October 24, 1990

Docket Nos. 50-280
and 50-281

DISTRIBUTION
See attached sheet

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

Dear Mr. Stewart:

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: HEATUP AND COOLDOWN CURVES (TAC NOS. 67386, 67387, 71555 AND 71556)

The Commission has issued the enclosed Amendment No. 147 to Facility Operating License No. DPR-32 and Amendment No. 143 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 application transmitted by letter dated January 29, 1988, as supplemented February 20, 1989.

These amendments revise the heatup and cooldown curves to be effective from 11 effective full power years (EFPY) of operation to 15 EFPY.

A copy of the Safety Evaluation and the Notice of Issuance are enclosed.

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 147 to DPR-32
2. Amendment No. 143 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

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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. 147
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 January 29, 1988, as supplemented February 20, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 147 , 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: October 24, 1990

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. 143
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 January 29, 1988, as supplemented February 20, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(B) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 143 , 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: October 24, 1990

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 147 FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 143 FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 3.1-6	TS 3.1-6
TS 3.1-7	TS 3.1-7
TS 3.1-8	TS 3.1-8
TS 3.1-9	TS 3.1-9
TS 3.1-10	TS 3.1-10
TS 3.1-11	TS 3.1-11
TS 3.1-12	TS 3.1-12
TS 3.1-23	TS 3.1-23
TS 3.1-25	TS 3.1-25
Figure 3.1-1	Figure 3.1-1
Figure 3.1-2	Figure 3.1-2
Figure 3.1-3	Figure 3.1-3
--	Figure 3.1-4
--	Table 3.1-1
--	Table 3.1-2
TS 4.2-4	TS 4.2-4

B. HEATUP AND COOLDOWN

Specification

1. Unit 1 and Unit 2 reactor coolant temperature and pressure and the system heatup and cooldown (with the exception of the pressurizer) shall be limited in accordance with TS Figures 3.1-1 and 3.1-2.

   Heatup:

   Figure 3.1-1 may be used for heatup rates of up to 40°F/hr.

   Cooldown:

   Allowable combinations of pressure and temperature for specific cooldown rates are below and to the right of the limit lines as shown in TS Figure 3.1-2. This rate shall not exceed 100°F/hr. Cooldown rates between those shown can be obtained by interpolation between the curves on Figure 3.1-2.

   Core Operation:

   During operation where the reactor core is in a critical condition (except for low level physics tests), vessel metal and fluid temperature shall be maintained above the reactor core criticality limits specified in 10 CFR 50 Appendix G. The reactor shall not be made critical when the reactor coolant temperature is below 522°F as specified in T.S. 3.1.E.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

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.

Basis

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.1-1 and 3.1-2.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b) Figures 3.1-1 and 3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods provided below.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) 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.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI according to the leak test limit line shown in Figure 3.1-1.

6) The reactor shall not be made critical when the reactor coolant temperature is below 522°F in accordance with Technical Specification 3.1.E.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-82, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of (15) Effective Full Power Years of service life. The (15) EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Tables 3.1-1 (Unit 1) and 3.1-2 (Unit 2). Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence, can be predicted using Figures 3.1-3 (Unit 1) and 3.1-4 (Unit 2) and the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of (15) EFPY (as well as adjustments for possible errors in the pressure and temperature sensing instruments).

Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 15 EFPY prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperture, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 40°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 15 EFPY. The adjusted reference temperature was calculated using results from a capsule removed after the eighth cycle. The results are documented in WCAP 11415 and WCAP 11492-1 for Unit 1 and WCAP 11499 and WCAP 11505 for Unit 2.

References

(1) FSAR 4.2

(2) FSAR 9.2

G. Reactor Coolant System Overpressure Mitigation

Specification

1. The Reactor Coolant system overpressure mitigating system shall be operable as described below.

   a. Whenever the reactor coolant average temperature is greater than 350°F, a bubble shall exist in the pressurizer with the necessary sprays and heaters operable.

   b. Whenever the reactor coolant average temperature is ≤ 350°F and the reactor vessel head is bolted:

      (1) A maximum of one charging pump operable

      (2) Two charging pumps shall be demonstrated inoperable at least once per 12 hours by verifying the motor circuit breakers have been removed from their power supply or the benchboard control switch is in the "PULL-TO-LOCK" position.

