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ACRS (16)
DRoss
TBAbernathy
JRBuchanan
OPA (Clare Miles)

Dockets Nos. 50-250
and 50-251

Florida Power and Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
P. O. Box 013100
Miami, Florida 33101

Gentlemen:

The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-31 and Amendment No. 23 to Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Units Nos. 3 and 4. The amendments consist of changes to the Technical Specifications in response to your application dated May 21, 1976.

These amendments revise the Technical Specifications to change the reactor coolant pressure-temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature. Please note that the same pressure-temperature limits apply to both Unit No. 3 and Unit No. 4 for five effective full-power years (EFPY).

Copies of the Safety Evaluation and the FEDERAL REGISTER Notice are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 24 to License DPR-31
2. Amendment No. 23 to License DPR-41
3. Safety Evaluation
4. FEDERAL REGISTER Notice

cc w/encls:
See next page

9

OFFICE	ORB #3	ORB #3	OELD	ORB #3		
SURNAME	CParrish	DElliott:mjf		GLear		
DATE	4/ /77	4/ /77	4/ /77	4/ /77		

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

April 20, 1977

Dockets Nos. 50-250
and 50-251

Florida Power and Light Company
ATTN: Dr. Robert E. Uhrig
Vice President
P. O. Box 013100
Miami, Florida 33101

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The Commission has issued the enclosed Amendment No. 24 to Facility Operating License No. DPR-31 and Amendment No. 23 to Facility Operating License No. DPR-41 for the Turkey Point Nuclear Generating Units Nos. 3 and 4. The amendments consist of changes to the Technical Specifications in response to your application dated May 21, 1976.

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Sincerely,

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

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1. Amendment No. 24 to License DPR-31
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3. Safety Evaluation
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cc w/encls:
See next page

cc:

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Washington, D. C. 20036

Mr. Ed Maroney
Bureau of Intergovernmental Relations
725 South Bronough Street
Tallahassee, Flordia 32304

Honorable Dewey Knight
County Manager of Metropolitan
Dade County
Miami, Florida 33130

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ATTN: Mr. Henry Yaeger
Plant Manager
Turkey Point Plant
P. O. Box 013100
Miami, Florida 33101

Chief, Energy Systems Analysis Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
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Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
345 Courtland Street, N. E.
Atlanta, Georgia 30308

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Florida International University
Miami, Florida 33199

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 24
License No. DPR-31

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated May 21, 1976, 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-31 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 24, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 20, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 24

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-31

DOCKET NO. 50-250

Replace pages i, v, 3.1-2 thru 3.1-4, and B3.1-1 thru B3.1-4 with the attached revised pages. Replace Figure 3.1-1 with Figures 3.1-1a and 3.1-1b. Add Figures B3.1-1 and B3.1-2.

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Amendment No. ~~14~~, 24

LIST OF FIGURES

2. PRESSURE-TEMPERATURE LIMITS

The Reactor Coolant System (except for the pressurizer) pressure and temperature shall be limited during heatup, cooldown, criticality (except for low power physics tests), and inservice leak and hydrostatic testing in accordance with the limit lines shown on Figures 3.1-1a and 3.1-1b. Allowable pressure-temperature combinations are BELOW AND TO THE RIGHT of the lines on the Figures. Heatup and cooldown rate limits are:

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

b. A maximum cooldown rate of 100 °F in any one hour.

c. A maximum temperature change of ≥ 5 °F in any one hour during hydrostatic testing operation above system design pressure.

The pressurizer pressure and temperature shall be limited in accordance with the following:

d. The pressurizer shall be limited to a maximum heatup or cooldown rate of 200 °F in any one hour.

e. The pressurizer shall be limited to a maximum Reactor Coolant System spray water temperature differential of 320 °F.

With any of the above limits exceeded, restore the temperature and/or pressure within the limits within 30 minutes; determine that the RCS or pressurizer remains acceptable for continued operations or, if at power, be in at least Hot Shutdown within the next 6 hours and Cold Shutdown within the following 30 hours.

The reactor shall not be made critical unless the moderator temperature coefficient is zero or negative. When the coefficient is greater than zero, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization. These moderator temperature coefficient conditions do not apply to low power physics tests.

