September 9, 1986

Docket No. 50-395

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Mr. D. A. Nauman
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
P.O. Box 764 (Mail Code 167)
Columbia, South Carolina 29218

Dear Mr. Nauman:

The Commission has issued the enclosed Amendment No. 53 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application dated February 7, 1986.

The amendment revises Technical Specification 3/4.4.9, "Pressure/Temperature Limits-Reactor Coolant System," and its bases, and changes the withdrawal order and schedule for the reactor vessel specimen capsules, as well as the plant heatup and cooldown rates. The amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/

Jon B. Hopkins, Project Manager
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 53 to NPF-12
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PAD#2	PM:PAD#2	D:PAD#2	OGC-BETH
DMiller	JHopkins:hc	LRubenstein	MYoung
8/25/86	8/25/86	8/26/86	8/28/86

w/ noted revision to SE
Dave. 9/9/86

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P PDR

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South Carolina Electric & Gas Company

Virgil C. Summer Nuclear Station

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

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. NPF-12

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

   A. The application for amendment by South Carolina Electric & Gas Company and South Carolina Public Service Authority (the licensees) dated February 7, 1986, 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 2.C.(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 53 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Leon B Engle for

Lester S. Rubenstein, Director
PWR Project Directorate #2
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 9, 1986

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

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 areas of change. Corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
B3/4 4-6	B3/4 4-6
B3/4 4-7	B3/4 4-7
B3/4 4-8	B3/4 4-8
B3/4 4-10	B3/4 4-10
B3/4 4-10a	B3/4 4-10a
B3/4 4-11	B3/4 4-11
B3/4 4-12	B3/4 4-12
B3/4 4-13	B3/4 4-13
B3/4 4-14	B3/4 4-14
-----	B3/4 4-14a

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period, and

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE IDENTIFICATION	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME, EFPY
U	343°	3.7	1st Refueling
V	107°	3.1	3rd Refueling
X	287°	3.1	5th Refueling
W	110°	2.7	10th Refueling
Y	290°	2.7	17th Refueling
Z	340°	2.7	STANDBY

