III, proposed non-mandatory Appendix G2000
- (4) 10 CFR 50, Appendix A, G, & H
- (5) Regulatory Guide 1.99, Revision 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"
- (6) USNRC Standard Review Plan, Section 5.3.2, 11/29/75, "Pressure - Temperature Limits"
- (7) Welding Research Council (WRC) Bulletin 175, "PVRC Recommendation on Toughness Requirements for Ferritic Materials"
- (8) WCAP - 7924-A, "Basis for Heatup and Cooldown Limit Curves"
- (9) Surry Reactor Vessel Radiation Surveillance Program WCAP 7723-Surry 1 (July, 1972), WCAP 8085-Surry 2 (June, 1973)
- (10) Battelle Columbus Laboratories Research Reports for Surry Pressure Vessel Irradiation Capsule Program.
 - (a) Surry 1 examination and analysis of capsule T (June, 1975)
 - (b) Surry 2 examination and analysis of capsule X (Sept., 1975)
- (11) ASTM: E185-73, E208, & E23
- (12) Surry T.S. Change 27 (Proposed Change 35)
- (13) Vepco letter to Mr. Robert W. Reid, NRC Chief Operating Reactors Branch 4, of February 15, 1978, Serial No. 081

UPPER PRESSURIZATION LIMITS
FOR HEATUP AND COOLDOWN
SURRY UNITS NO. 1 AND 2

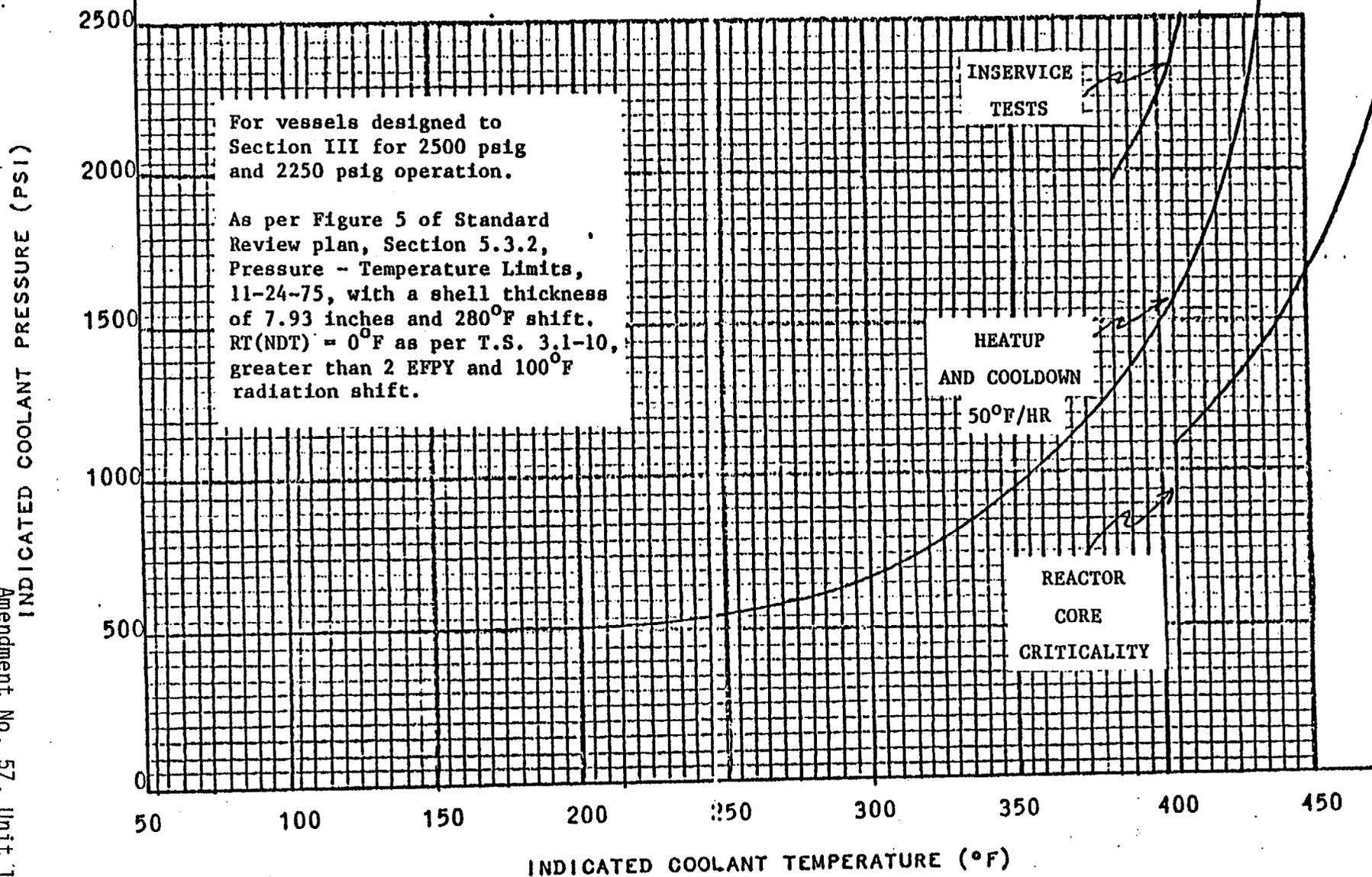
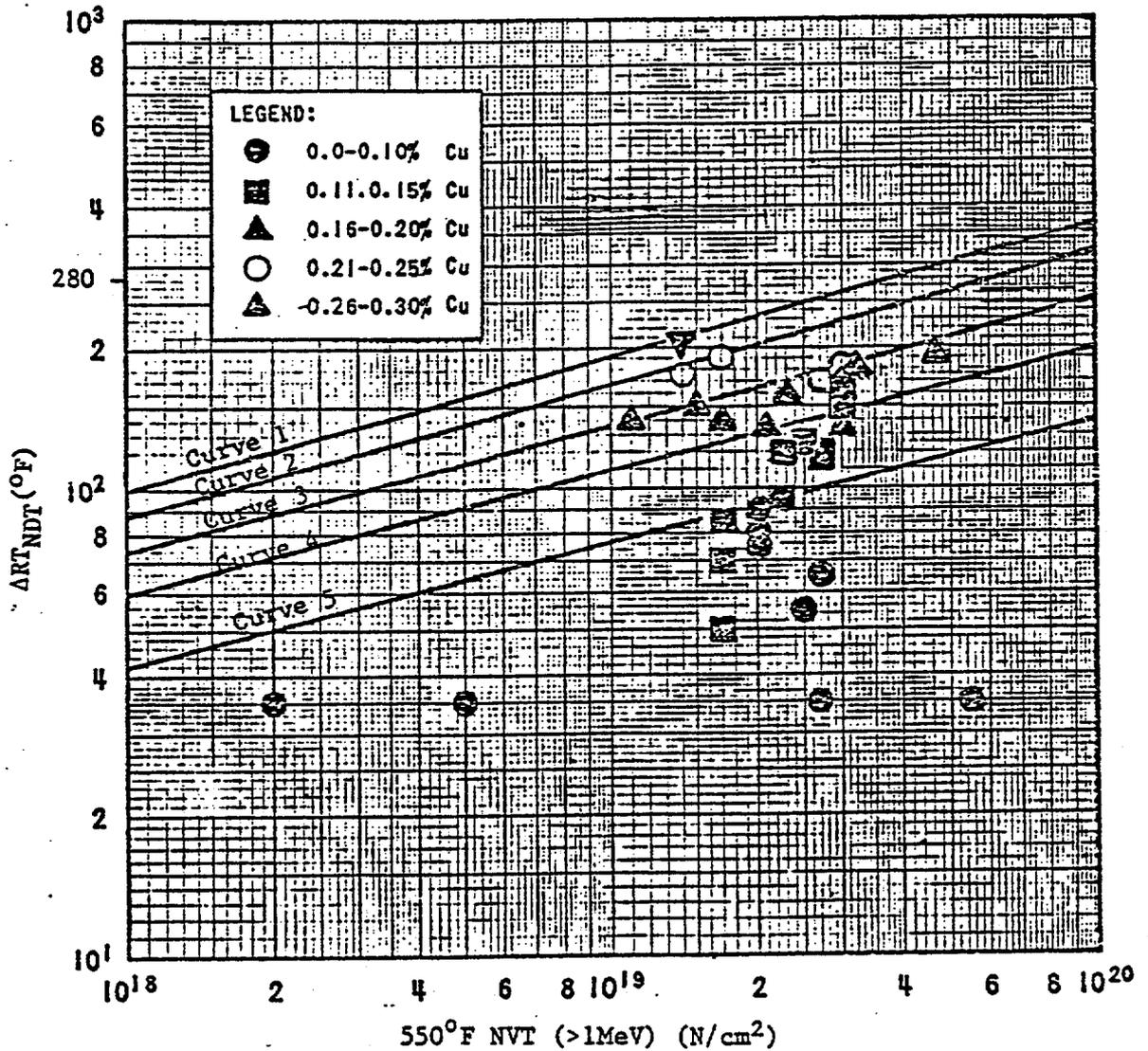