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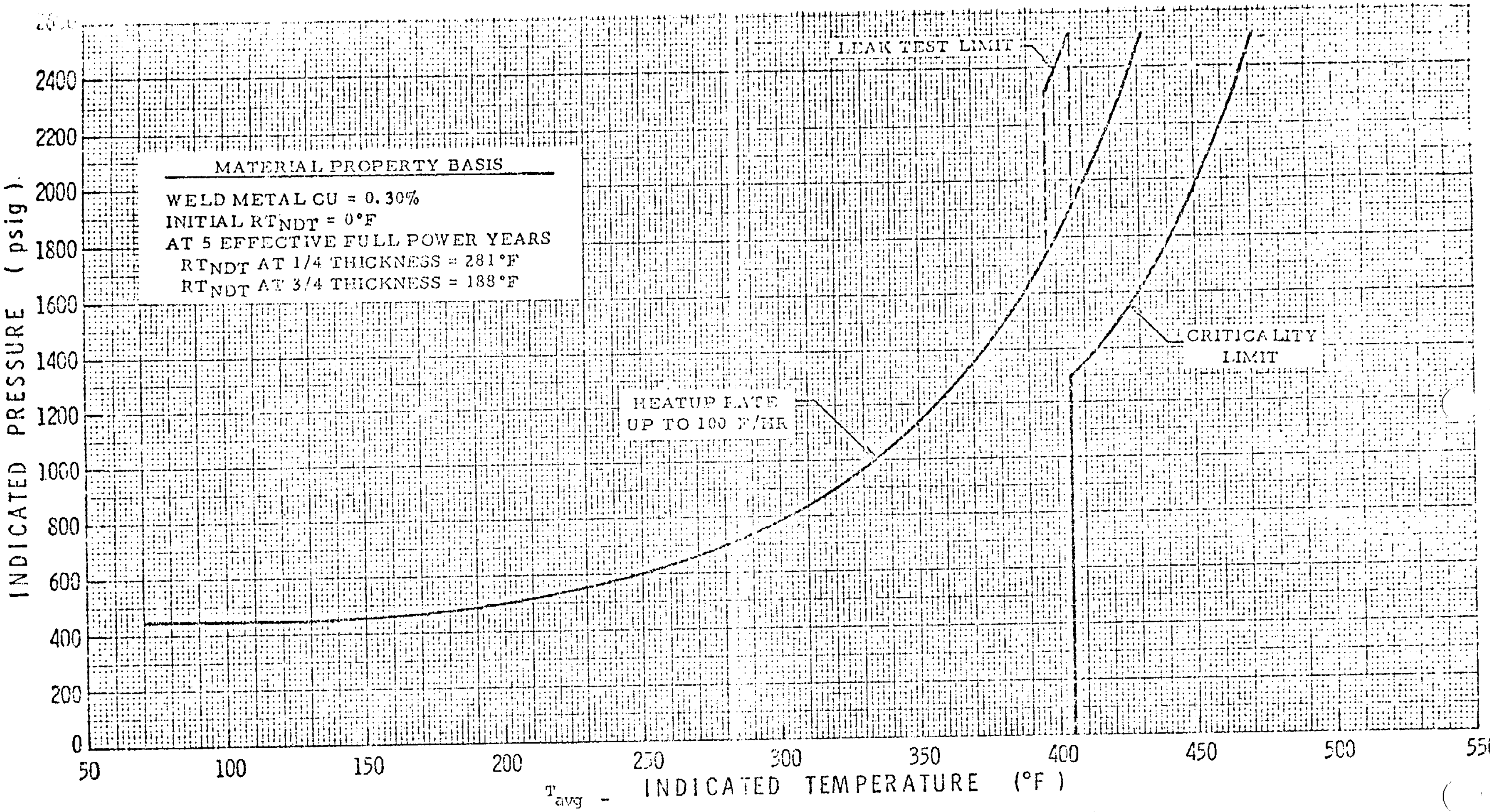
3. LEAKAGE

a. Any reactor coolant system leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication (ex. water inventory changes, radiation level increases, visual or audible indication). A leak shall be assumed to exist until it is determined that no unsafe condition exists and that the indicated leak cannot be substantiated. Leakage of reactor coolant through reactor pump seals and system valves to connecting closed systems from which coolant can be returned to the reactor coolant system shall not be considered as leakage except that such losses shall not exceed 30 gpm.

b. If a reactor coolant system leakage indication is proven real, and is not evaluated as safe, or exceeds 10 gpm, reactor shutdown shall be initiated within 24 hours of the initial indication.

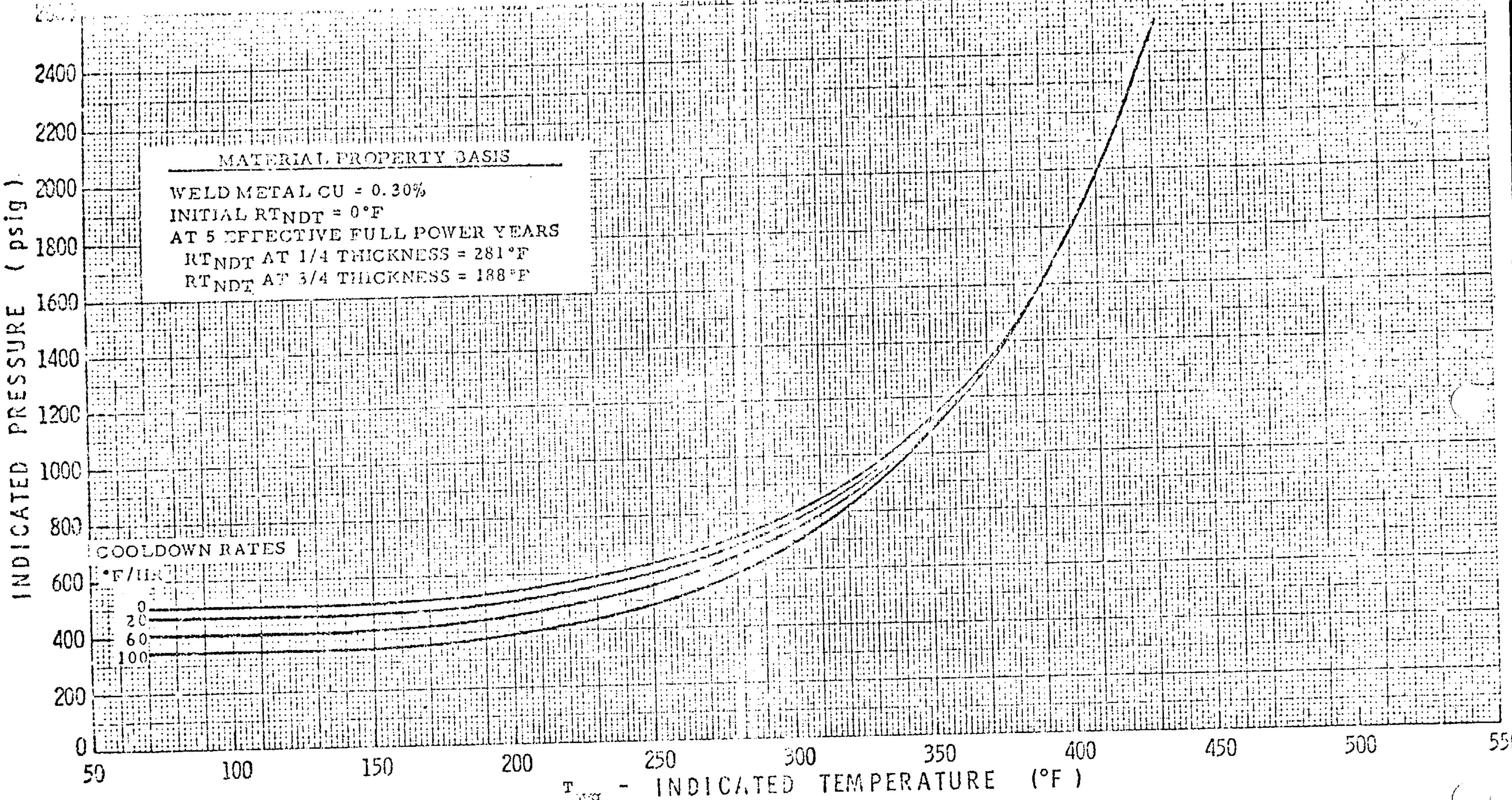
c. If reactor coolant leakage exists through a fault in the system boundary that cannot be isolated (ex. vessels, piping, valve bodies) the reactor shall be shutdown and cool down to cold shutdown shall be initiated within 24 hours.