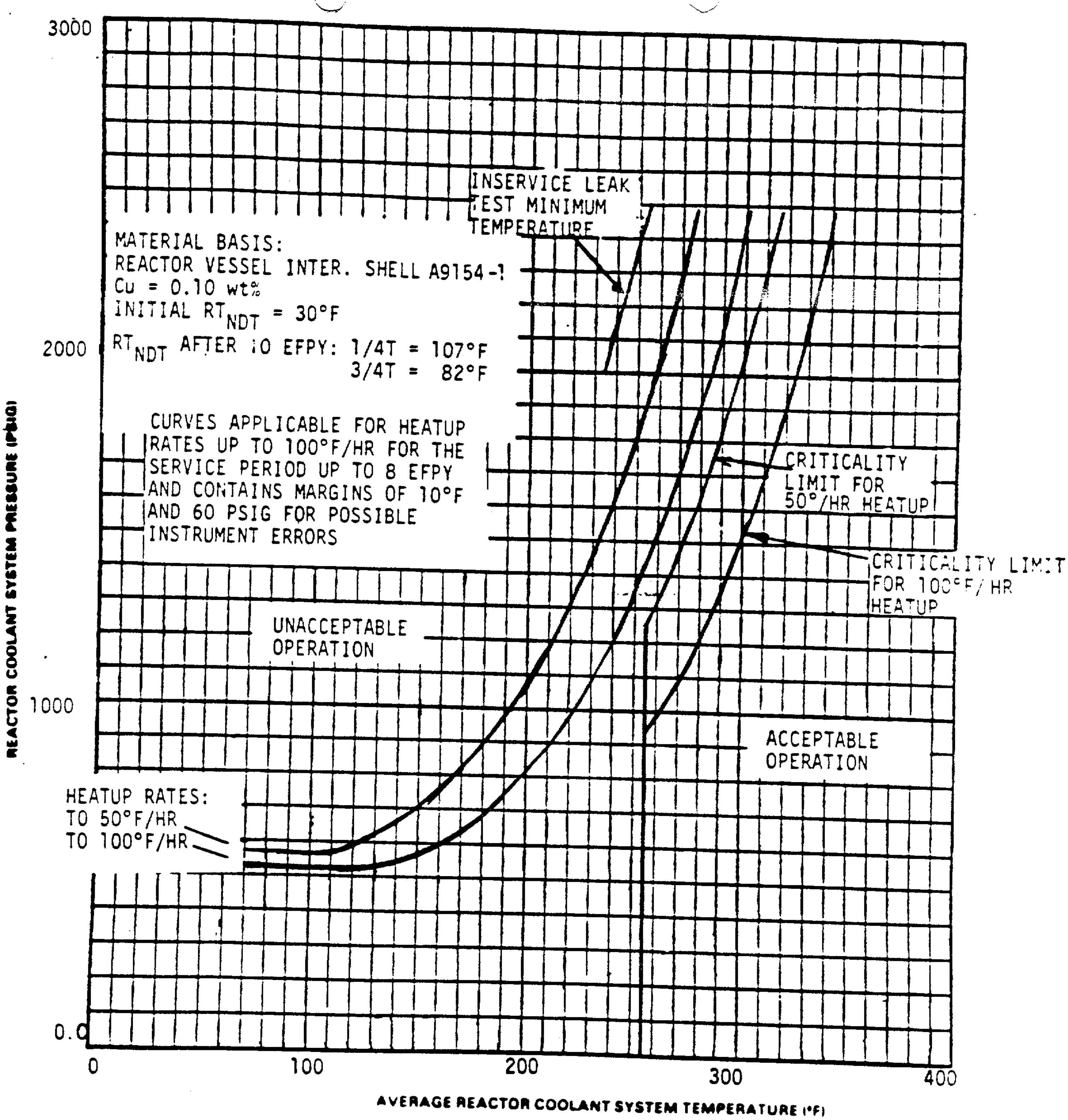


Figure 3.4-2 Reactor Coolant System Pressure - Temperature Limits Versus 100°F/Hour and 50°F/Hour Heatup Rate - Criticality Limit and Inservice Leak Test Limit

