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 a[illegible]ance with 10 CFR Part 51 of the Commission's regulations and [illegible] 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 accordi g o 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 .5 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) FSAP 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.

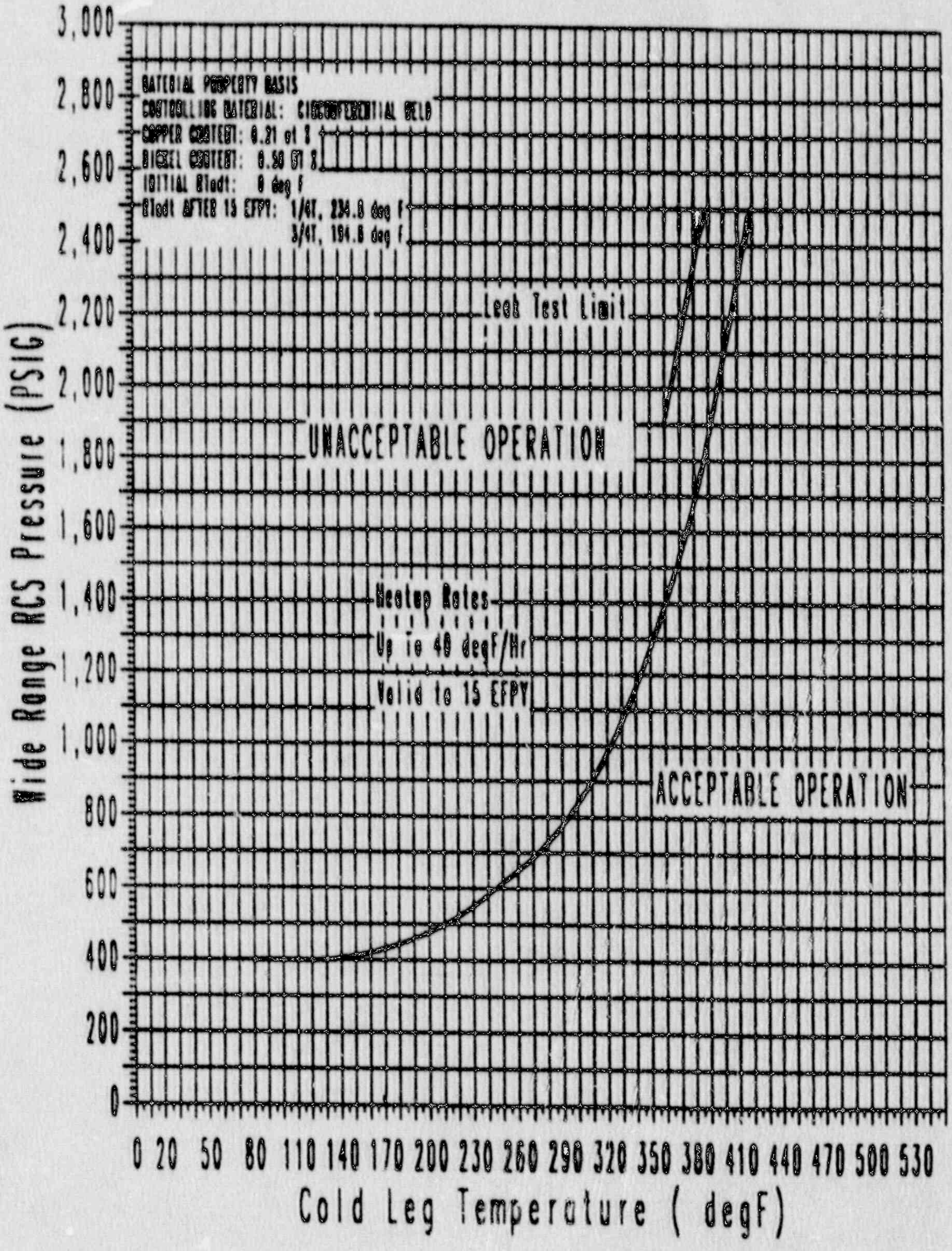

Figure 3.1-1 RCS Heatup Limitations

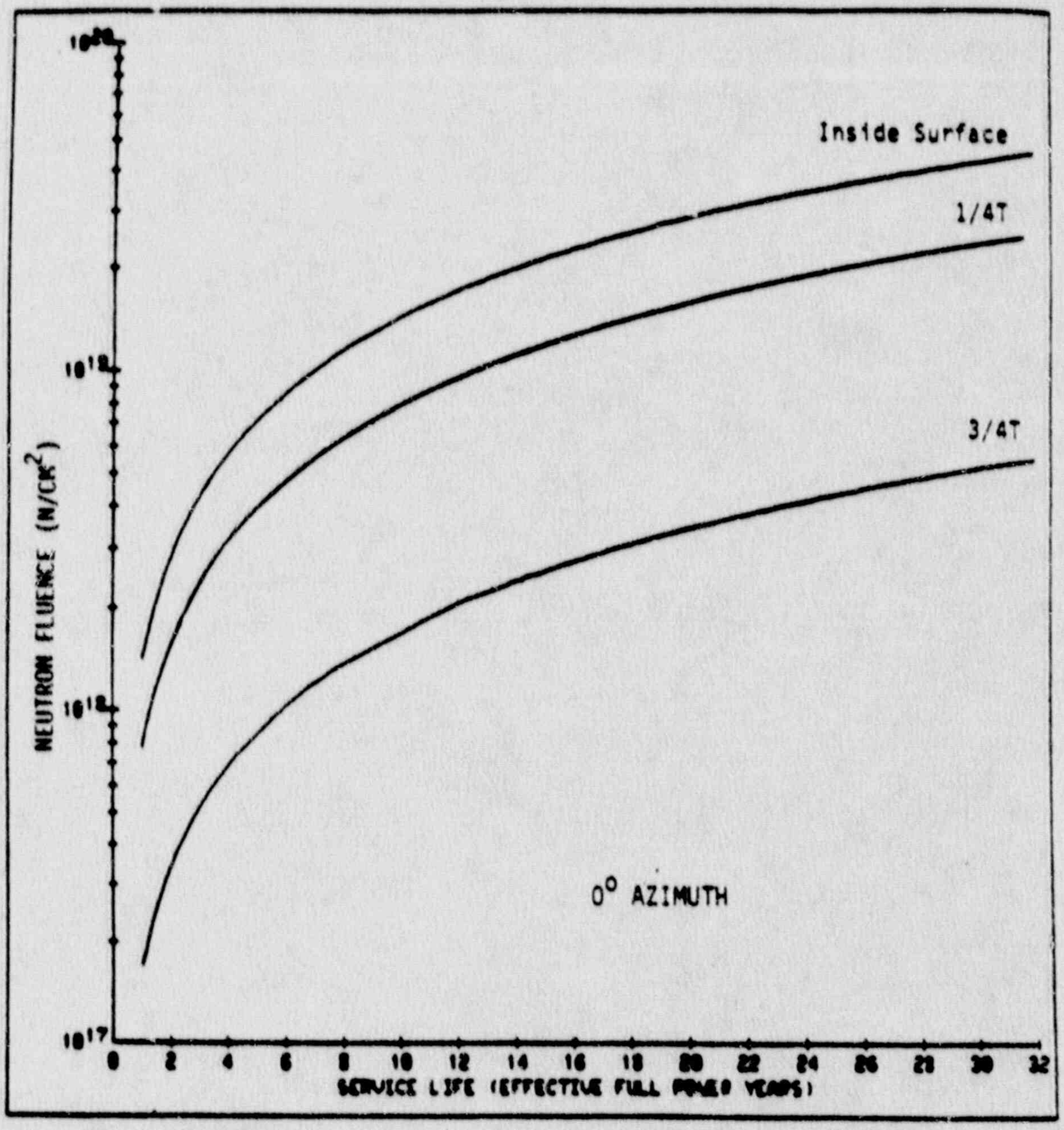

Figure 3.1-3 **Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life (EFPY) for Surry Unit 1**

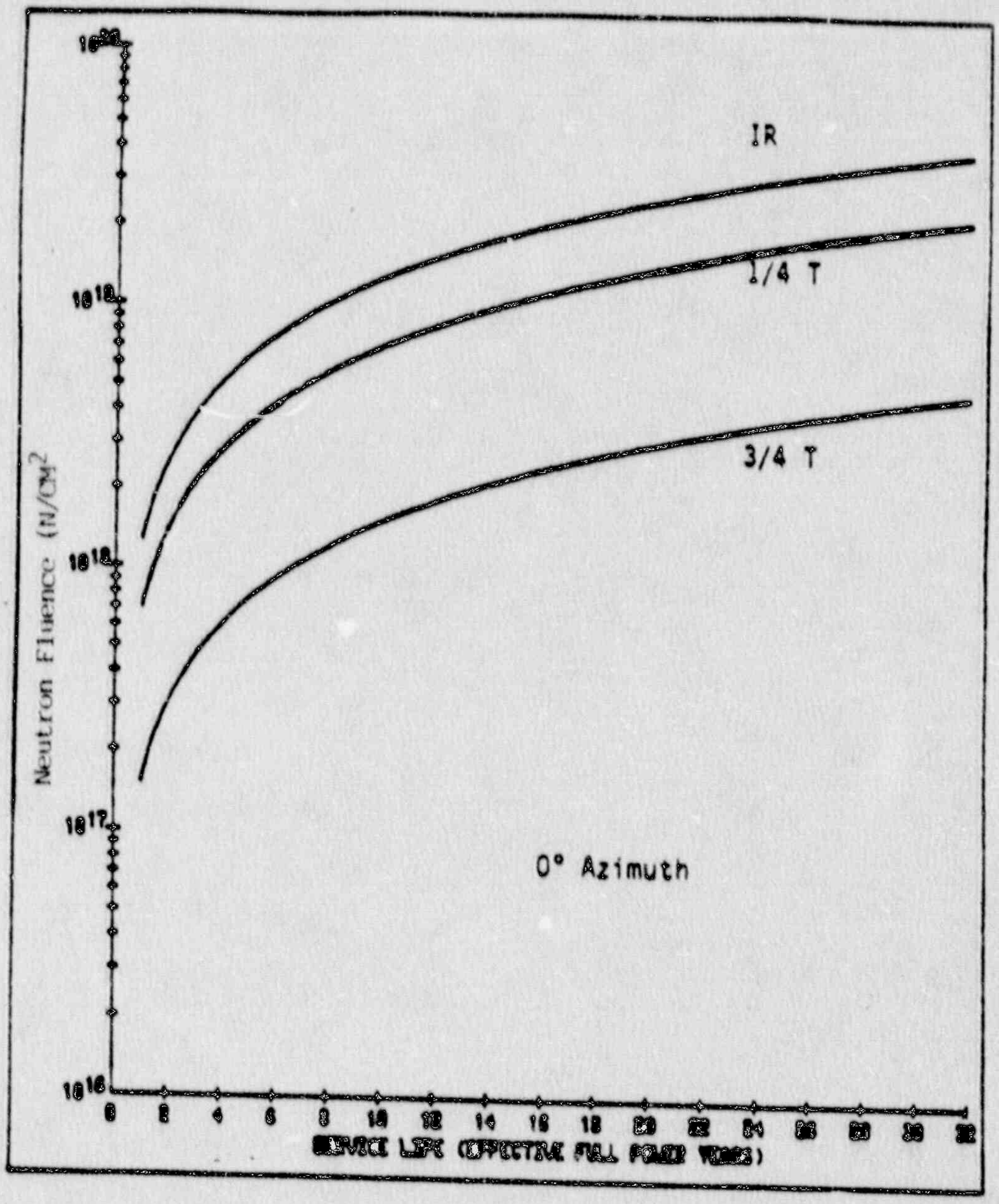

Figure 3.1-4 Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life (EFPY) for Surry Unit 2

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	A533B Cl. 1	.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. 1	.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. 1	.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	-[illegible]	-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%)	8T1762 & Linde 80		.29	.55	.015	-	- 6	70
Seam L2 (100%)	flux 8579							
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.