d. The safety evaluation shall consider the source and magnitude of the leak, rates of change of detection variables, and if shutdown is required this evaluation shall be used to determine shutdown rates and conditions. A written log of the action taken shall be made as soon as practicable. The evaluation shall assure that no potential gross leak is developing and that potential release of activity will be within the guidelines of 10CFR20.



TURKEY POINT REACTOR COOLANT HEATUP LIMITATIONS APPLICABLE FOR PERIODS UP TO 5 EFFECTIVE FULL POWER YEARS.

Figure 3.1-1a



TURKEY POINT REACTOR COOLANT COOLDOWN LIMITATIONS APPLICABLE FOR PERIODS UP TO 5 EFFECTIVE FULL POWER YEARS

Figure 3.1-16

B3.1 BASES FOR LIMITING CONDITIONS FOR OPERATION, REACTOR COOLANT SYSTEM

1. Operational Components

The specification requires that a sufficient number of reactor coolant pumps be operating to provide coast down core cooling in the event that a loss of flow occurs. The flow provided will keep DNBR well above 1.30. When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the reactor coolant system volume in approximately one half hour.

Each of the pressurizer safety valves is designed to relieve 293,330 lbs. per hr. of saturated steam at the valve set point.[1] Below 350 F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve lifting pressure would be less than the capacity of a single valve. Also, two safety valves have capacity greater than the maximum surge rate resulting from complete loss of load.[2]

2. Pressure/Temperature Limits

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in

Section 4.1.5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for prevention of brittle fracture.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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. Subsequently, 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 curves are composite curves 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. The cooldown limit curves 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 the service period.

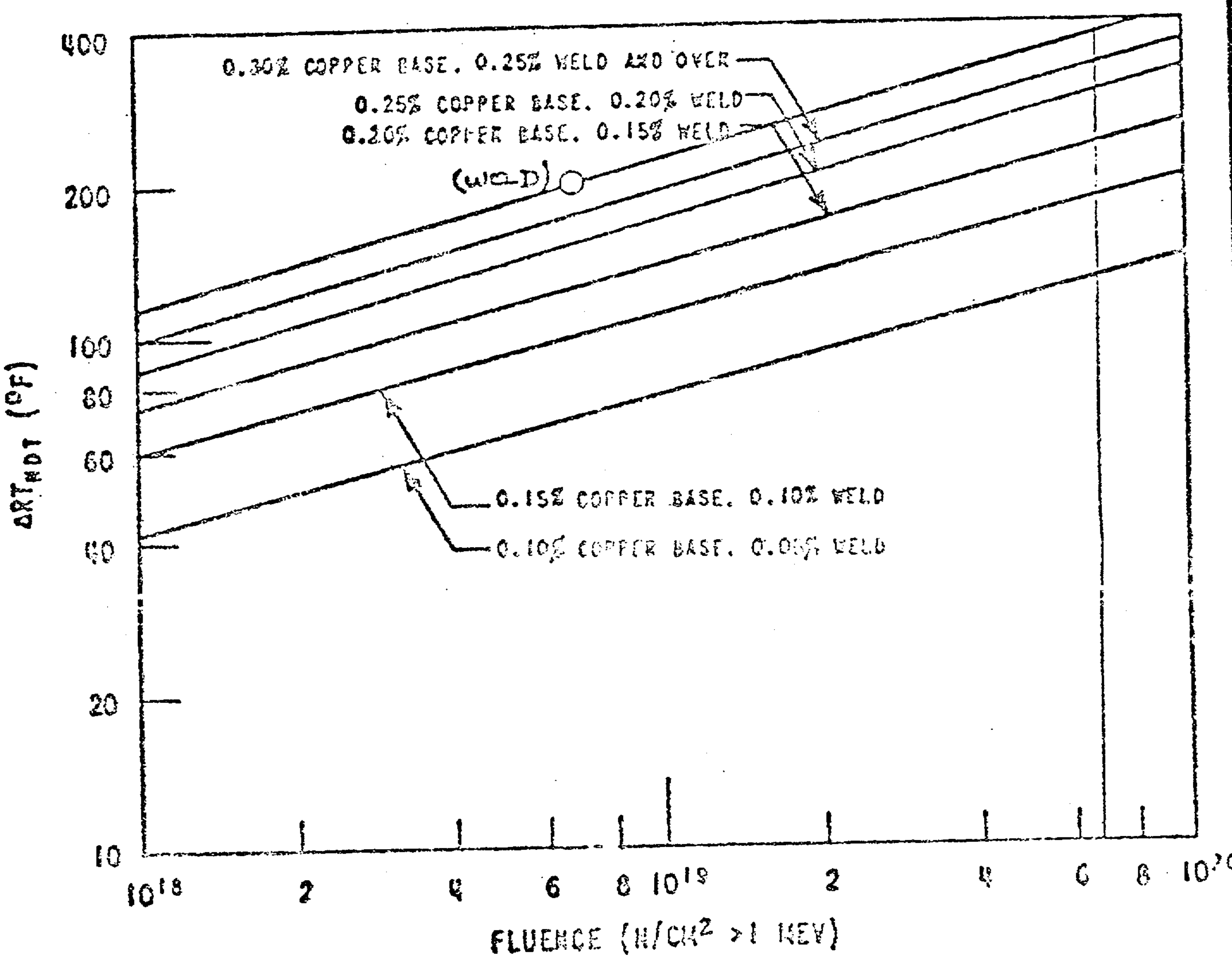
The reactor vessel materials have been tested to determine their initial RT_{NDT}. Adjusted reference temperatures, based upon the fluence and copper content of the material in question, are then determined. The heatup and cooldown limit curves include the shift in RT_{NDT} at the end of the service period shown on the heatup and cooldown curves.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, 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 has a definite relationship to the spectra at the vessel inside radius, the measured transition shift for a sample can be related with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown 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.

