DMB-016

September 23, 1985

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
Post Office Box 14042; M.A.C. H-2
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 82 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TSs) in partial response to your application dated February 14, 1985.

This amendment updates the reactor pressure-temperature limits to eight effective full power years, based on the analysis of the first surveillance capsule. That portion of your amendment request dealing with deletion from the TSs of surveillance requirements for reactor vessel irradiation specimens has been addressed in Amendment 80 to the operating license.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next Biweekly Federal Register notice.

Sincerely,

Original signed by

Harley Silver, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 82 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD:OR:DL
RIngram	HSilver;cr	JStolz	M. KARMAN	GLainas
9/17/85	9/17/85	9/17/85	9/19/85	9/23/85

Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
Generating Plant

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 82
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated February 14, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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P PDR

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. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 82 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 23, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 82

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 4-24

3/4 4-26

3/4 4-27

3/4 4-28

B 3/4 4-9

B 3/4 4-10

B 3/4 4-11

B 3/4 4-12

B 3/4 4-13

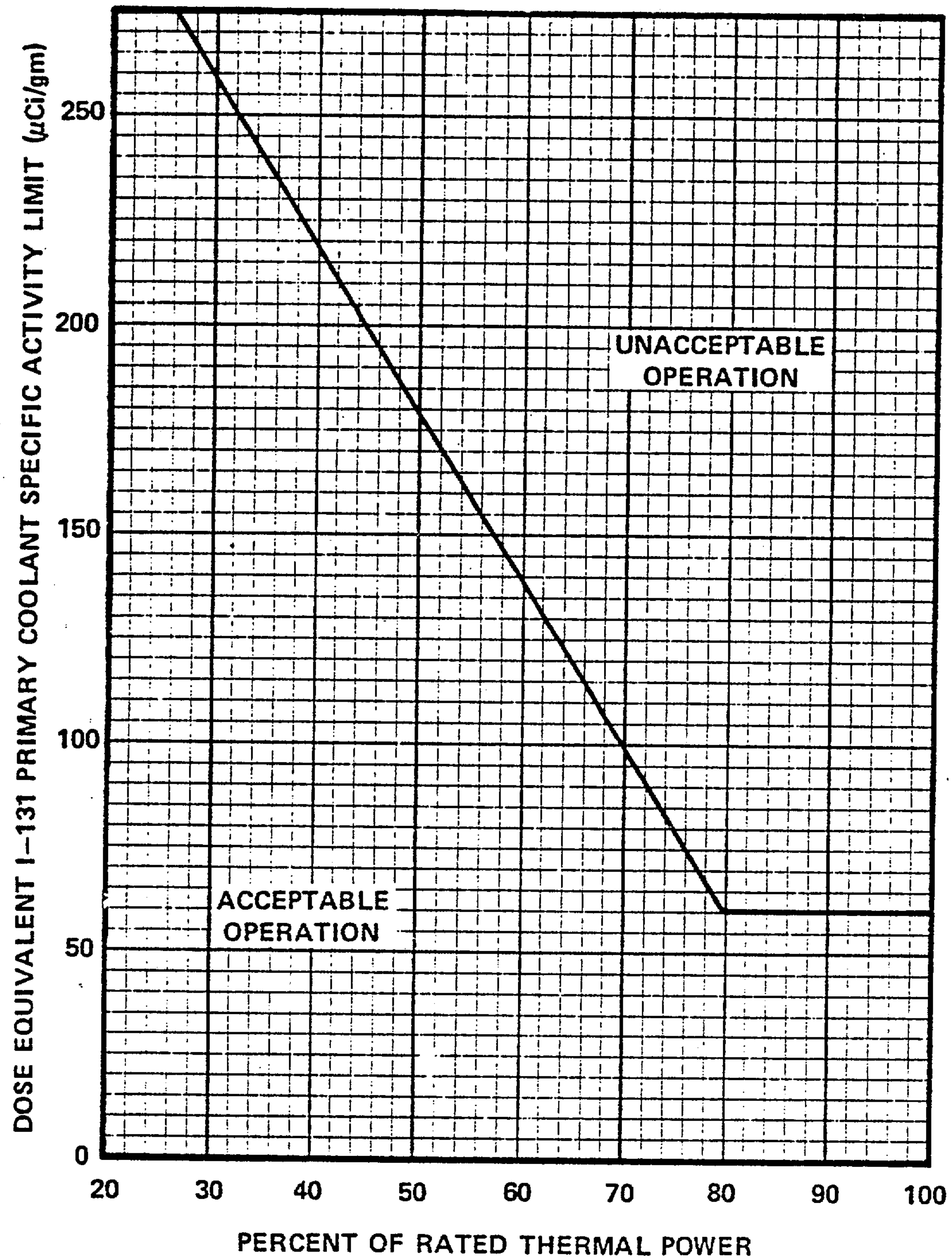


FIGURE 3.4-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity $> 1.0 \mu$Ci/gram Dose Equivalent I-131

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, 3.4-3, and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

b. For the temperature ranges specified below, the cooldown rates should be as specified (in any one hour period):

i.	$T > 270^{\circ}F$	$\leq 100^{\circ}F/Hr$,
ii.	$270^{\circ}F \geq T > 170^{\circ}F$	$\leq 50^{\circ}F/Hr$,
iii.	$170^{\circ}F \geq T$	$\leq 10^{\circ}F/Hr$,

and

c. A maximum temperature change of less than or equal to 5°F in any one hour period during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.

ACTION:

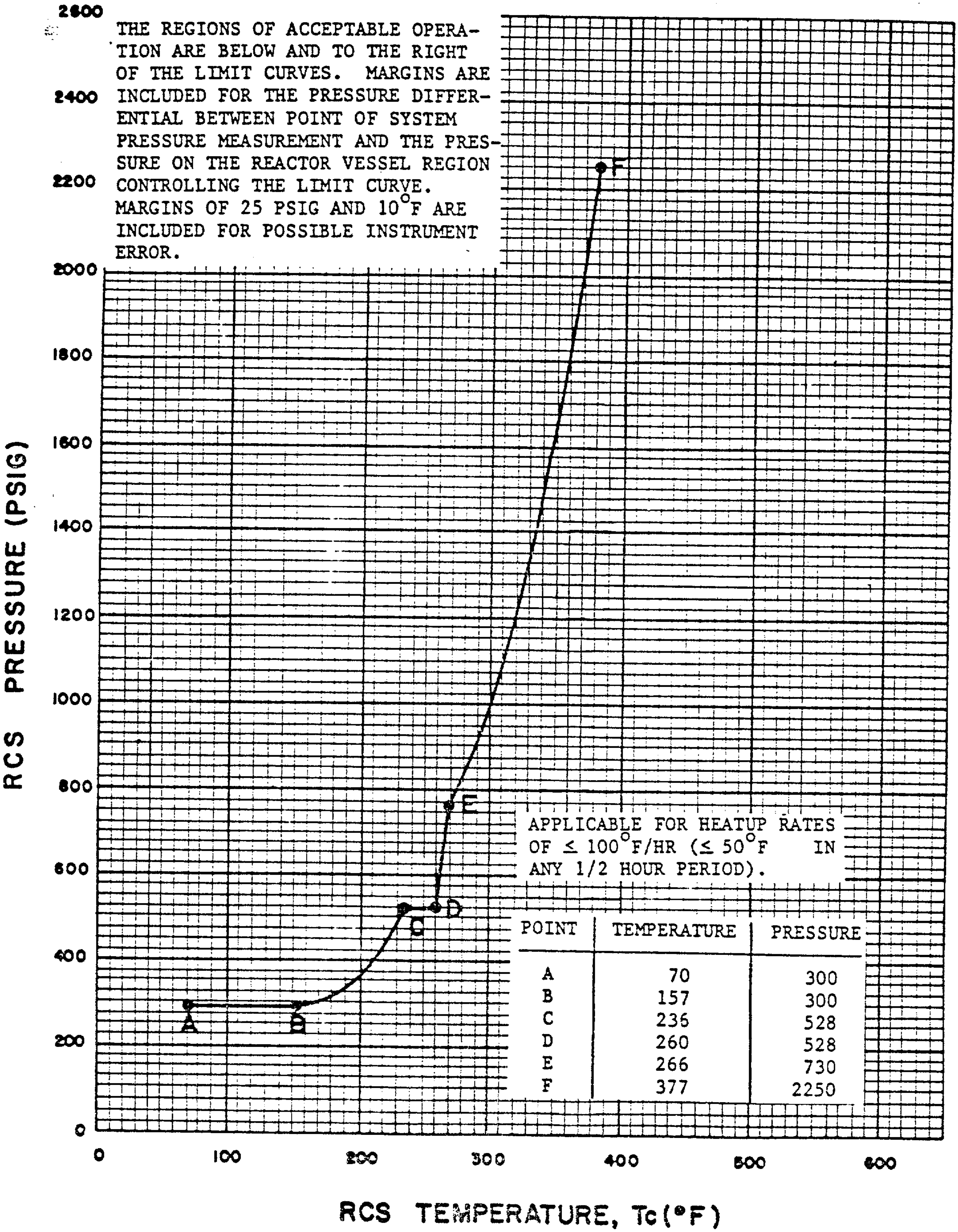
With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce 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.