FIGURE VALID UP TO 11 EFPY.

Amendment No. 57, Unit 1
 Amendment No. 56, Unit 2

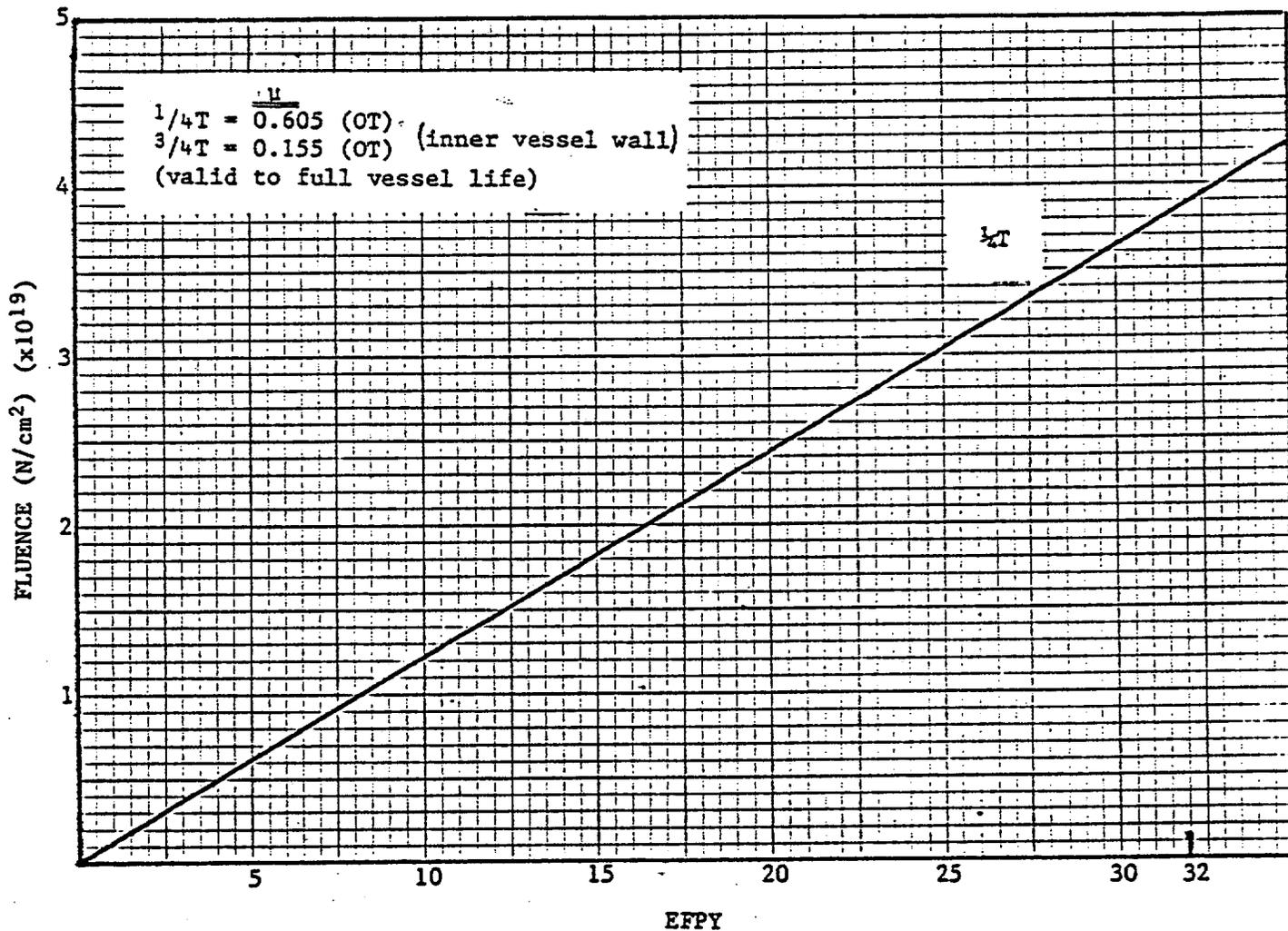
TS FIG 3.1-1



Curve 1 - 0.30% Cu base, 0.25% Cu weld
 Curve 2 - 0.25% Cu base, 0.20% Cu weld
 Curve 3 - 0.20% Cu base, 0.15% Cu weld
 Curve 4 - 0.15% Cu base, 0.10% Cu weld
 Curve 5 - 0.10% Cu base, 0.05% Cu weld

Figure 3.1-2. Radiation Induced Increase In Transition Temperature

SURRY 1&2



NOTE: Slope of line (ϕ) above was determined from T.S. Figure 3.1-2 with a 0.25% Cu weld for 32 EFPY's and a 2800°F shift as determined for T.S. Figure 3.1-1. It was calculated as follows:

$$\phi = \frac{3.85 \times 10^{19} \text{N/cm}^2}{32 \text{ EFPY}} = 1.203 \times 10^{18} \text{N}/(\text{cm}^2\text{-EFPY})$$

Figure valid up to 11 EFPY

TABLE 4.2-1

SECTION F. VALVE PRESSURE BOUNDARY (Continued)

<u>Item No.</u>	<u>Category</u>	<u>Required Examination Areas</u>	<u>Required Examination Methods</u>	<u>Extent of Examination Planned During First 5-Year Interval</u>	<u>Tentative Inspection During 10-Year Interval</u>	<u>Remarks</u>
6.7	(Continued)					The support settings of constant and variable spring-type hangers, snubbers and shock absorbers would be inspected to verify proper distribution of design loads among the associated support components.

SECTION G. MISCELLANEOUS INSPECTIONS

7.1	Materials Irradiation Surveillance	Tensile and Charpy V notch (wedge open loading) and Dosimetry	Capsule 1 shall be removed and examined at the first region replacement. Capsule 2 shall be examined after 5 years.	Capsules shall be removed and examined after 10 years	Capsule 4 shall be removed and examined after 20 years. Capsules 5-8 are extra capsules for complementary or duplicate testing.
7.2	Low Head SIS Piping Located in Valve Pit	Visual	(See Remarks)	Not Applicable	This pipe shall be visually inspected at each refueling shutdown.
7.3	Low Pressure Turbine Rotor	Visual and magnetic particle or dye penetrant	100% of blades	Not Applicable	



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 57 AND 56 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 December 15, 1978, Virginia Electric and Power Company (VEPCO) submitted an application to amend the Technical Specifications appended to Facility Operating Licenses DPR-32 and DPR-37 for Surry Units 1 and 2. The requested change would extend the acceptable operating period of the present operating limits from 3.8 EFPY to 11 EFPY.

