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

February 7, 1991

Docket No. 50-302

Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Operations
Licensing
P. O. Box 219-NA-2I
Crystal River, Florida 32629

Dear Mr. Beard:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT RE: PRESSURE/TEMPERATURE CURVES (TAC NOS. 75304 AND 71483)

The Commission has issued the enclosed Amendment No. 133 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 (TS) in partial response to your application dated October 31, 1989, as supplemented by your letters dated March 30, 1990, and August 10, 1990.

This amendment revises Technical Specification 3.4.9.1, including Figures 3.4-2, 3.4-3, and 3.4-4. This revision provides reactor coolant system heatup and cooldown pressure/temperature (P/T) curves for operation up to 15 effective full power years.

During the staff's review of the proposed TS changes, it was noted on page B3/4 4-11 of your application that a line was inadvertently dropped from the first paragraph. This error has been corrected by the staff.

Your application also proposed revised and new TS for low temperature overpressure protection (LTOP). Proposed LTOP TS are still under review, and are not part of this amendment. The application also included TS in the format of the Technical Specification Improvement Program (TSIP). Only current format TS changes have been reviewed and approved by the staff in this amendment.

This completes our efforts on TAC No. 71483. TAC No. 75304 will remain open until issues regarding the LTOP TS are resolved.

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

Sincerely,

(Original signed by)

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.133 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	LA:PD22	PM:PD22	PM:PD22	D:PD22	OGC	SRXB	EMCB
NAME	DMiller	JWilliams	HSilver	HBerkow	DAJ	RJones	CYCheng
DATE	1/23/91	1/24/91	1/24/91	2/7/91	1/28/91	/ /91	/ /91

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DATED: February 7, 1991

AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. DPR-72-CRYSTAL RIVER UNIT 3

Docket File
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Florida Power Corporation

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72GLL

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. 133
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 October 31, 1989, as supplemented on March 30, 1990 and August 10, 1990, 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. 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. 133 , 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 and shall be implemented within 30 days of issuance.

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: February 7, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 133

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached 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.

Remove	Insert
3/4 4-24	3/4 4-24
3/4 4-26	3/4 4-26
3/4 4-27	3/4 4-27
3/4 4-28	3/4 4-28
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11