FIGURE 3.4-2

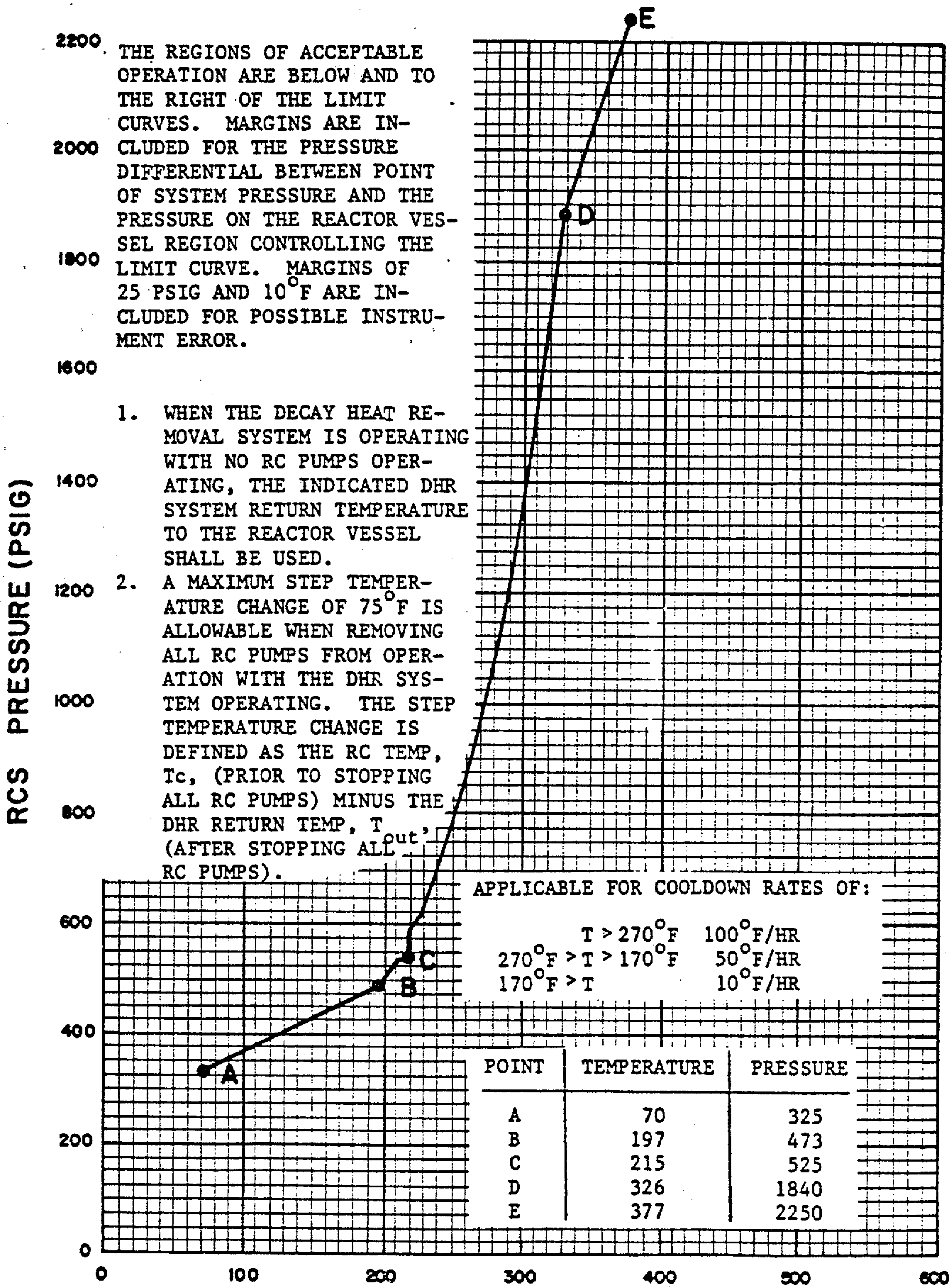
REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR HEATUP FOR FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	300
B	157	300
C	236	528
D	260	528
E	266	730
F	377	2250

FIGURE 3.4-3

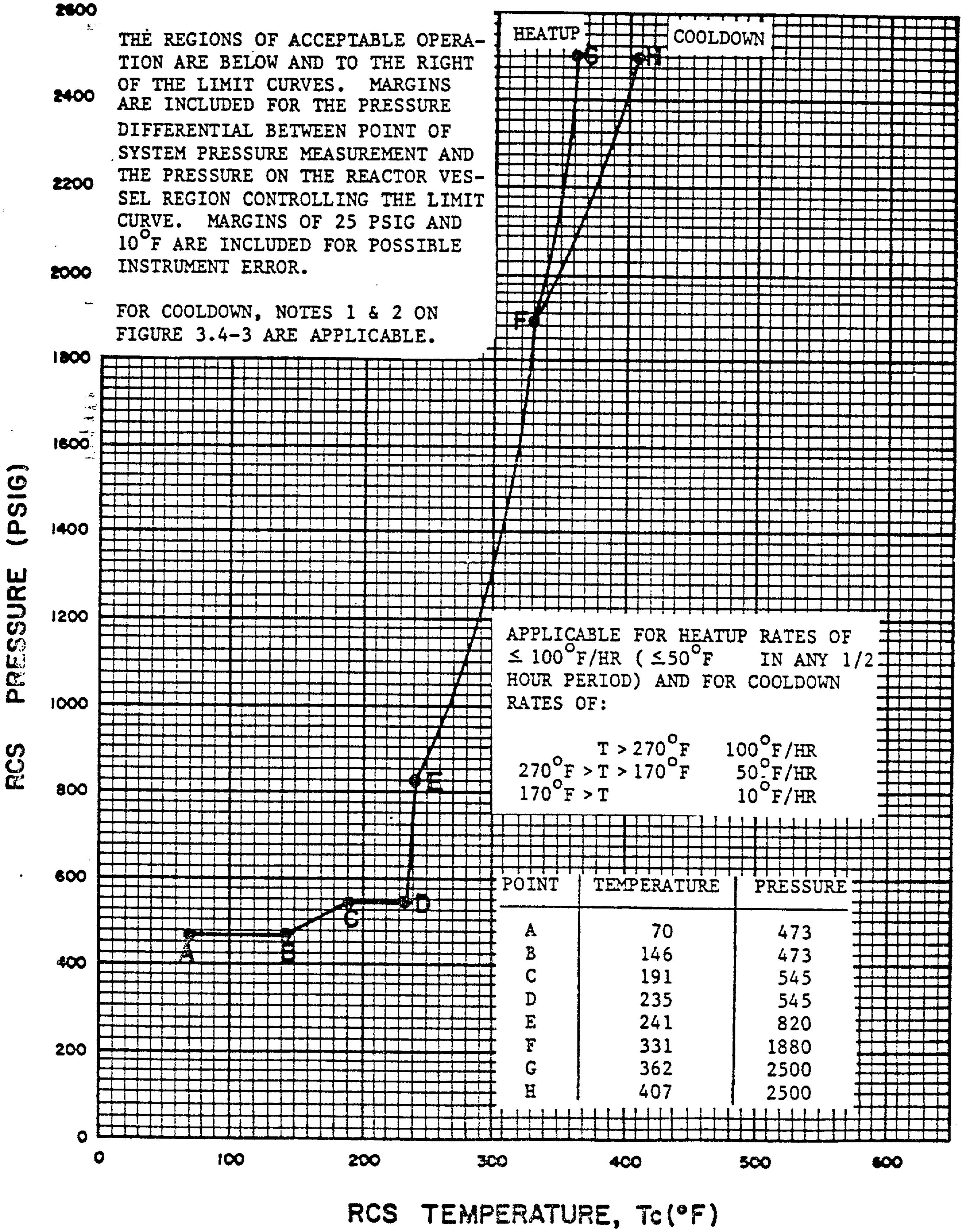
REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR COOLDOWN FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	325
B	197	473
C	215	525
D	326	1840
E	377	2250