Discussion

10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The pressure-temperature limits depend upon the metallurgical properties of the reactor vessel materials. The properties of materials in the vessel beltline region vary over the lifetime of the vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the vessel material nil-ductility temperature (RT_{NDT}) to increase with time. The pressure-temperature operating limits must be modified periodically to account for this radiation-induced increase in RT_{NDT} by increasing the temperature required for a given pressure. The operating limits for a particular operating period are based on the material properties at the end of the operating period. By periodically revising the pressure-temperature limits to account for radiation damage, the stresses and stress intensities in the reactor vessel are maintained within acceptable limits.

The magnitude of the shift in RT_{NDT} is proportional to the neutron fluence to which the materials are subjected. The shift in RT_{NDT} can be predicted from Regulatory Guide 1.99. To check the validity of the predicted shift in RT_{NDT} , a reactor vessel material surveillance program is required. Surveillance specimens are periodically removed from the vessel and tested. The results of these tests are compared to the predicted shifts in RT_{NDT} , and the pressure-temperature operating limits are revised accordingly.

Evaluation

The operating limits currently in the Technical Specifications are acceptable for operation to 3.8 EFPY. The same pressure-temperature operating limits are used for both Surry reactor vessels. The limits are based on the weld metal in the Surry 1 vessel which is the most limiting material. To date one material surveillance capsule has been removed from each vessel and tested. The test results on these capsules clearly show that the Unit 1 surveillance weld metal is the limiting material. However, this weld material is not identical to the weld metal used to fabricate the reactor vessel; i.e., these welds were not made from the same heats of weld wire and flux. Therefore, VEPCO recalculated the amount of radiation damage using damage prediction curves supplied by Westinghouse. This calculation showed that the present limit curves are acceptable through 32 EFPY.

We have reviewed the proposed change to the operating limits and have performed independent calculations to verify compliance with Appendix G, 10 CFR 50. We agree that the surveillance weld metal in Unit 1 is not identical to that used to fabricate the reactor vessel. However, since it was made by the same type of weld wire and flux, we consider it to be representative of the vessel weld. The NRC staff is currently studying the effects of different heats of weld wire-flux combinations on radiation damage. Although this study is not complete, it indicates that the surveillance data is applicable for the determination of radiation damage on the Surry 1 reactor vessel.

The results of the tests on the two surveillance capsules were reviewed again. It is concluded that the limiting material is the Unit 1 weld metal. At a fluence of 2.5×10^{18} n/cm² this weld metal shows an increase in RT_{NDT} of 165°F. This increase in RT_{NDT} is approximately on the upper limit line of Regulatory Guide 1.99, Revision 1 and is consistent with data generated from other surveillance programs on similar weld metal samples. Using the above shift in RT_{NDT} and the procedures in Regulatory Guide 1.99, Revision 1, the staff finds that the operating limits currently in the Technical Specifications are acceptable for operation through 11 EFPY. We have discussed this with the licensee and he agrees with this change. These operating limits are acceptable for both Surry 1 and 2 since the limiting material in the Surry 2 reactor vessel has a greater resistance to radiation damage than the limiting material in the Unit 1 vessel. For this operating period these operating limits are in accordance with Appendix G, 10 CFR 50. Conformance with Appendix G to 10 CFR Part 50 in establishing safe operating limitations will ensure adequate safety margins during operation, testing,

maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requirements to NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

VEPCO also requested a change to Technical Specification 4.2, Section G regarding materials irradiation testing. The proposed change would delete the requirement for conducting Charpy tests on irradiated surveillance specimens. This change is not in accordance with Appendix H, 10 CFR 50 and is therefore not acceptable. This specification should remain unchanged. We have discussed this with the licensee and he agrees with not making this change.

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 §51.5(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 does 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: March 4, 1980

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. 57 and 56 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 facility) located in Surry County, Virginia. The amendments are effective as the date of issuance.

The amendments revise the Technical Specifications to specify new pressure-temperature limits for heatup and cooldown operations.

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. Prior public notice of these amendments was not required since they do not involve a significant hazards consideration.

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 December 15, 1978; (2) Amendment Nos. 57 and 56 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 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 4th day of March, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION



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