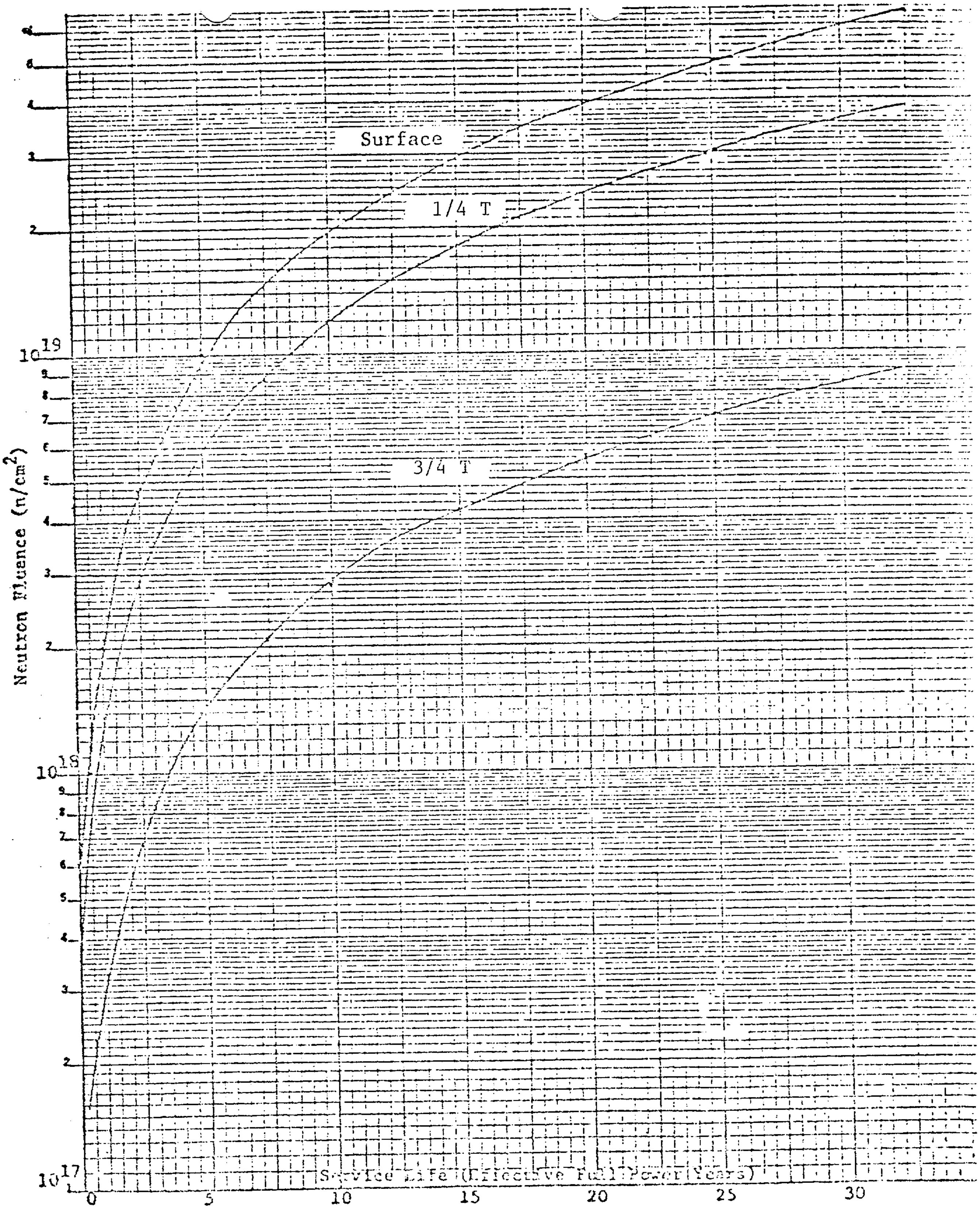
The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.2-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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



Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550 °F Temperature

Figure B3.1-1



Fast Neutron Fluence (E > 1 MEV) as a
Function of Effective Full Power Years

Figure B3.1-2

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. DPR-41

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated May 21, 1976, 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-41 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 23, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 20, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 23

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-251

Replace pages i, v, 3.1-2 thru 3.1-4, and B3.1-1 thru B3.1-4 with the attached revised pages. Replace Figure 3.1-1 with Figures 3.1-1a and 3.1-1b. Add Figures B3.1-1 and B3.1-2.