FIGURE 3.4-4

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR HEATUP & COOLDOWN LIMITS FOR INSERVICE LEAK AND HYDROSTATIC TESTS FOR FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	473
B	146	473
C	191	545
D	235	545
E	241	820
F	331	1880
G	362	2500
H	407	2500

BASES TABLE 4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	CU %	P %	S %	RT NDT F	TRANS UPPER SHELF FT-LB	RT ADJUSTED NDT FOR 8 FULL POWER YEARS @ 1/4 T, °F	@ 3/4 T, °F
Nozzle Belt	SA-508 CL 2	.054	.008	.006	+10	183	26	17
*Upper Shell	SA-533B	.20	.008	.016	+20	88	90	
**Upper Shell	SA-533B	.20	.008	.016	+20	90	90	54
Lower Shell	SA-533B	.12	.013	.015	-20	119	26	2
Lower Shell	SA-533B	.12	.013	.015	+45	88	91	67
***Surveillance	Weld	.30	.020	.005	+43	63		
Upper Long	Weld	.20	.009	.009	(+20)****	66****	130	73
Upper Long	Weld	.105	.091	.004	(+20)****	66****	130	73
Upper Circum (60%)	Weld	.106	.014	.013	(+20)****	66****	NA	53
Upper Circum (40%)	Weld	.19	.021	.016	(+20)****	66****	128	NA
Middle Circum (100%)	Weld	.27	.014	.011	(+20)****	66****	177	96
Lower Long (100%)	Weld	.22	.015	.013	(+20)****	66****	134	75
Lower Circum (100%)	Weld	.20	.015	.021	(+20)****	66****	30	20
Out 1st Nozzle	Weld	.19	.021	.016	(+20)****	66****		
Middle Circum	Atypical weld	--	---	---	+90		136	112

* Surveillance Base Metal A
** Surveillance Base Metal B
*** Surveillance Weld
**** Estimated Value

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2, 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 100°F per hour. During cooldown, similar types of thermal stress occur. Thus, the cooldown limit curve, Figure 3.4-3, is also a composite curve which was prepared based upon the same type analysis as the heatup curve 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. Additionally, during cooldown and heatup at the higher temperatures, the most conservative limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are the vertical segments of the limit lines on Figures 3.4-3 and 3.4-4, respectively. (These limits will not require adjustments due to the neutron fluences.)