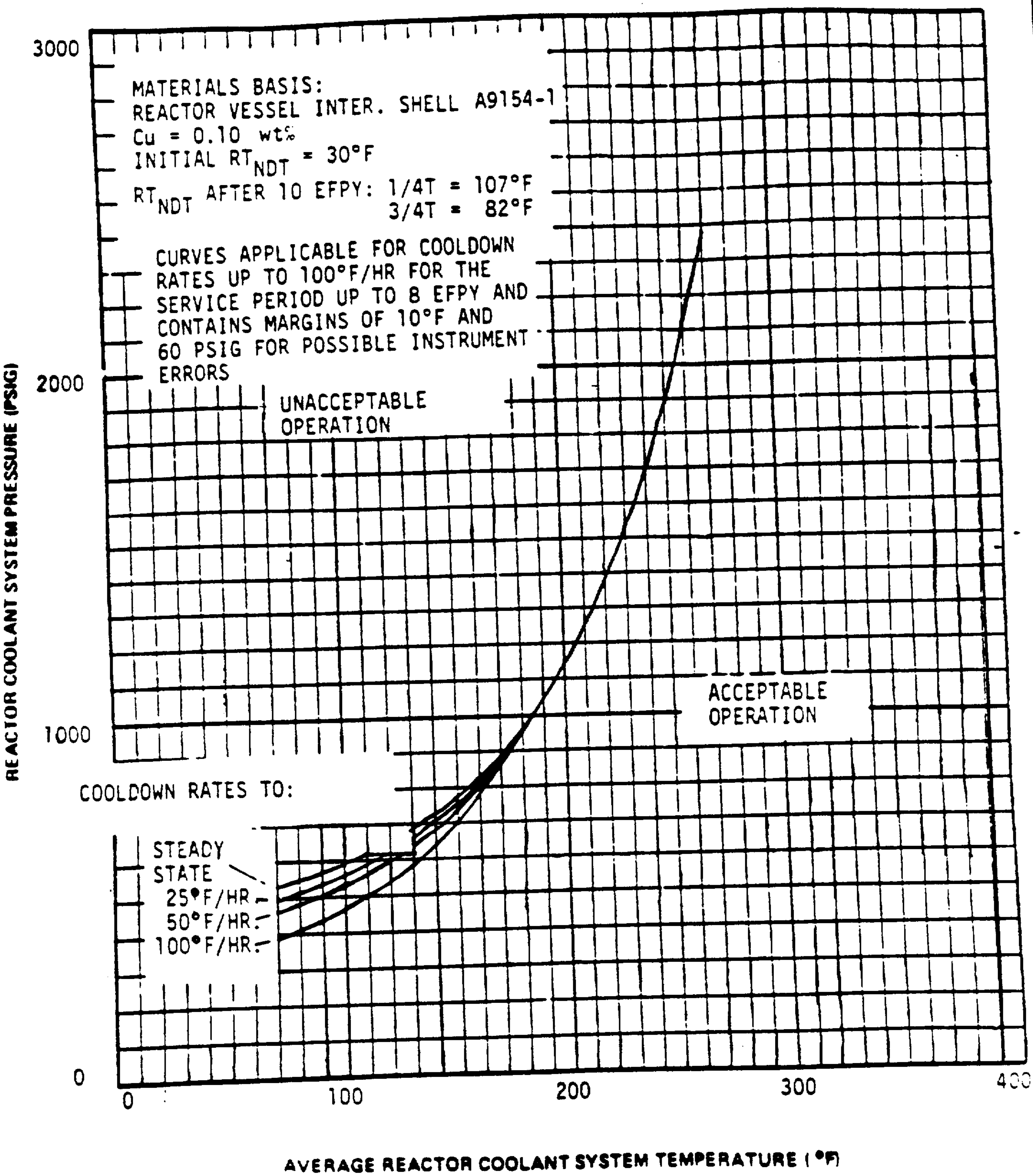


Figure 3.4-3
Reactor Coolant System Pressure - Temperature Limits
Versus Cooldown Rates

OPERATIONAL LEAKAGE (Continued)

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

SPECIFIC ACTIVITY (Continued)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcuries/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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 3.4-2 and 3.4-3.

    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.

PRESSURE/TEMPERATURE LIMITS (Continued)

b) Figures 3.4-2 and 3.4-3 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 625°F.

5) System in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature). The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper) present in reactor vessel steels. Design curves which show the effect of fluence and copper content on ΔRT_{NDT} for reactor vessel steels are shown in Figure B 3/4 4-2.

PRESSURE/TEMPERATURE LIMITS (Continued)

Given the copper content of the most limiting material, the radiation-induced ΔRT_{NDT} can be estimated from Figure B 3/4.4.2. Fast neutron fluence (E > 1 Mev) at the vessel inner surface, the 1/4 T (wall thickness), and 3/4 T (wall thickness) vessel locations are given as a function of full-power service life in Figure B 3/4.4.1. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to RT_{NDT}.

The preirradiation fracture-toughness properties of the V. C. Summer Unit 1 reactor vessel materials are presented in Table B 3/4.4-1. The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan.[1] The postirradiation fracture-toughness properties of the reactor vessel beltline material were obtained directly from the V. C. Summer Unit 1 Vessel Material Surveillance Program.

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 of the ASME Code.[2] The K_{IR} curve is given by the equation:

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

[1]"Fracture Toughness Requirements," Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

[2]ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendices, "Rules for Construction of Nuclear Vessels," Appendix G, "Protection Against Nonductile Failure," pp. 559-564, 1983 Edition, American Society of Mechanical Engineers, New York, 1983.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	Cu %	P %	NDTT (°F)	MIN. 50 FT-LB 35 MIL TEMP. (°F)	RT_{NDT} (°F)	AVG. UPPER SHELF (FT-LB)
Closure Head	A533-B-Class 1			-20	40	-20	106
Head Flange	SA508 Class 2			10	<60	10	129
Vessel Flange	SA508 Class 2			0	<60	0	172
Inlet Nozzle	SA508 Class 2			-20	<40	-20	130
Inlet Nozzle	SA508 Class 2			0	<60	0	114.5
Inlet Nozzle	SA508 Class 2			-20	<40	-20	135
Outlet Nozzle	SA508 Class 2			-10	<50	-10	146
Outlet Nozzle	SA508 Class 2			-10	<50	-10	165
Outlet Nozzle	SA508 Class 2			0	<50	0	150
Nozzle Shell	A533-B-Class 1	.13	.010	-20	78	18	100.5
Nozzle Shell	A533-B-Class 1	.12	.009	-30	86	26	91
Inter. Shell	A533-B-Class 1	.10	.009	-20	90	30	80.5
Inter. Shell	A533-B-Class 1	.09	.006	-20	40	-20	106.5
Lower Shell	A533-B-Class 1	.08	.005	-10	70	10	91.5
Lower Shell	A533-B-Class 1	.08	.005	-30	70	10	106
Trans. Ring	A533-B-Class 1			-40	23	-37	107
Bottom Head	A533-B-Class 1			-10	42	-10	134
Core Region Weld		.06	.013	-50	16	-44	84
Weld HAZ				-70	-37	-70	130