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LIST OF FIGURES

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The pressurizer pressure and temperature shall be limited in accordance with the following:

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3. LEAKAGE

a. Any reactor coolant system leakage indication in excess of 1 gpm shall be the subject of an investigation and evaluation initiated within 4 hours of the indication (ex. water inventory changes, radiation level increases, visual or audible indication). A leak shall be assumed to exist until it is determined that no unsafe condition exists and that the indicated leak cannot be substantiated. Leakage of reactor coolant through reactor pump seals and system valves to connecting closed systems from which coolant can be returned to the reactor coolant system shall not be considered as leakage except that such losses shall not exceed 30 gpm.

b. If a reactor coolant system leakage indication is proven real, and is not evaluated as safe, or exceeds 10 gpm, reactor shutdown shall be initiated within 24 hours of the initial indication.

c. If reactor coolant leakage exists through a fault in the system boundary that cannot be isolated (ex. vessels, piping, valve bodies) the reactor shall be shutdown and cool down to cold shutdown shall be initiated within 24 hours.

d. The safety evaluation shall consider the source and magnitude of the leak, rates of change of detection variables, and if shutdown is required this evaluation shall be used to determine shutdown rates and conditions. A written log of the action taken shall be made as soon as practicable. The evaluation shall assure that no potential gross leak is developing and that potential release of activity will be within the guidelines of 10CFR20.

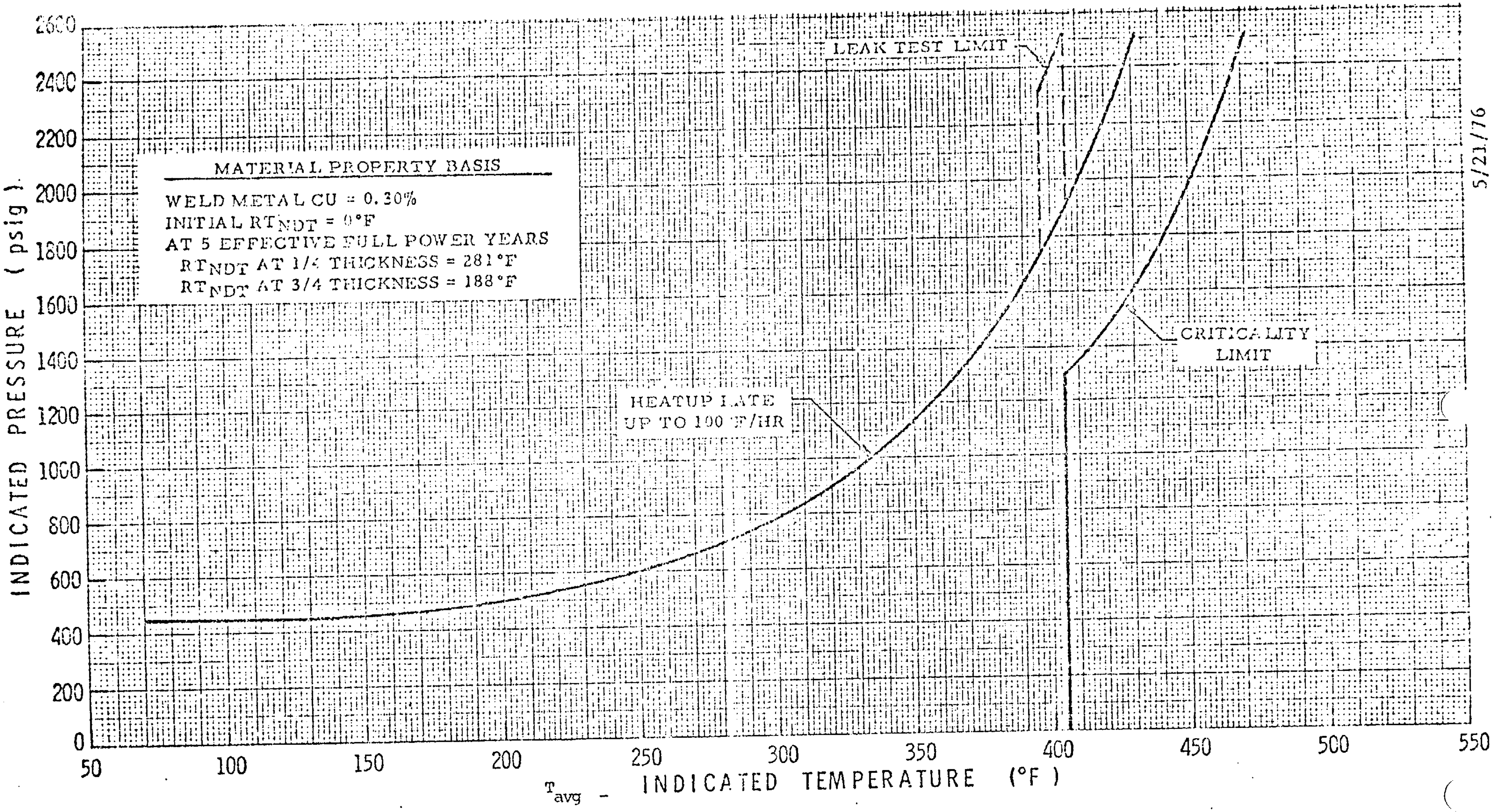


Figure 3.1-1a TURKEY POINT UNIT 4 REACTOR COOLANT HEATUP LIMITATIONS APPLICABLE FOR PERIODS UP TO 5 EFFECTIVE FULL POWER YEARS.