      (3) Two operable Power Operated Relief Valves (PORV's) with a lift setting of ≤ 385 psig, or

      (4) A bubble in the pressurizer with a maximum pressurizer narrow range level of 33%. After a period of 72 hours, two PORV's must also be operable, or

      (5) The Reactor Coolant system vented through one opened PORV, or an equivalent size opening.

2. The requirements of Specification 3.1.G.1.b may be modified as follows:

   a. One PORV may be inoperable for a period not to exceed 7 days. If the inoperable PORV is not restored to operable status within 7 days, then depressurize the RCS and open one PORV within the next 8 hours.

capability to protect the Reactor Vessel from overpressurization when the transient is limited to either (1) the start of an idle Reactor Coolant Pump with the secondary water temperature of a steam generator ≤ 50°F above the RCS cold leg temperature or (2) the start of a charging pump and its injection into a water solid RCS.

The limitation for a maximum of one charging pump allowed operable and the surveillance required to verify that two charging pumps are inoperable below 350°F provide assurance that a mass addition pressure transient can be relieved by the operation of a single PORV, or equivalent.

A maximum pressurizer narrow range level of 33% has been selected to provide sufficient time, approximately 10 minutes, for operator response in case of a malfunction resulting in maximum charging flow from one charging pump (530 gpm). Operator action would be initiated by at least two alarms that would occur between the normal operating level and the maximum allowable level (33%). When both PORVs are inoperable and it is impossible to manually open at least one PORV, additional administrative controls shall be implemented to prevent a pressure transient that would exceed the limits of Appendix G to 10 CFR Part 50.

The requirements of this specification are only applicable when the Reactor Vessel head is bolted. When the Reactor Vessel head is unbolted, a RCS pressure of < 100 psig will lift the head, thereby creating a relieving capability equivalent to at least one PORV.