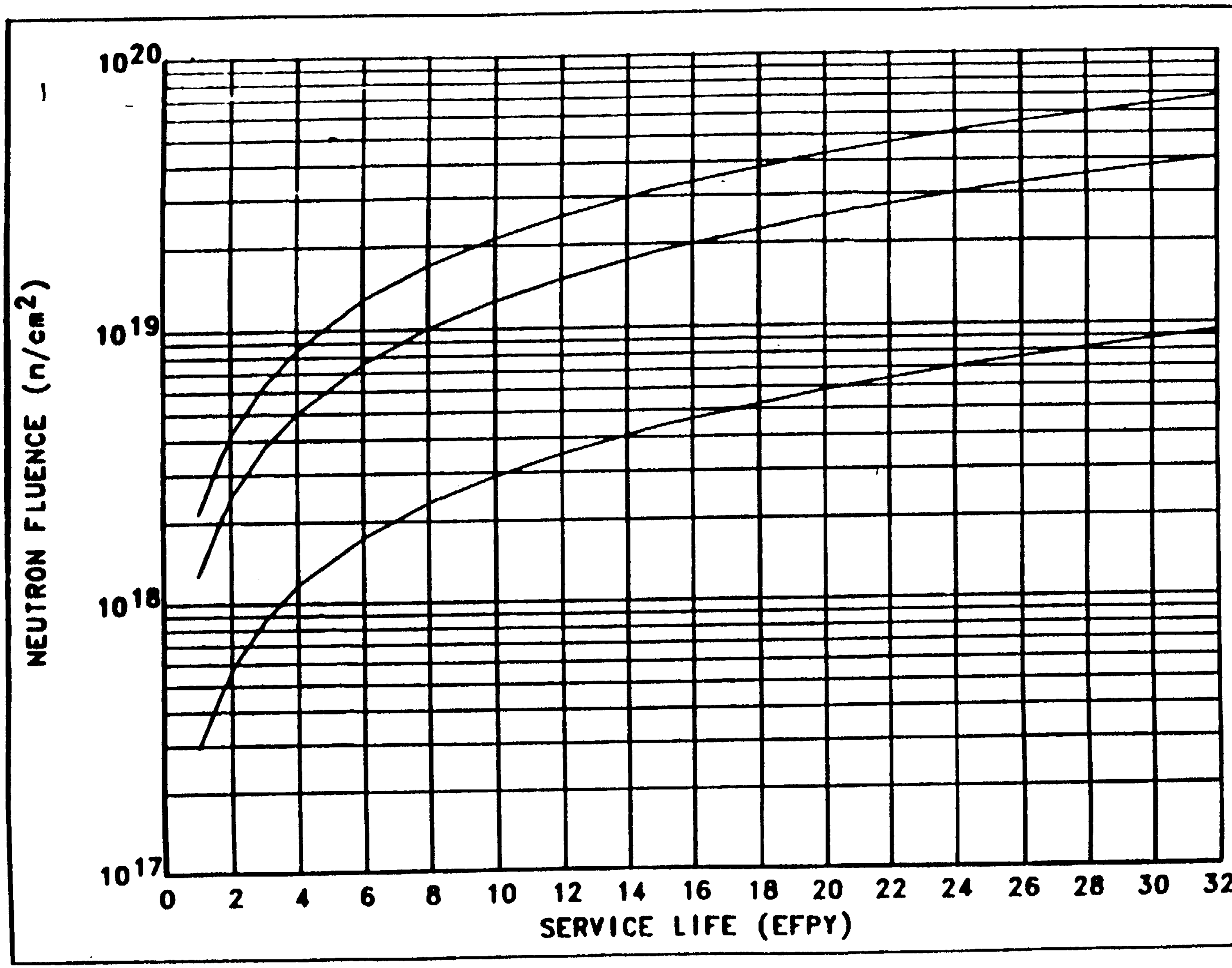
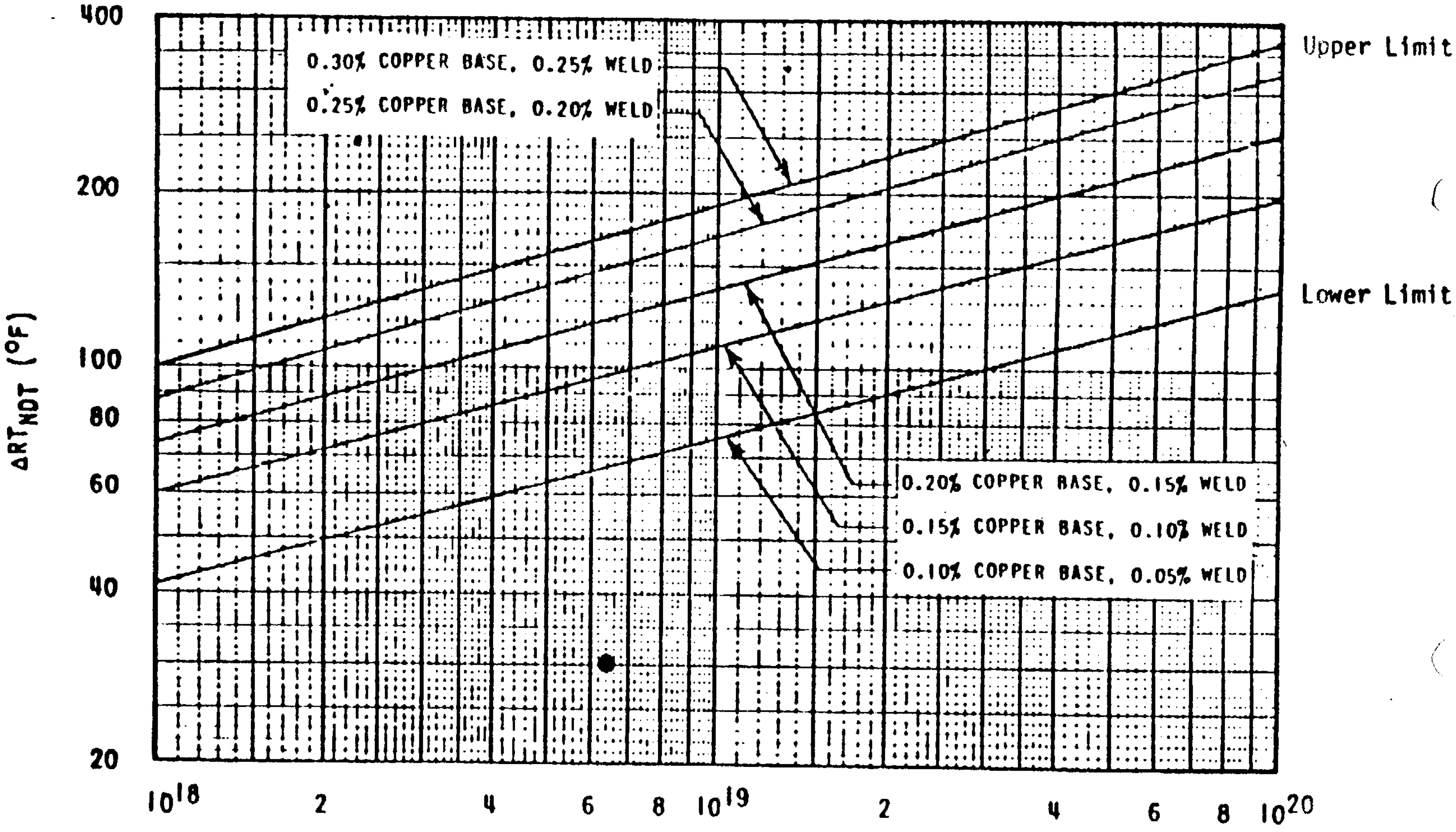