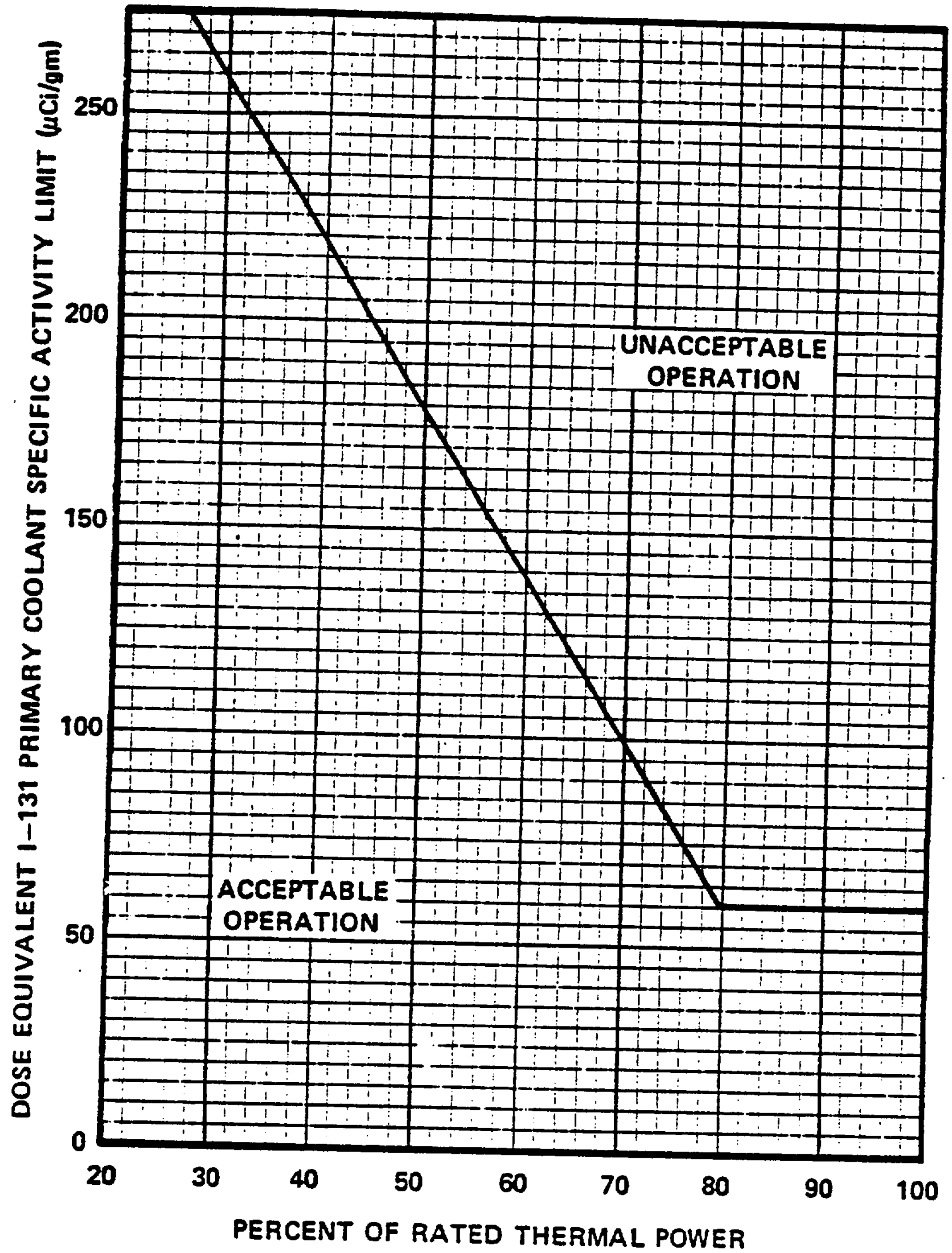


FIGURE 3.4-1

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

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

a. A maximum heatup of 50°F in any one hour period,

b. For the temperature ranges specified below, the cooldown rates should be as specified:

i.	$T > 280°F$	$\leq$ 50°F in any 1/2 hour period
ii.	$150°F < T \leq 280°F$	$\leq$ 25°F in any 1/2 hour period
iii.	$T \leq 150°F$	$\leq$ 10°F in any 1/2 hour period

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:

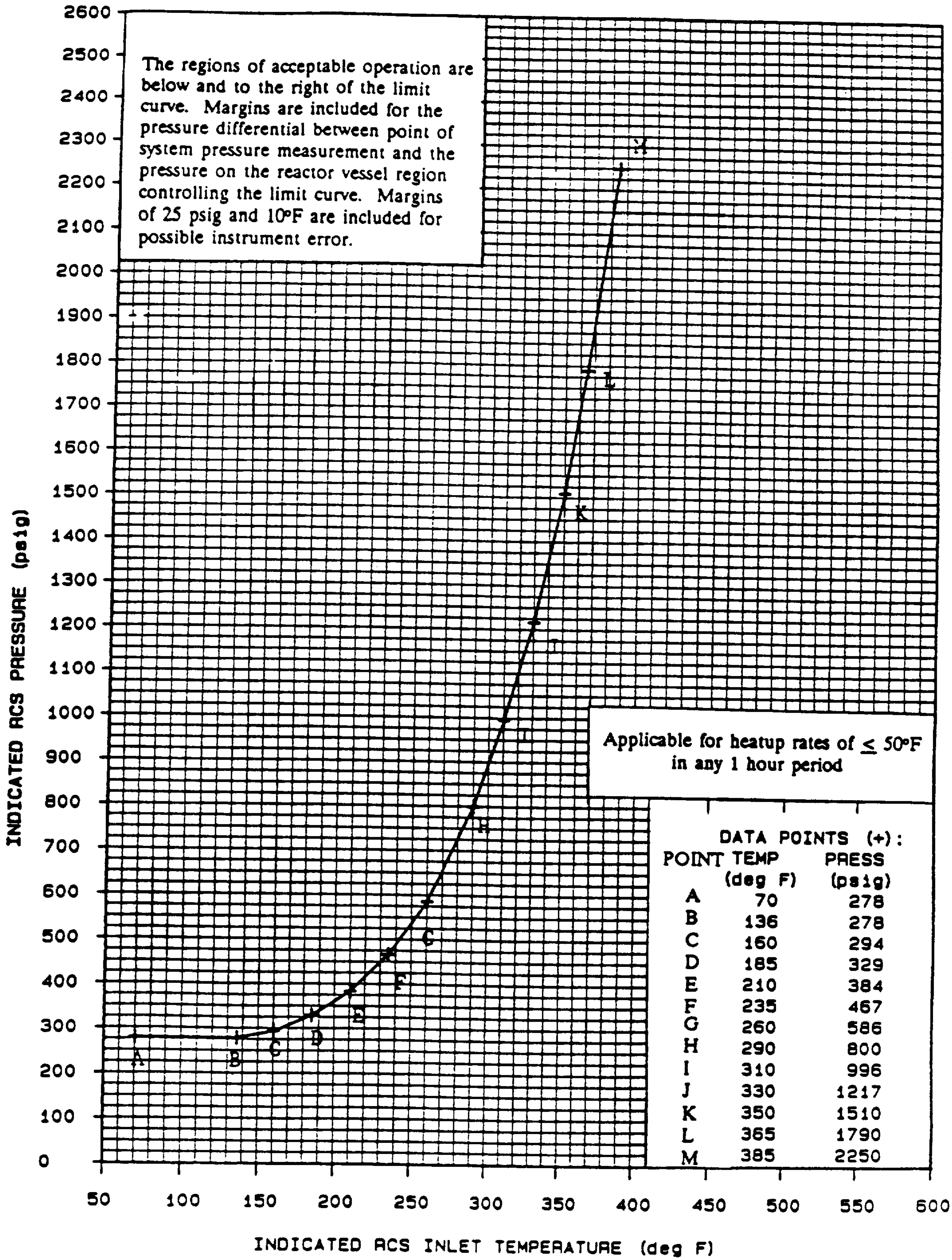
With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor 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 15 EFPY

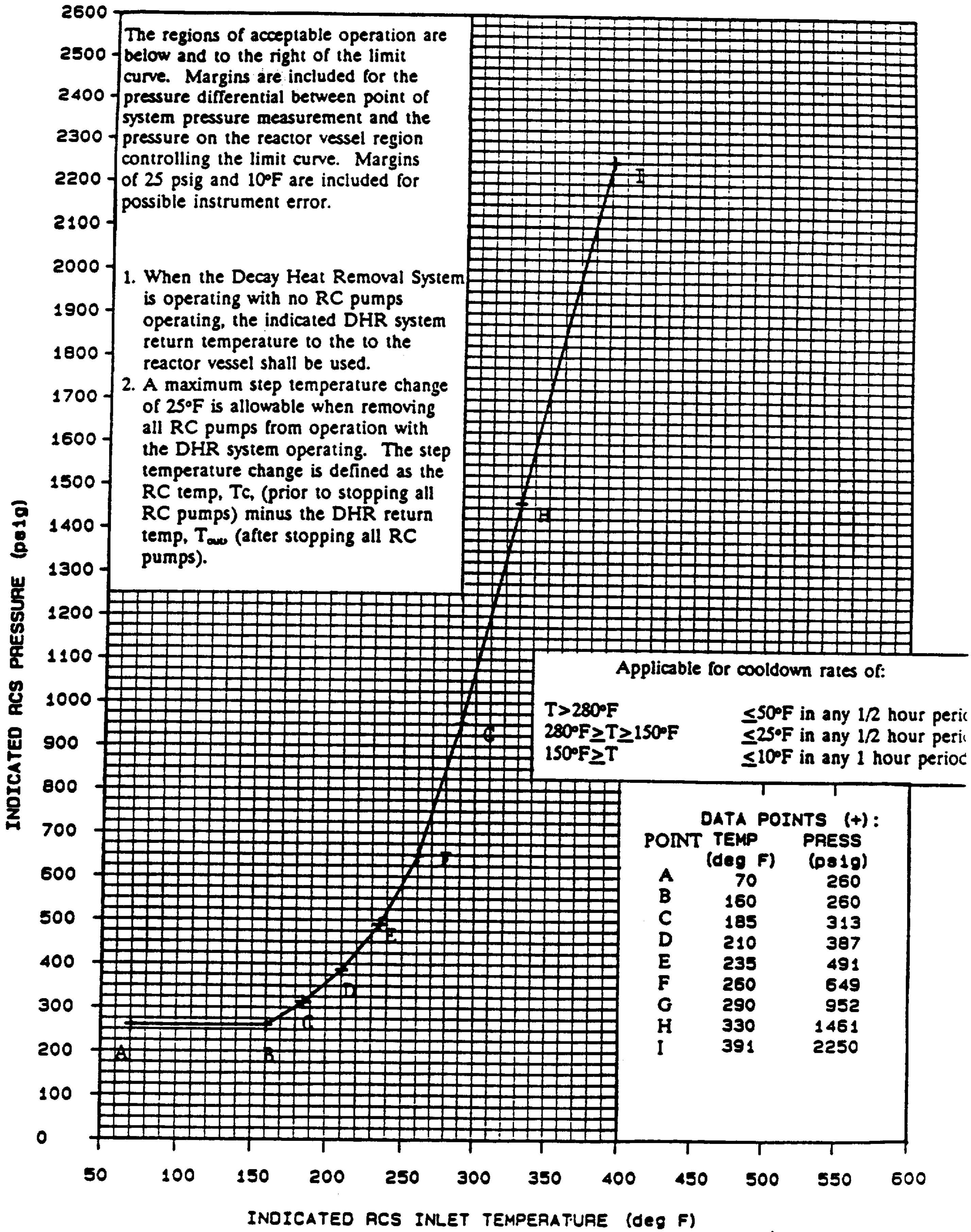


DATA POINTS (+):

POINT	TEMP (deg F)	PRESS (psig)
A	70	278
B	136	278
C	160	294
D	185	329
E	210	384
F	235	467
G	260	586
H	290	800
I	310	996
J	330	1217
K	350	1510
L	365	1790
M	385	2250

Figure 3.4-3

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR COOLDOWN FOR FIRST 15 EFPY



The regions of acceptable operation are below and to the right of the limit curve. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins of 25 psig and 10°F are included for possible instrument error.

1. When the Decay Heat Removal System is operating with no RC pumps operating, the indicated DHR system return temperature to the to the reactor vessel shall be used.
2. A maximum step temperature change of 25°F is allowable when removing all RC pumps from operation with the DHR system operating. The step temperature change is defined as the RC temp, Tc, (prior to stopping all RC pumps) minus the DHR return temp, T_{out} (after stopping all RC pumps).

Applicable for cooldown rates of:

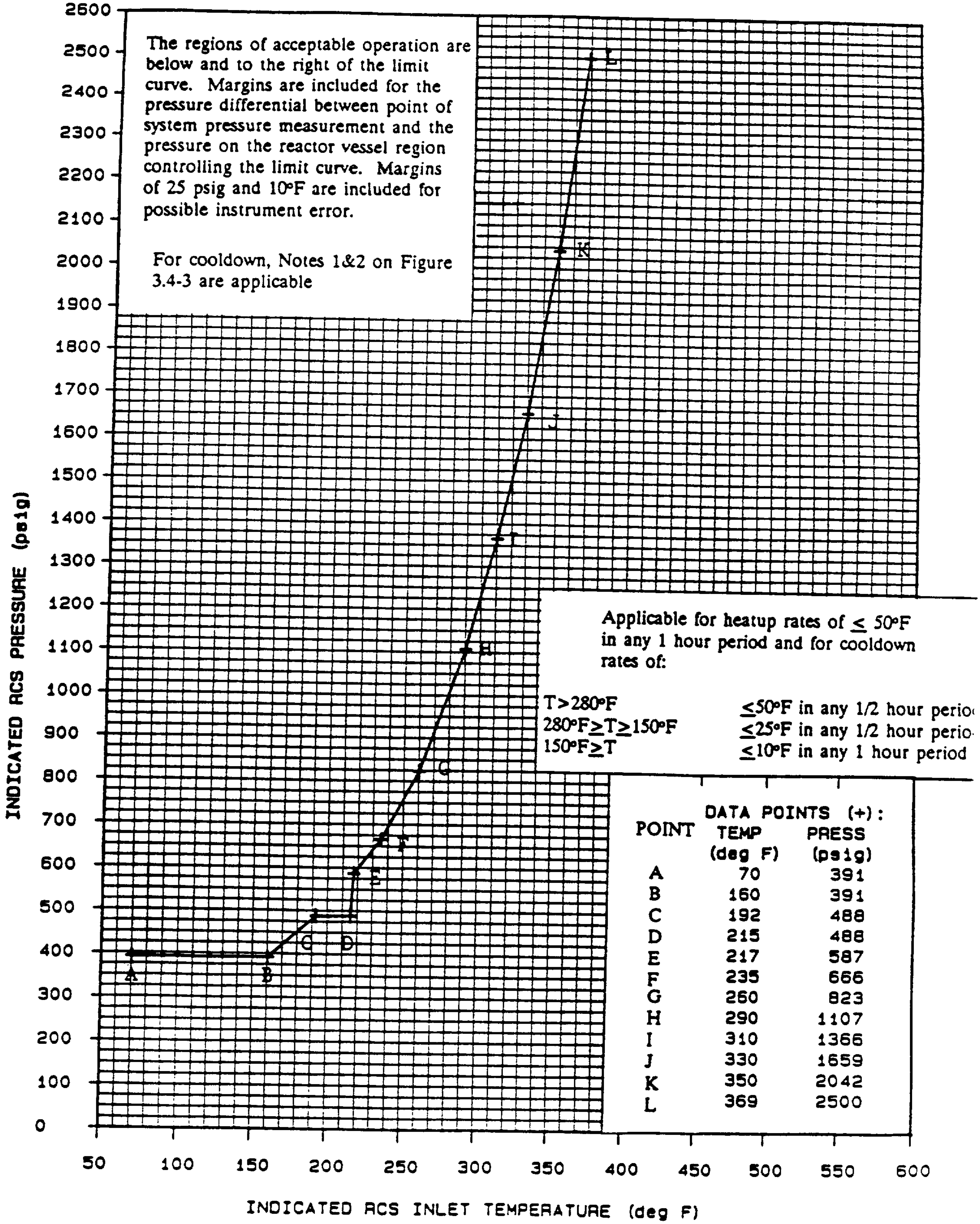
T>280°F	≤50°F in any 1/2 hour peri
280°F≥T≥150°F	≤25°F in any 1/2 hour peri
150°F≥T	≤10°F in any 1 hour period

DATA POINTS (+):

POINT	TEMP (deg F)	PRESS (psig)
A	70	260
B	160	260
C	185	313
D	210	387
E	235	491
F	260	649
G	290	952
H	330	1461
I	391	2250

Figure 3.4-4

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR INSERVICE LEAK & HYDROSTATIC TESTS FOR FIRST 15 EFPY



DATA POINTS (+):

POINT	TEMP (deg F)	PRESS (psig)
A	70	391
B	160	391
C	192	488
D	215	488
E	217	587
F	235	666
G	260	823
H	290	1107
I	310	1366
J	330	1659
K	350	2042
L	369	2500

DELETED

DELETED

BASES TABLE 4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	CU W/O	NI W/O	RT NDT/F	TRANS UPPER SHELF FT-LB	RT ADJUSTED NDT FOR 21 FULL POWER YEARS @ 1/4 t, °F	RT ADJUSTED NDT FOR 21 FULL POWER YEARS @ 3/4 T, °F
Nozzle Belt	SA-508 CL 2	.10	.72	+10	183	117	101
*Upper Shell	SA-533B	.20	.54	+20	88	162	126
**Upper Shell	SA-533B	.20	.54	+20	90	162	126
Lower Shell	SA-533B	.12	.58	-20	119	87	66
Lower Shell	SA-533B	.12	.58	+45	88	142	121
***Surveillance	Weld	.30	---	+43	63		
Upper Long	Weld	.29	.55	(+20)****	66****	199	153
Upper Long	Weld	.29	.55	(+20)****	66****	199	153
Upper Circum (60%)	Weld	.18	.63	(+20)****	66****	NA	133
Upper Circum (40%)	Weld	.26	.61	(+20)****	66****	187	NA
Middle Circum (100%)	Weld	.35	.59	(+20)****	66****	222	168
Lower Long (100%)	Weld	.29	.55	(+20)****	66****	199	153
Lower Circum (100%)	Weld	.31	.59	(+20)****	66****	69	65
Out 1st Nozzle	Weld	.19	---	(+20)****	66****		
Middle Circum (100%)	A typical weld	.41	.10	+90		211	180

* 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 50°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 comprehensive stresses at the outside wall.

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 twenty first full power year are listed on Table 4-1 for the one-quarter and three-quarter wall thickness of the vessel wall.

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 Figure 3.4-4 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. 133 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

By letter dated October 31, 1989, as supplemented March 30, 1990 and August 10, 1990, Florida Power Corporation (FPC, or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DRP-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3).

The proposed amendment revises the reactor coolant system (RCS) pressure/temperature (P/T) limits. The current P/T limits are for 8 effective full power years (EFPY) of operation. The proposed amendment provides P/T limits based upon predicted reactor vessel neutron embrittlement for up to 15 EFPY, using the methods of Regulatory Guide (RG) 1.99, Rev. 2. Generic Letter 88-11 requested that licensees use RG 1.99, Rev. 2 to predict neutron irradiation effects. The proposed amendment also changes limits on plant heatup and cooldown rate to be more representative of actual plant capability.

FPC also requested TS revisions for low temperature overpressure protection (LTOP). This Safety Evaluation does not address the proposed LTOP changes, which will be addressed in a separate amendment.

This Safety Evaluation applies only to P/T limits for the current format TSs. Revisions proposed in the format of the Technical Specification Improvement Program will be reviewed as part of that effort.

EVALUATION

To evaluate the P/T limits, the staff used 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 evaluated the effect of neutron irradiation embrittlement on each beltline material in the CR-3 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 15 EFPY was the upper-to-lower shell girth weld, WF-70, with 0.35% copper (Cu), 0.59% nickel (Ni), and an initial RT_{ndt} of -6°F.

The licensee has removed four surveillance capsules from CR-3. The results from capsules B, C, D, and F were published in Babcock & Wilcox reports BAW-1679, BAW-1898, BAW-1899, and BAW-2049, respectively. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld WF-70, the staff calculated the ART to be 191.1°F at 1/4T (T = reactor vessel beltline thickness) and 141.4°F for 3/4T at 15 EFPY. The staff used a neutron fluence of 2.99E18 n/cm^2 at 1/4T and 1.08E18 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2.

The licensee calculated a higher ART (203°F) than the staff at 1/4T for the limiting material WF-70 because the licensee used a conservative safety margin. At the 3/4 location, the licensee selected the atypical weld as the limiting material. In 1978 Babcock & Wilcox performed chemical analyses of samples of beltline welds in CR-3. The results indicated that one of the welds had atypical concentrations of nickel and silicone for the Linde 80 submerged-arc welds in the reactor vessel. Based on the analysis, the atypical weld showed a higher than normal value of RT_{ndt}. Thus, the licensee calculated a more conservative ART of 171°F at the 3/4 location. The staff judges that the licensee's ARTs are conservative and, therefore, acceptable. Substituting the ARTs of 203°F and 171°F into equations in SRP 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 of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 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% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions 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 60°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Finally, during the staff's review of the proposed TS changes, it was noted on page B3/4 4-11 of the application that a line was inadvertently dropped from the first paragraph. This error has been corrected by the staff.

Based on the above evaluation, the staff concludes that the proposed changes to the P/T limits for up to 15 EFPY are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to 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: February 7, 1991

Principal Contributor:
John Tsao