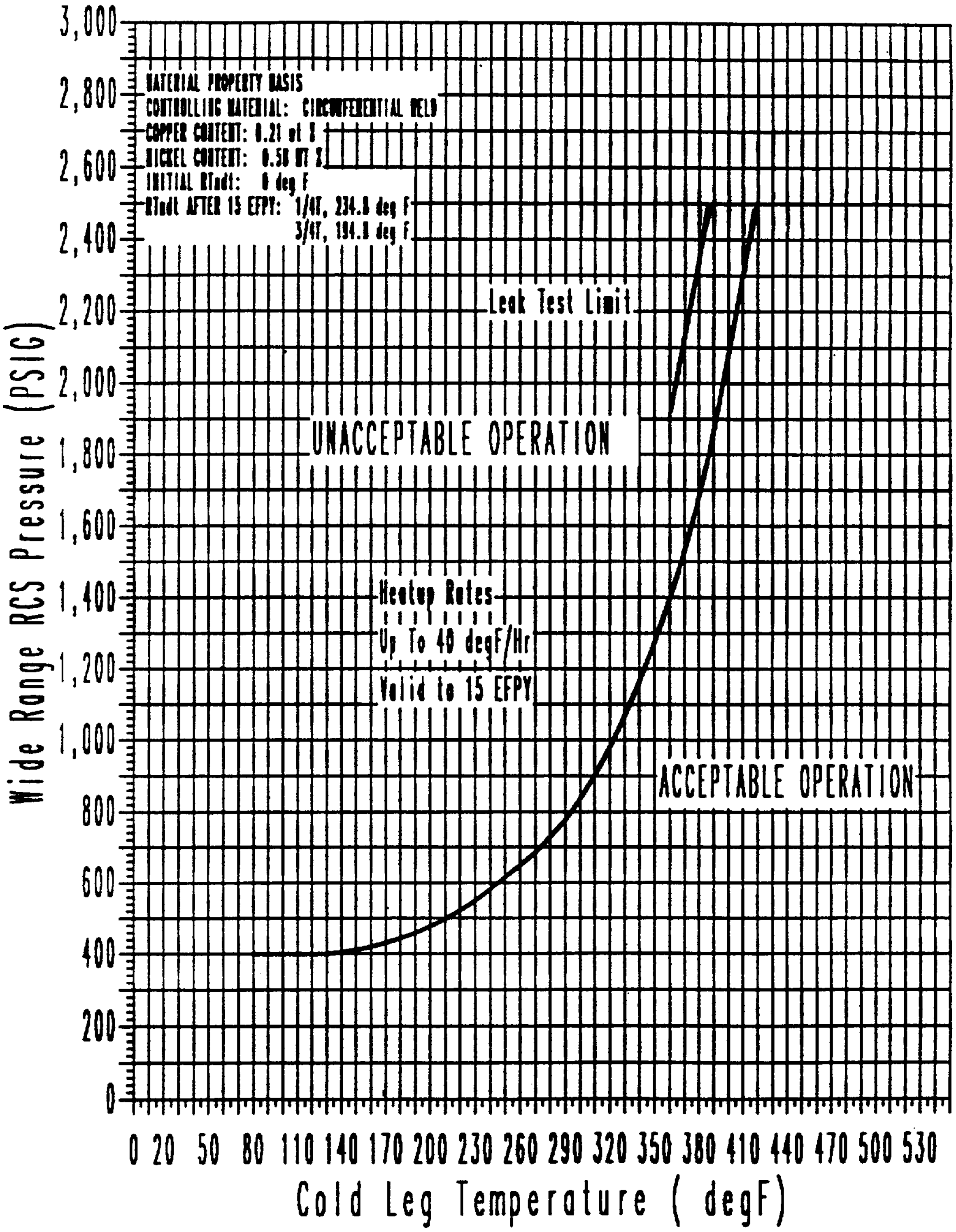

Figure 3.1-1 RCS Heatup Limitations

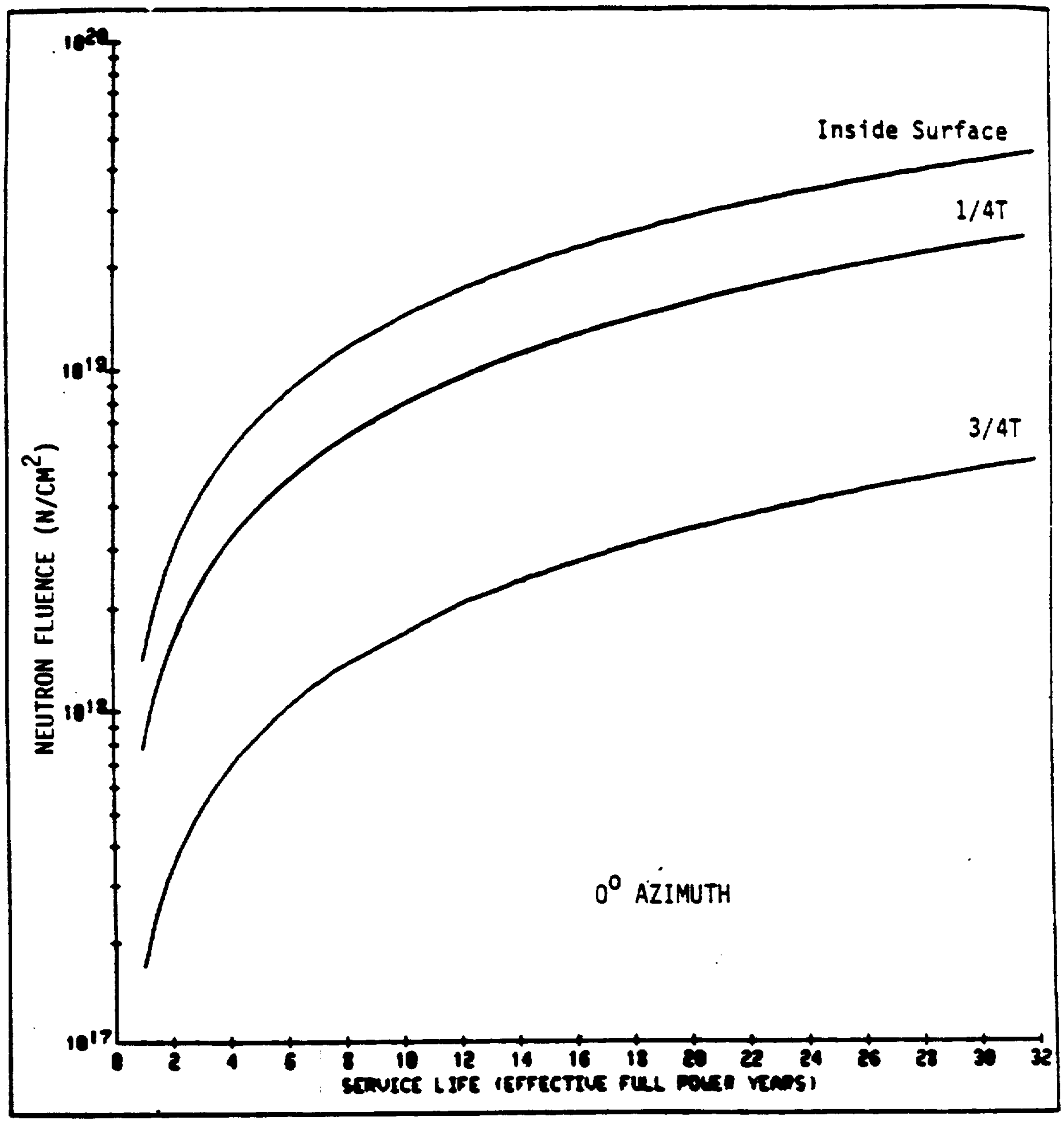

Figure 3.1-3 **Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life (EFPY) for Surry Unit 1**

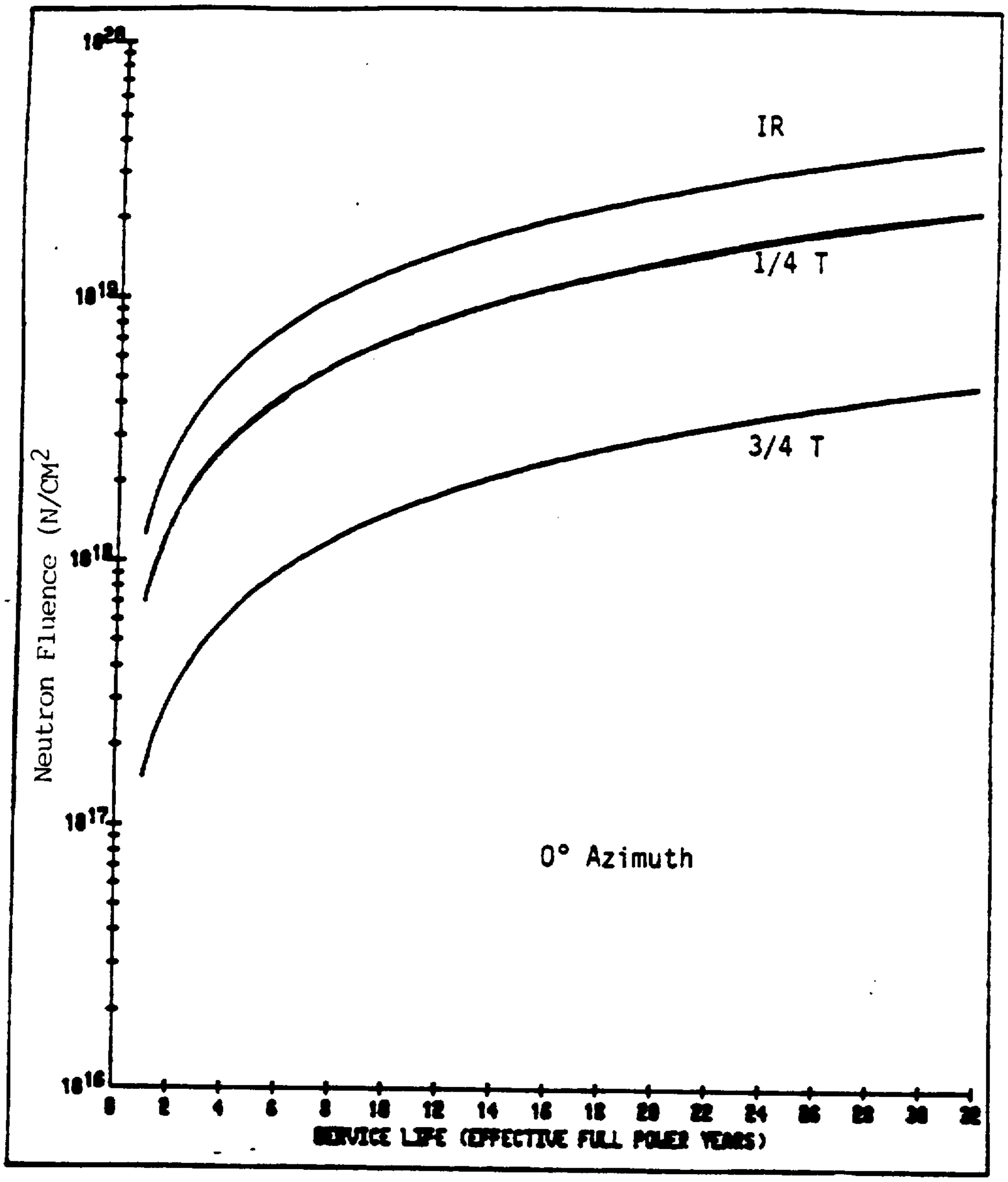

Figure 3.1-4 **Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life (EFPY) for Surry Unit 2**

TABLE 3.1-1

UNIT 1 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)[d]

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Closure head dome	C4315-2	A53313 Cl. I	.14	.59	.011	0	0	75
Head flange	FV-1894	A508 Cl. 2	.13	.64	.010	10[a]	10	125
Vessel flange	FV-1870	A508 Cl. 2	.10	.65	.009	10[a]	10	74
Inlet nozzle	9-5078	A508 Cl. 2	-	.87	.007	60[a]	60	64
Inlet nozzle	9-4819	A508 Cl. 2	-	.84	.008	60[a]	60	68
Inlet nozzle	9-4787	A508 Cl. 2	-	.85	.007	60[a]	60	64
Outlet nozzle	9-4762	A508 Cl. 2	-	.83	.007	60[a]	60	85
Outlet nozzle	9-4788	A508 Cl. 2	-	.84	.007	60[a]	60	72
Outlet nozzle	9-4825	A508 Cl. 2	-	.85	.008	60[a]	60	68
Upper shell	122V109	A508 Cl. 2	.07	.74	.010	40	40	83
Intermediate shell	C4326-1	A533B Cl. I	.11	.55	.008	10	10	115[c]
Intermediate shell	C4326-2	A533B Cl. 1	.11	.55	.008	0	0	93
Lower shell	C4415-1	A533B Cl. I	.11	.50	.014	20	20	103[c]
Lower shell	C4415-2	A533B Cl. 1	.11	.50	.014	0	0	80
Bottom head ring	123T338	A508 Cl. 2	-	.69	.020	50	50	86
Bottom dome	C4315-3	A533B Cl. 1	.14	.59	.011	0	0	85
Inter. & lower shell vertical weld seam L1, L3, & L4	8T1554 & Linde 80 flux		.18	.63	.014	0[a]	0	N/A

TABLE 3.1-1 (Continued)

UNIT 1 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)[d]