FIGURE B 3/4 4.1 FAST NEUTRON FLUENCE (E>1 MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)



● Weld and Plate (A9154-1) Metals

FIGURE B 3/4 4.2 Effect of Fluence and Copper on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to Irradiation at 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

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

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

where

K_{IM} is the stress intensity factor caused by membrane (pressure) stress

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

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

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 factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

COOLDOWN (Continued)

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and insures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4 T crack during heatup is lower than the K_{IR} for the 1/4 T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. Then the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Finally, the new 10 CFR 50[3] rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. The 10 CFR 50 rule states that the metal temperature of th closure flange regions must exceed the material RT_{NDT} by at least 120°F for normal operation when the pressure

[3]Code of Federal Regulations, 10 CFR 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Amended May 17, 1983 (48 Federal Register 24010).

exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for V. C. Summer Unit 1). Table B 3/4.4.1 indicates that the limiting RT_{NDT} of 10°F occurs in the head flange of V. C. Summer Unit 1, and the minimum allowable temperature of this region is 130°F at pressures greater than 621 psig.

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using the methods discussed. The derivation of the limit curves is presented in the NRC Regulatory Standard Review Plan.[4]

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. Charpy test specimens from Capsule U indicate that both the surveillance weld metal and core region intermediate shell plate code no. A9154-1 exhibited shifts in RT_{NDT} of 30°F at a fluence of 6.39 x 10^{18} n/cm^2. This shift is well within the appropriate design curve (Figure B 3/4.4.2) prediction. Therefore, the heatup and cooldown curves in Figures 3.4-2 and 3.4-3 are based on the trend curve in Figure B 3/4.4.2 and these curves are applicable up to 8 effective full power years (EFPY). The heatup curve in Figure 3.4-2 is not impacted by the new 10 CFR 50 rule. However, the cooldown curve in Figure 3.4-3 is impacted by this 10 CFR 50 rule.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown on the heatup and cooldown curves. The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 3.4-2. This is in addition to other criteria which must be met before the reactor is made critical.

The leak test limit curve shown in Figure 3.4-2 represents minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of References 2 and 4.

Figures 3.4-2 and 3.4-3 define limits for insuring prevention of nonductile failure.

[4]"Pressure-Temperature Limits," Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

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.

The OPERABILITY of two RHRSRVs or an RCS vent opening of at least 2.7 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 300°F. Either RHRSRV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of an HPSI pump and its injection into a water solid RCS.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

Introduction

In a letter from D.A. Nauman to H.R. Denton dated February 7, 1986, the South Carolina Electric & Gas Company (the licensee) proposed changes to Summer Technical Specification Section 3/4.4.9, "Pressure/Temperature Limits-Reactor Coolant System" and its bases. The licensee requested changes to the pressure temperature limits described in Figures 3.4-2 and 3.4-3 and surveillance capsule withdrawal schedule described in Table 4.4-5. The bases for these changes are the test results from the Summer surveillance program, which are contained in Report WCAP-10814, "Analysis of Capsule U from the South Carolina Electric and Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program." WCAP-10814 was submitted for staff review in a letter from D.A. Nauman to H.R. Denton dated November 8, 1985.

Evaluation

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material. The Summer reactor vessel was procured to ASME Code requirements, which specified fracture toughness testing to determine the initial RT_{NDT} for each vessel material. The test results indicate that the initial RT_{NDT} for the limiting beltline and closure flange region materials are 30°F and 10°F, respectively.