During the first several years of service life, the most limiting Reactor Coolant System regions are the closure head region (due to mechanical loads resulting from bolt pre-load) and the reactor vessel outlet nozzles. Nozzle sensitivity is caused by the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the beltline region of the reactor vessel becomes the most limiting region due to material irradiation.

For the service period for which the limit curves are established, the pressure/temperature limits were obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and the most sensitive material in the beltline region. The lowest pressure calculated for these three regions becomes the maximum allowable pressure for the fluid temperature used in the calculation. The calculated pressure/temperature curves are adjusted by 25 PSI and 10°F for possible instrument errors. The pressure limit is also adjusted for the pressure differential between the point of pressure measurement and the limiting component for all combinations of reactor coolant pump operations.

Irradiation damage to the beltline region can be quantified by determining the decrease in the temperature at which the metal changes from ductile to brittle fracture (ΔRT_{NDT}). The unirradiated transverse impact properties of the beltline region have been determined for those materials for which sufficient amounts of materials were available and are listed on Table 4-1. The adjusted reference temperatures on Table 4-1 are calculated by adding the predicted radiation-induced change in the reference temperature (ΔRT_{NDT}) and the unirradiated reference temperature. (The assumed unirradiated RT_{NDT} of the closure head region and of the outlet nozzle steel forgings was 60°F.) The adjusted RT_{NDT}s of the beltline region materials at the end of the eighth full power year are listed on Table 4-1 for the one-quarter and three-quarter wall thickness of the vessel wall.

Bases Figure 4-1 illustrates the calculated peak neutron fluence, for several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules, as a function of exposure time. Bases Figure 4-2 illustrates the design curves for predicting the radiation-induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. Thus, using these two figures and information on Table 4-1, shifts in the RT_{NDT} can be predicted over the full service life of the vessel.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating the reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the RT_{NDT} determined from the surveillance capsule is different from the calculated RT_{NDT} for the equivalent capsule radiation exposure. The pressure and temperature limits shown on Figures 3.4-2 and 3.4-4 for reactor criticality, and for inservice leak and hydrostatic testing, have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4 4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrate that the valves are fully open at the forces assumed in the safety analysis.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

INTRODUCTION

In a letter from G. R. Westafer to H. R. Denton dated February 14, 1985, the Florida Power Corporation (the licensee) requested an amendment to the Technical Specifications for Crystal River Unit 3 (CR-3). In part, the proposed amendment would update the reactor pressure-temperature limits to eight effective full power years (EFPY) based on the analyses of the first surveillance capsule. The analyses of the first surveillance capsule are documented in Babcock & Wilcox Report BAW 1679, Rev. 1, entitled, "Analyses of Capsule CR-3B, Florida Power Corporation, Crystal River Unit 3, Reactor Vessel Materials Surveillance Program." The portion of the licensee's request dealing with deletion of surveillance requirements for reactor vessel irradiation specimens from the Technical Specifications has been addressed by the Safety Evaluation previously issued in support of Amendment 80 to the CR-3 operating license.

DISCUSSION

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 beltline materials.

The CR-3 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the RT_{NDT} for each reactor vessel material. Hence, the initial RT_{NDT} for materials in the closure flange and beltline region of the CR-3 reactor vessel could not be determined in accordance with the test requirements of the ASME Code. Therefore, the initial RT_{NDT} for these materials must be estimated from test data from other similar materials used for fabrication of reactor vessels in the nuclear industry.

The limiting closure flange material is the Closure Flange Head Plate, which was fabricated to ASME Code SA 533 GR.B requirements. The RT_{NDT} for this material was estimated as 60°F. This value was estimated in accordance with the criteria in B&W Topical Report BAW-10046A, Rev. 1, July 1977. This topical report was approved by the NRC staff for referencing in licensing applications in a letter from S. A. Varga to J. H. Taylor dated June 22, 1977.

The licensee indicates that the limiting materials in the CR-3 reactor vessel beltline region are the weld metals. The unirradiated RT_{NDT} for the weld metals were estimated in accordance with the criteria in B&W Topical Report BAW-10046A, Rev. 1, July 1977.

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, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of copper and phosphorus in the beltline material.

The neutron flux was calculated using a discrete ordinate solution of the Boltzman Transport equation with the two dimensional code, DOT 3.5. The calculated flux at the peak vessel flux location was normalized to the measured activities from the Capsule CR-3B dosimetry. There were six different detector reactions measured from the CR-3B dosimeters. The measured activities from these reactions were 20-30 percent less than the calculated activities. The corresponding normalization factor for the six detector reactions varied from .68 to .81. Since the .81 normalizing factor would produce the least amount of neutron flux reduction, it was used for predicting neutron fluences. Based on a normalizing constant of .81, the vessel's maximum fluence after eight EFPY was predicted to be 1.9×10^{18} n/cm^2 (E > 1 MeV) at the 1/4 location and 4.4×10^{17} n/cm^2 (E > 1 MeV) at the 3/4 T location.

The limiting materials in the CR-3 reactor vessel beltline are the weld metals. The estimated values for the amounts of copper and phosphorus in the beltline welds are documented in Table 6 of B&W Topical Report BAW-1511P, dated October 1980. The source of these values are chemical analyses from vessel weld dropouts and surveillance weld samples, which were fabricated using the same heats of weld wire as the CR-3 beltline welds. Since the amounts of copper and phosphorus in a weld are based upon the amounts of these elements in the weld wire, the use of chemical analyses from welds fabricated using the same heats of weld wire as the CR-3 beltline welds will provide acceptable values for the amounts of copper and phosphorus in the CR-3 beltline welds.

The CR-3B surveillance capsule contains plate and HAZ material from heat C-4344-1 and atypical weld metal designated as WF-209-1. In Table 1 of this Safety Evaluation, we have compared the amount of increase in RT_{NDT} predicted using Regulatory Guide 1.99, Rev. 1, to the amount observed from the surveillance weld metal and plate material. The test results from the HAZ material produced large data scatter. Hence, we have not included the test results from this material in our evaluation. Since the amount of increase in RT_{NDT} predicted using Regulatory Guide 1.99, Rev. 1, is greater than that observed on the surveillance material, the Regulatory Guide 1.99, Rev. 1, method of predicting neutron irradiation will provide conservative estimates of the increase in RT_{NDT} resulting from neutron irradiation damage.

EVALUATION

We 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 to the beltline materials was estimated using the method recommended by the staff in Regulatory Guide 1.99, Rev. 1. The amounts of copper and phosphorus in the limiting CR-3 beltline weld were the values reported in Topical Report BAW-1511P. The neutron fluence used to predict neutron irradiation damage was based on a .81 normalizing factor. Our conclusion is that the proposed pressure-temperature limits meet the safety margins of Appendix G, 10 CFR 50, for eight EFPY and are therefore acceptable and may be incorporated into the plant's Technical Specifications.

The licensee has also proposed to delete the criticality curve from Figure 3.4-2. This is acceptable since the requirements of TS 3.1.1.4 continue to apply and will assure that criticality will not occur at unacceptable conditions of temperature and pressure.

According to B&W Topical Report 1543A, Rev. 2, entitled, "Integrated Reactor Vessel Material Surveillance Program," surveillance capsules CR-3C and CR-3D have been irradiated and removed from the CR-3 vessel. The dosimeter test results from these capsules should be compared to the test results from capsule CR-3B. If these test results indicate that the normalizing factor from the CR-3B capsule dosimeters are nonconservative, the licensee's pressure-temperature limits should be adjusted accordingly.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have 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 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.

Dated: September 23, 1985

Principal Contributor: B. Elliot

Table I

Increase in RT_{NDT} for Capsule CR-3B Surveillance Material

Surveillance Material	Neutron Fluence x 10^{18} n/cm^2 (E> 1 MeV)	Increase in RT_{NDT} (°F) Observed from Capsule Test Data	Predicted by Reg. Guide 1.99, Rev. 1
Plate C4344-1	1.0	21	57
Weld Metal WF-209-1	1.17	28	105