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Lower shell vertical weld seam, L2	299L44 & Linde 80 flux		.35	.67	.014	0[a]	0	70[a]
Inter. to lower shell girth seam	72445 & Linde 80 flux		.21	.58	.016	0[a]	-6	N/A

NOTES:

(a) Estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(b) Normal to major working direction - estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(c) Actual values

(d) Reactor Vessel Fabricator Certified Test Reports

TABLE 3.1-2

UNIT 2 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Closure head dome	C4361-2	A533B Cl. 1	.15	.52	.010	-20	7	81
Head flange	ZV-3475	A508 Cl. 2	.11	.60	.011	<10[a]	<10	129
Vessel flange	ZV-3476	A508 Cl. 2	.10	.64	.013	-65[a]	-65	129
Inlet nozzle	9-4815	A508 Cl. 2	-	.87	.008	60[a]	60	66
Inlet nozzle	9-5104	A508 Cl. 2	-	.84	.006	60[a]	60	73
Inlet nozzle	9-5205	A508 Cl. 2	-	.86	.007	60[a]	60	66
Outlet nozzle	9-4825	A508 Cl. 2	-	.85	.009	60[a]	60	74
Outlet nozzle	9-5086	A508 Cl. 2	-	.86	.009	60[a]	60	79
Outlet nozzle	9-5086	A508 Cl. 2	-	.87	.011	60[a]	60	73
Upper shell	123V303	A508 Cl. 2	.09	.73	.010	30	30	103
Intermediate shell	C4208-2	A533B Cl. 1	.15	.55	.008	-30	-30	94
Intermediate shell	C4339-1	A533B Cl. 1	.11	.54	.012	-10	11	94[c]
Lower shell	C4331-2	A533B Cl. 1	.12	.60	.009	-10	10	84
Lower shell	C4339-2	A533B Cl. 1	.11	.54	.012	-20	10	83
Bottom head ring	123T321	A508 Cl. 2	-	.71	.010	10	10	101
Bottom dome	C4361-3	A533B Cl. 1	.15	.52	.010	-20	-15	80
Intermediate shell vertical weld seam L3, & L4	72445 & Linde 80 flux Lot 8579		.21	.59	.016	-	- 6	70
Lower shell vertical welds								
Seam L1 (ID 63%) Seam L2 (100%)	8T1762 & Linde 80 flux 8579		.29	.55	.015	-	- 6	70
Seam L1 (37%)	8T1762 & Linde 80 flux 8632		.29	.55	.010	-	- 6	70

TABLE 3.1-2 (Continued)

UNIT 2 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Inter. to lower shell girth seam	0227 Grau Lo Flux	LW320	.19	.56	.017	0[a]	0	90

NOTES:

(a) Estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(b) Normal to major working direction - estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(c) Actual value based on surveillance tests normal to the major working direction

TABLE 4.2-1

SECTION A. MISCELLANEOUS INSPECTIONS

Item No.	Required Examination Area	Required Examination Methods	Tentative Inspection During 10-Year Interval	Remarks
1.1	DELETED			
1.2	Low Head SIS piping located in valve pit	Visual	Non-applicable	This pipe shall be visually inspected at each refueling shutdown.

Note 1: 1 year corresponds to 1 year effective full power operation.

Note 2: The results obtained from these examinations shall be used to update Figure 3.1-1 as required.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.147 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO.143 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

1.0 INTRODUCTION

By letter dated January 29, 1988, as supplemented February 20, 1989, Virginia Electric and Power Company (VEPCO, the licensee) submitted proposed Technical Specification revisions for the Surry Power Station, Units 1 and 2. The proposed revisions affect the reactor coolant system pressure/temperature (P/T) limits and the low temperature overpressure protection (LTOP) system setpoints. The revisions define operating limits through 15 effective full-power years (EFPY) of plant operation. The current operating limits are for 11 EFPY. This Safety Evaluation documents acceptance of the proposed changes.

2.0 EVALUATION

The proposed Technical Specifications revisions affect the P/T limits shown in Figure 3.1-1. This curve is being split into two curves, Figures 3.1-1 and 3.1-2, representing P/T limits for heatup and cooldown, respectively. In addition, the maximum heatup rate is limited to 40°F per hour. The proposed P/T limits on the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 15 EFPY, because the limits conform to the requirements of 10 CFR Part 50, Appendices G and H. VEPCO has also satisfied Generic Letter 88-11, because the method in Regulatory Guide (RG) 1.99, Rev. 2 was used to calculate Adjusted Reference Temperature (ART). Hence, the proposed P/T limits will be incorporated into the Surry Power Station Units 1 and 2 Technical Specifications. The LTOP system setpoint is changed based on the revised P/T limits.

2.1 P/T Curve Revision

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11. Appendix G requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2 to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H to 10 CFR Part 50 requires licensees to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standard which, in turn, requires that the capsules be installed in the vessel before startup and that they contain test specimens that are made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

The staff has evaluated the effect of neutron irradiation embrittlement on each beltline material in the Surry Units 1 and 2 reactor vessels. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff determined that the material with the highest ART at 15 EFPY for both units was the circumferential weld (W05) between the intermediate and lower shells in Unit 1 with a 0.21% copper (Cu), 0.58% nickel (Ni), and an initial RT_{ndt} of -6°F.

The licensee has removed three surveillance capsules each from Surry Unit 1 and Surry Unit 2. Results from the three surveillance capsules from Unit 1 were published in Battelle-Columbus Laboratory reports for Capsules T and W and in Westinghouse Report WCAP-11415 for Capsule V. Results from the three surveillance capsules from Unit 2 were published in Battelle-Columbus Laboratory Reports for Capsules X and W and in Westinghouse Report WCAP-11499 for Capsule V. All surveillance capsules contained Charpy impact specimens and tensile specimens which were made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, girth weld W05 between the intermediate and lower shells in Unit 1, the staff calculated the ART at 15 EFPY at 1/4T (T = reactor vessel beltline thickness) to be 228.9°F using Section 1 of RG 1.99, Rev. 2.

The licensee used the method in Section 1 of RG 1.99, Rev. 2, to calculate an ART of 234.8°F for the limiting material, girth weld W05. The licensee's ART of 234.8°F is more conservative than the staff's ART of 228.9°F; therefore, it is acceptable. By substituting the ART of 234.8°F into equations in SRP Section 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

2.2 PORV Setpoint Revision

At Surry, LTOP is provided by Technical Specification controls on charging pump operability, and by reactor coolant system (RCS) vent paths through operable power operated relief valves (PORVs). These controls ensure anticipated mass or energy addition transients cannot result in excessive RCS pressurization in low temperature conditions. The proposed revision reduces the PORV setpoint from 435 to 385 psig.

The most limiting mass addition transient was analyzed assuming an inadvertent actuation of a charging pump. The present Technical Specification 3.1.G.1.b allows only one charging pump to be operable when the RCS temperature is less than or equal to 350°F, which is the maximum RCS temperature for which LTOP is required. The analysis was performed to determine the pressure overshoot past the LTOP setpoint such that the Appendix G curves are not exceeded during the transient. Separate analyses were performed for each unit since the P/T limits are different for each unit. However, for ease of operation, the more restrictive (lower) limit was selected for the proposed Technical Specification.

The heat input transient was analyzed assuming a 50°F temperature difference between the steam generator and the RCS. A reactor coolant pump startup in one loop was assumed to maximize the heat transfer effect. As was the case for the mass addition transient, the pressure overshoot was calculated such that the Appendix G P/T curves for each unit were not exceeded.

The final setpoint of 385 psig was selected as the value which bounds both limiting transients. Considering the above factors, the staff concludes that the assumptions applied to the licensee's analyses are reasonably conservative and acceptable.

The licensee's analysis was performed using RETRAN 02/MOD02 to support the proposed Technical Specification changes. Both RETRAN 01/MOD03 and RETRAN 02/MOD02 have been generically approved by the NRC staff. Also, a VEPCO topical report on its plant-specific application of RETRAN 01/MOD03 has been reviewed and approved by the NRC staff. In addition, VEPCO submitted comparisons between RETRAN 01/MOD03 and RETRAN 02/MOD02 for a series of plant transients. This information demonstrated that the RETRAN 01/MOD03 and RETRAN 02/MOD02 code results are nearly identical for the VEPCO plant-specific models, except for the changes caused by the nonequilibrium pressurizer model in RETRAN 02/MOD02. However, the LTOP transient analysis deals with the RCS in water solid conditions, thus, it is not affected by the nonequilibrium pressurizer model. While VEPCO's application on the use of RETRAN 02/MOD02 has not been reviewed by the staff, the staff considers that reasonable assurance exists that the results of the licensee's analysis using RETRAN 02/MOD02 supports the proposed Technical Specification changes on LTOP.

In the modified Appendix G heatup limit curve, the cold leg temperature corresponding to the pressurizer safety valve setpoint of 2485 psig is 420°F. This point is used to bound all of the low temperature transient analyses. Below 420°F, the anticipated low temperature overpressurization transients may be adequately mitigated by the automatic action of the pressurizer PORVs or by allowing sufficient time for operator response. Based on the results of the most limiting LTOP transient, the licensee-proposed Technical Specification PORV setpoint is less than or equal to 385 psig when the RCS average temperature is less than or equal to 350°F.

The licensee-proposed PORV setpoint change in Technical Specification 3.1.G.1.b and the associated Bases section reflect the above-discussed LTOP alignment temperatures and the heatup and cooldown rates identified by the updated Figures 3.1-1 and 3.1-2 in Technical Specification 3.1.B. The staff finds that they are reasonably conservative and acceptable.

3.0 SUMMARY

VEPCO has submitted proposed Technical Specification changes for the reactor coolant system pressure/temperature operating limits. The proposed limits have been developed consistent with the requirements of 10 CFR Part 50 Appendices G and H, and incorporate the methodology described in RG 1.99, Rev. 2, as requested in Generic Letter 88-11. VEPCO also proposes to change the PORV setpoint for low temperature overpressure protection. This setpoint provides adequate protection for anticipated pressurization transients for low temperature operations. The proposed changes to P/T limits and PORV setpoints provide acceptable operating conditions up to 15 effective full power years. Therefore, we find the proposed changes to be acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on October 24, 1990 (55 FR 42919). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of these amendments will not have a significant effect on the quality of human environment.

5.0 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: October 24, 1990

Principal Contributors:
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UNITED STATES NUCLEAR REGULATORY COMMISSION

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

NOTICE OF ISSUANCE OF AMENDMENTS TO

FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 147 to Facility Operating License No. DRP-32 and Amendment No. 143 to Facility Operating License No. DRP-37, issued to the Virginia Electric and Power Company (the licensee), which revised the Technical Specifications for operation of the Surry Plant, Units 1 and 2 (the facilities), located in Surry County, Virginia. The amendments were effective as of the date of issuance, to be implemented within 30 days.

The amendments revised the Technical Specifications for the heatup and cooldown curves to be effective to 15 effective full power years of operation.

The application for 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.

Notice of Consideration of Issuance of Amendments and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on April 13, 1988 (53 FR 12210).

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact, which was published in the FEDERAL REGISTER on October 24, 1990 (55 FR 42919).

For further details with respect to the action, see (1) the application for amendments dated January 29, 1988, as supplemented February 20, 1989, (2) Amendment No. 147to License No. DPR-32, and Amendment No.143 to License No. DRP-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, 2120 L Street, N.W., Washington D.C., and at the Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Dated at Rockville, Maryland this 24th day of October , 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Bart C. Buckley

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

DATED: October 24, 1990

AMENDMENT NO. 147 TO FACILITY OPERATING LICENSE NO. DPR-32 - SURRY UNIT 1
AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-37 - SURRY UNIT 2

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