The increase in RT_{NDT} resulting from neutron irradiation damage was estimated by the licensee using the empirical relationship documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicated amount of neutron fluence and the amounts of residual elements (copper and phosphorus) in the beltline material. The neutron fluence used to predict neutron irradiation damage is based on the calculated neutron flux at the vessel location with peak flux. These neutron fluence predictions were verified by measurements from passive neutron flux monitors and by analysis, which was made with the DOT two-dimensional discrete ordinates code. Inputs into the analysis included 47 neutron energy groups, P3 expansion of the scattering cross section, and power distributions representative of time-averaged conditions derived from statistical studies of long-term operation of Westinghouse 4-loop plants. The cross sections used in the analysis were obtained from the SAILOR cross section library. Using this method of analysis, the measured average fast (E > 1.0 MeV) neutron flux derived from the five threshold reaction dosimeters, which were contained in Capsule U, is 1.80×10^{11} n/cm^2-sec with a standard deviation of ±7.5 percent. The calculated flux value of 2.09×10^{11} n/cm^2-sec exceeds all of the measured values, with calculation to experimental ratios ranging from 1.06 to 1.25. Since the calculated flux is greater than the measured flux from the capsule dosemetry, neutron fluence calculations using the calculated flux should conservatively predict neutron fluence.

The predicted amounts of neutron irradiation damage are based on design basis calculated neutron fluences and the increase in reference temperature (ΔRT_{NDT}) using the curves in Regulatory Guide 1.99, Rev. 1. The prediction curves in Regulatory Guide 1.99, Rev. 1 are dependent upon the amounts of residual elements in the beltline material. In Table B 3/4.4-1 of the Technical Specification, the licensee identified the residual elements in the core region welds and plates. Based on the chemical composition of the beltline materials that were reported in this Table, the limiting beltline material would be Plate No. A9154-1. Specimens from this plate were irradiated and tested as part of the Summer Surveillance program. The increase in ΔRT_{NDT} predicted for Plate No. A9154-1 by Regulatory Guide 1.99, Rev. 1 is 52°F. The increases in ΔRT_{NDT} measured from longitudinal and transversely oriented specimens were 40°F and 30°F, respectively. Since the predicted increase in ΔRT_{NDT} for the plate material is greater than the increase in ΔRT_{NDT} measured from the surveillance material, the prediction method in Regulatory Guide 1.99, Rev. 1, should conservatively predict the increase in ΔRT_{NDT} for the Summer beltline plate material.

The NRC staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981, to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage was calculated using design basis calculated neutron fluences and the Regulatory Guide 1.99, Rev. 1, prediction curves. The NRC staff concludes that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50 for 8 EFPY and therefore may be incorporated into the plant's Technical Specifications.

The reactor vessel material surveillance program must meet the requirements of Appendix H, 10 CFR 50, which was effective on July 26, 1983. Appendix H requires that the surveillance program meet the requirements of ASTM E 185-82 to the extent practical. ASTM E 185-82 requires that the Summer surveillance program have a minimum of four capsules. The time for capsule withdrawal recommended in the ASTM specification is dependent upon the effective full power years of operation, the capsule and vessel neutron fluences and the predicted increase in transition temperature of the encapsulated materials.

The Summer surveillance program contains six capsules; five are scheduled for removal and one is standby. The Summer surveillance capsules are scheduled for withdrawal at refueling outages that are either immediately before or after the ASTM recommended targets. As capsules can only be scheduled for withdrawal during refueling outages, the capsule withdrawal schedule documented in Table 4.4-5 of the Summer Technical Specification meets, to the extent practical, the withdrawal schedule tabulated in ASTM E 185-82.

The proposed changes to the reactor vessel capsule withdrawal schedule meet the requirements of Appendix H, 10 CFR 50, are acceptable to the NRC staff, and therefore may be incorporated into the plant's technical specifications.

The deletion of the reference to Figure 3.4-4 is acceptable to the NRC staff, because there is no Figure 3.4-4 in Technical Specifications.

Environmental Consideration

This amendment involves a change in the use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes in an inspection or surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

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

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