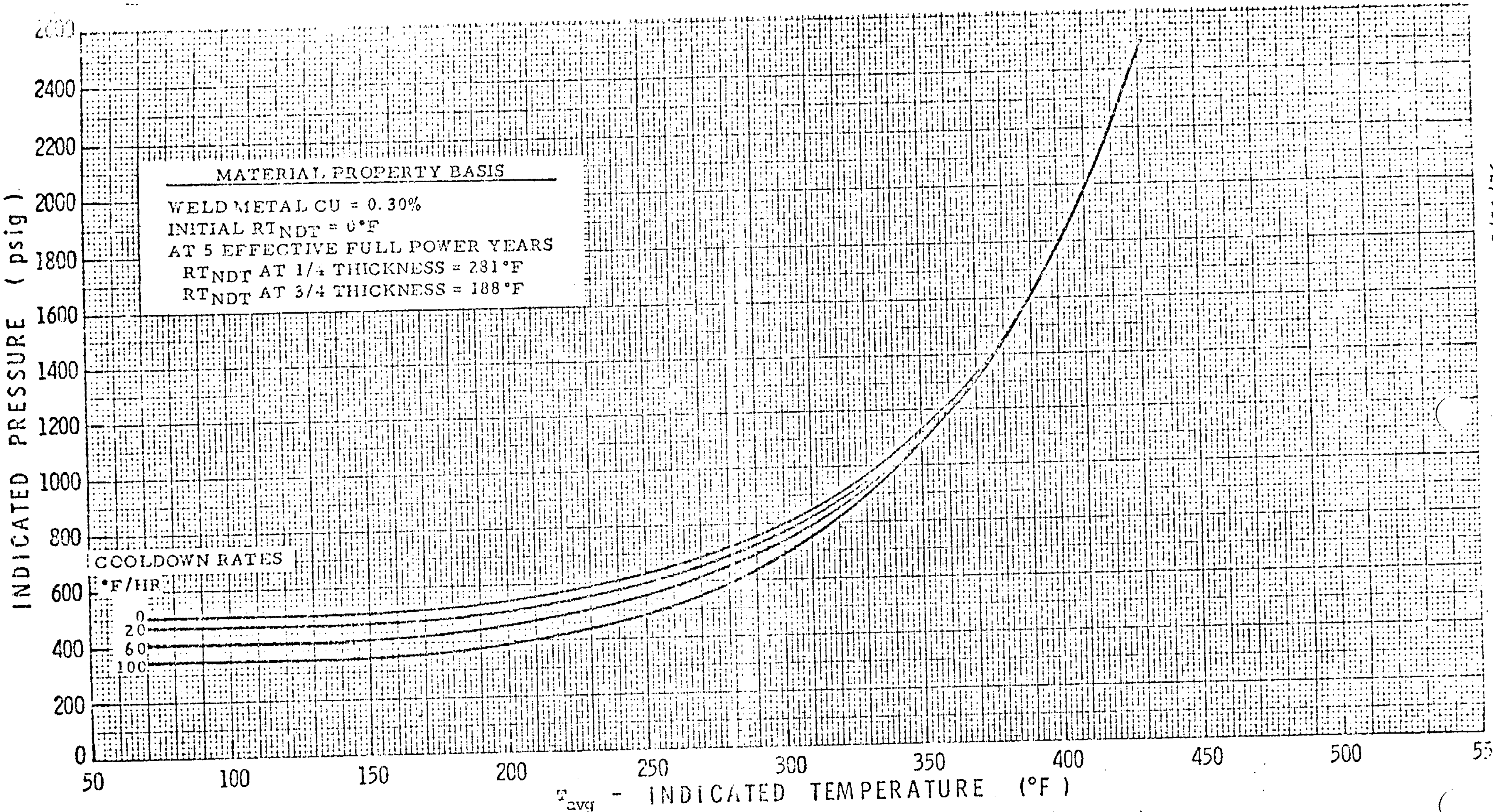


Figure 3.1-16. TURKEY POINT UNIT 4 REACTOR COOLANT COOLDOWN LIMITATIONS APPLICABLE FOR PERIODS UP TO 5 EFFECTIVE FULL POWER YEARS.

5/21/76

B3.1 BASES FOR LIMITING CONDITIONS FOR OPERATION, REACTOR COOLANT SYSTEM

1. Operational Components

The specification requires that a sufficient number of reactor coolant pumps be operating to provide coast down core cooling in the event that a loss of flow occurs. The flow provided will keep DNBR well above 1.30. When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the reactor coolant system volume in approximately one half hour.

Each of the pressurizer safety valves is designed to relieve 293,330 lbs. per hr. of saturated steam at the valve set point.(1) Below 350 F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve lifting pressure would be less than the capacity of a single valve. Also, two safety valves have capacity greater than the maximum surge rate resulting from complete loss of load.(2)

2. Pressure/Temperature Limits

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in

Section 4.1.5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for prevention of brittle fracture.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

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. Subsequently, 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 curves are composite curves 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. The cooldown limit curves 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 the service period.

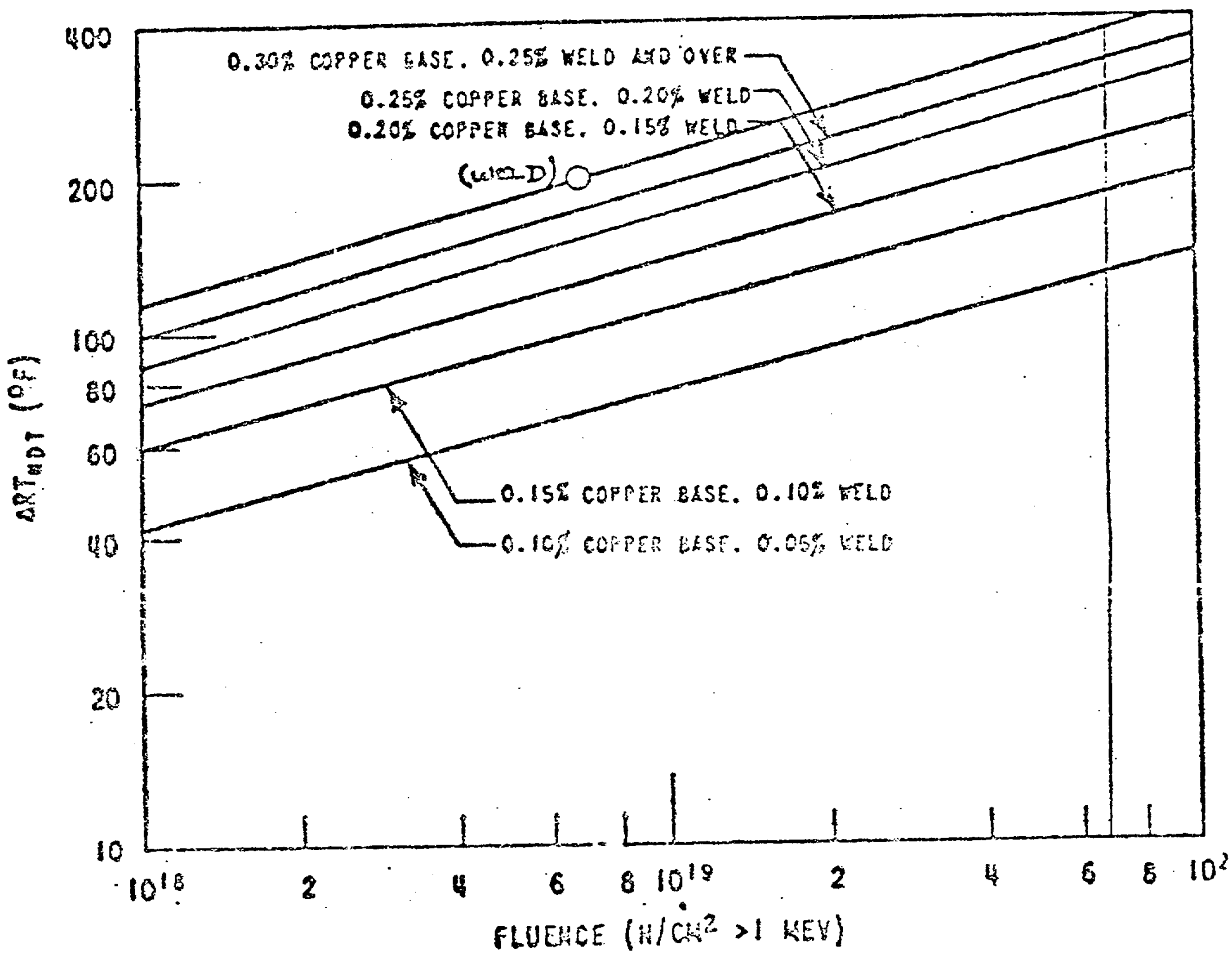
The reactor vessel materials have been tested to determine their initial RT_{NDT}. Adjusted reference temperatures, based upon the fluence and copper content of the material in question, are then determined. The heatup and cooldown limit curves include the shift in RT_{NDT} at the end of the service period shown on the heatup and cooldown curves.

The actual shift in NDTT of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, 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 has a definite relationship to the spectra at the vessel inside radius, the measured transition shift for a sample can be related with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown 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.

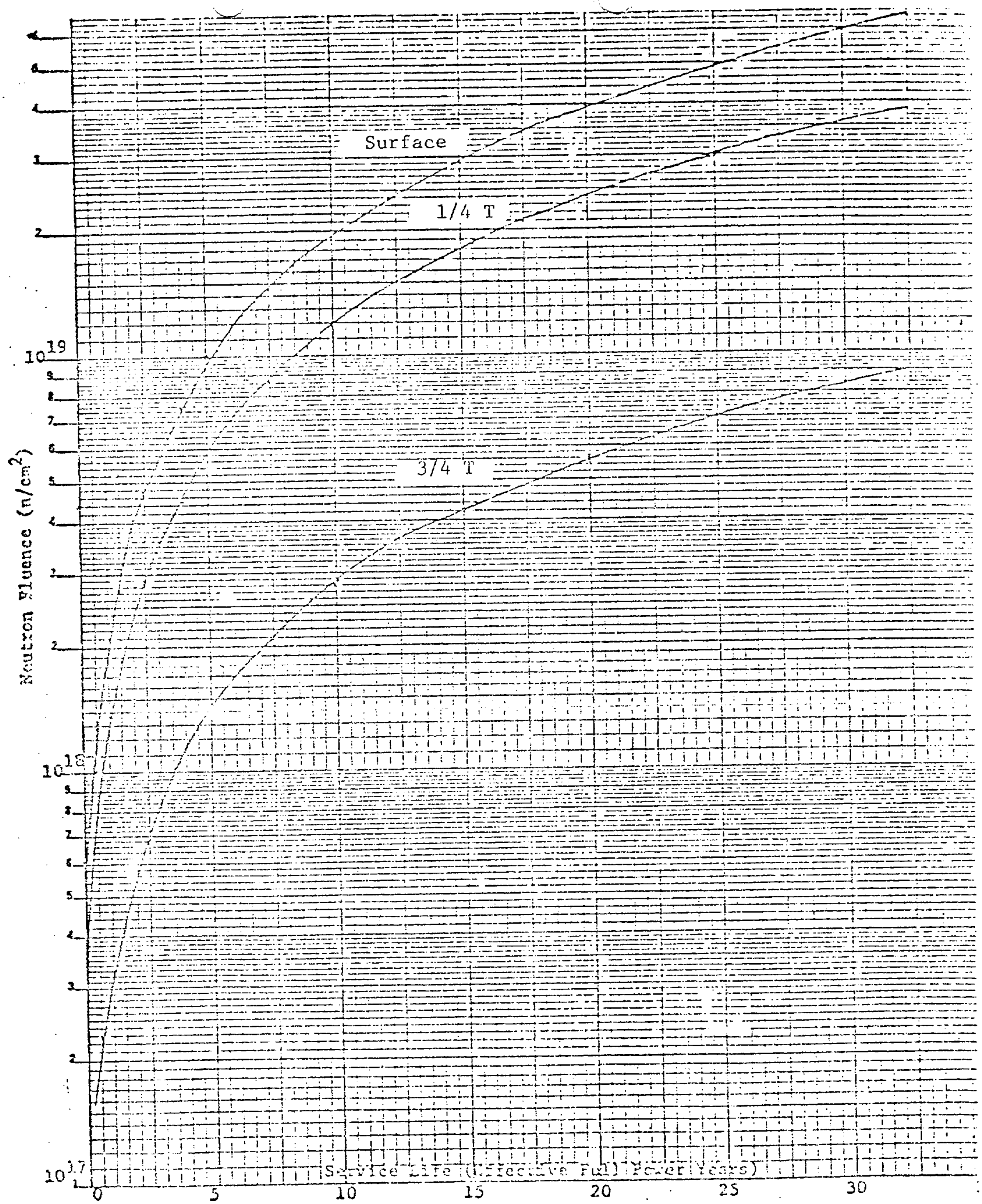
The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.2-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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



Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550 °F Temperature

Figure B3.1-1



Fast Neutron Fluence (E > 1 MEV) as a
Function of Effective Full Power Years

Figure B3.1-2

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 24 TO LICENSE NO. DPR-31 AND

AMENDMENT NO. 23 TO LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT NUCLEAR GENERATING UNITS NOS. 3 AND 4

DOCKETS NOS. 50-250 AND 50-251

Introduction

By letter dated May 21, 1976, Florida Power and Light Company (FPL) requested changes to the Technical Specifications appended to Facility Operating Licenses DPR-31 and DPR-41, for Turkey Point Nuclear Generating Units Nos. 3 and 4. The requested changes would modify the reactor coolant system pressure-temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT})[1/] in accordance with the requirements of Appendix G to 10 CFR Part 50.

Discussion

Title 10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material and the design service conditions. However, the metallurgical properties vary over the lifetime of the reactor vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature (RT_{NDT}) to increase or shift with time. The practical results of the RT_{NDT} shift is that, for any given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT}, the stresses in the reactor vessel are maintained within acceptable limits.

The magnitude of the shift in RT_{NDT} is proportional to the integrated amount of neutron irradiation experienced by the reactor vessel. The presently specified reactor vessel material surveillance program will determine the validity of the predicted increases in RT_{NDT}. Surveillance specimens are periodically removed from the reactor vessel for testing and analysis. The results of the tests and analysis are compared with the predicted shifts in RT_{NDT}; then the pressure-temperature limits are revised for future operation as required.

Evaluation

FPL has proposed revised reactor coolant system pressure-temperature limits which reflect the evaluation of the first neutron irradiation specimens examined as part of the reactor vessel material surveillance program. Separate specimens have been examined for both Unit No. 3 and Unit No. 4. Because of different neutron irradiation induced variations in the material properties of the specimens removed from each unit, FPL proposed separate pressure-temperature limits for Unit No. 3 and Unit No. 4. In addition, FPL proposed separate pressure-temperature limits applicable for reactor operating periods up to 5 effective full power years (EFPYs) and from five to ten EFPYs. The proposed limits were calculated by FPL using the methods presented in Appendix G to the ASME Code Section III, as specified in Appendix G to 10 CFR Part 50. The proposed pressure-temperature limits were determined to satisfy the heatup and cooldown limitations of the weld metal located at the reactor pressure vessel beltline, the most limiting reactor vessel material at both facilities.

The Unit No. 3 reactor vessel beltline material has a copper content of 0.31%. The initial value of RT_{NDT} was determined prior to reactor operation, from the results of drop weight and Charpy V-notch tests to be 3°F. Based on the data obtained from the specimens examined in the material surveillance program, FPL estimated the shift in RT_{NDT} for Unit No. 3 to be 191°F following 5 EFPYs of operation and 233°F following 10 EFPYs of operation.

The Unit No. 4 reactor vessel beltline material has a copper content of 0.30%. FPL determined the initial value of RT_{NDT} for this material to be 0°F. Based on the data obtained from the material surveillance program the shift in RT_{NDT} for Unit No. 4 was estimated by FPL to be 281°F following 5 EFPYs of operation and 342°F following 10 EFPYs of operation.

The greater shift in RT_{NDT} for Unit No. 4: (1) indicated that the Unit No. 4 reactor pressure vessel weld material may be more sensitive to neutron exposure and (2) resulted in more restrictive pressure-temperature limits being proposed by FPL for Unit No. 4 than for Unit No. 3. The sensitivity of the reactor vessel weld material, at both units, to increased neutron exposures will be determined by the surveillance capsules removed in the future from the reactor vessel. Until additional examinations are performed on the sensitivity of the reactor pressure vessel weld material with increased neutron exposure, we have concluded that the more restrictive pressure-temperature limits proposed for Unit No. 4 should also be applied to Unit No. 3. In addition, we are deferring approval of the proposed limits for the period between 5 to 10 EFPYs pending further analysis of the welds in the reactor beltline region.

We have reviewed: (1) the predicted shifts in RT_{NDT} following 5 EFPYs of operation and (2) the reactor coolant system pressure-temperature limits based on these shifts in RT_{NDT}. Based on our review, we have concluded that the revised pressure-temperature limits proposed for Unit No. 4: (1) have been properly determined, (2) specify conservative reactor coolant system heatup and cooldown limits which are applicable to both Units Nos. 3 and 4 and (3) conform to Appendix G of 10 CFR Part 50. Therefore, the pressure-temperature limits proposed for Unit No. 4 for operation up to 5 EFPYs are acceptable for both Units Nos. 3 and 4.

Environmental Considerations

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusions

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, and changes do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 20, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-250 AND 50-251

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 24 and 23 to Facility Operating Licenses Nos. DPR-31 and DPR-41, respectively, issued to Florida Power and Light Company which revised Technical Specifications for operation of the Turkey Point Nuclear Generating Units Nos. 3 and 4, located in Dade County, Florida. The amendments are effective as of the date of issuance.

These amendments consist of changes to the Technical Specifications which will change the reactor coolant pressure-temperature limits to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated May 21, 1976, (2) Amendments Nos. 24 and 23 to Licenses Nos. DPR-31 and DPR-41 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental & Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 20 day of April 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors