NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL
(TEMPORARY FORM)

CONTROL NO: 1017

FILE:

FROM:	DATE OF DOC	DATE REC'D	LTR	TWX	RPT	OTHER
Carolina Power & Light Co Raleigh, N.C. 27602 Mr. E.E. Utley	1-24-75	1-30-75	X			

TO:	ORIG	CC	OTHER	
E.G. Case	3 signed			SENT AEC PDR XXX SENT LOCAL PDR XXX

CLASS	UNCLASS	PROP INFO	INPUT	NO CYS REC'D	DOCKET NO:
	XXX		XXX	40	50-261

DESCRIPTION:

Ltr notarize 1-24-75...trans the following...

PLANT NAME: H.B. Robinson

ENCLOSURES:

Amdt to the OL...consist of changes to the tech specs....concerning heatup & cooldown limitations......

ACKNOWLEDGED

(40 cys encl rec'd)

DO NOT REMOVE

FOR ACTION/INFORMATION
1-30-75 JB

BUTLER (L) W/ Copies
CLARK (L) W/ Copies
PARR (L) W/ Copies
KNIEL (L) W/ Copies
SCHWENCER (L) W/ Copies
STOLZ (L) W/ Copies
VASSALLO (L) W/ Copies
PURPLE (L) W/ Copies
ZIEMANN (L) W/ Copies
DICKER (E) W/ Copies
KNIGHTON (E) W/ Copies
YOUNGBLOOD (E) W/ Copies
REGAN (E) W/ Copies
✓ LEAR (L) W/ 6 Copies
SPELS W/ Copies
W/ Copies

INTERNAL DISTRIBUTION

✓ REG FILE
✓ ~~NRC PDR~~
✓ OGC, ROOM P-506A
✓ GOSSICK/STAFF
CASE
GIAMBUSSO
BOYD
MOORE (L)
DEYOUNG (L)
SKOVHOLT (L)
✓ GOLLER (L) (Ltr)
P. COLLINS
DENISE
✓ REG OPR
FILE & REGION (2)
T.R. WILSON
STEELE

TECH REVIEW
SCHROEDER
MACCARY
KNIGHT
PAWLICKI
SHAO
STELLO
HOUSTON
NOVAK
ROSS
IPPOLITO
TEDESCO
LONG
LAINAS
BENAROYA
VOLLMER

DENTON
GRIMES
GAMMILL
KASTNER
BALLARD
SPANGLER

ENVIRO
MULLER
DICKER
KNIGHTON
YOUNGBLOOD
REGAN
✓ PROJECT LDR Dittman
HARLESS

LIC ASST
R. DIGGS (L)
H. GEARIN (L)
E. GOULBOURNE (L)
P. KREUTZER (E)
J. LEE (L)
M. MAIGRET (L)
S. REED (E)
M. SERVICE (L)
S. SHEPPARD (L)
M. SLATER (E)
H. SMITH (L)
✓ S. TEETS (L)
G. WILLIAMS (E)
V. WILSON (L)
R. INGRAM (L)

A/T IND
BRAITMAN
SALTZMAN
MELTZ

PLANS
MCDONALD
CHAPMAN
✓ DUBE (Ltr)
✓ E. COUPE
PETERSON
HARTFIELD (2)
KLECKER
EISENHUT
WIGGINTON

EXTERNAL DISTRIBUTION

Appl R (2)

✓ 1 – LOCAL PDR Hartsville, S.C.
✓ 1 – TIC (ABERNATHY)
✓ 1 – NSIC (BUCHANAN)
1 – ASLB
1 – Newton Anderson
✓ 16 – ACRS ~~HOLDING~~/SENT

(1)(2)(10) – NATIONAL LABS
1 – W. PENNINGTON, Rm E-201 GT
1 – CONSULTANTS NEWMARK/BLUME/AGBABIAN

1 – PDR-SAN/LA/NY
1 – BROOKHAVEN NAT LAB
1 – G. ULRIKSON, ORNL
1 – AGMED (RUTH GUSSMAN) Rm B-127 GT
1 – J. D. RUNKLES, Rm E-201 GT

CP&L
Carolina Power & Light Company

Regulatory Docket File

January 24, 1975

50-261

DOCKETED USAEC JAN 1975 REGULATORY MAIL SECTION DOCKET CLERK

RECEIVED JAN 30 1975 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

FILE: NG-3514 (R)

SERIAL: NG-75-030

Mr. Edson G. Case, Acting Director
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Case:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
REQUEST FOR LICENSE AMENDMENT
REVISION OF TECHNICAL SPECIFICATIONS

In accordance with the Code of Federal Regulations, Title 10, Part 50.59, Carolina Power & Light Company submits a proposed revision to the Technical Specifications for its H. B. Robinson Unit No. 2 plant. The revision provides heatup and cooldown limitations for the reactor coolant system based on the Summer 1972 Addenda to the ASME Boiler and Pressure Vessel Code, Section III, and results of the reactor vessel surveillance capsule removed during the 1973 refueling outage.

Following removal of Surveillance Capsule S from the reactor vessel during the refueling outage mentioned above, Carolina Power & Light Company submitted revisions to the present Technical Specifications based on materials testing in accord with applicable industry codes and Commission regulations in effect at the time of initial licensing of the plant. These revisions were filed on September 19, 1973. On October 16, 1973, the Commission responded to this request, asking for additional information to demonstrate that the requirements of Appendices G and H of 10 CFR50 would be satisfied by the proposed revisions to the Technical Specifications, and that the proposed heatup and cooldown curves were at least as conservative as those that would be derived following the guidance of the Summer 1972 Addenda to the ASME B&PV code. This information was provided by CP&L to the Commission on December 20, 1973 and included new heatup and cooldown curves based on application of the requested analyses and methodologies. No formal submittal of revised Technical Specifications was made at that time so following discussions with your staff, we are herewith providing the proper revisions such that this

1017

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

matter may be closed and the specifications may reflect the applicable restrictions on Reactor Coolant System heatup and cooldown. The attached revision supercedes all previously requested and outstanding revisions to the subject Technical Specifications and Bases.

As required by Commission Regulations, this submittal is signed under oath by a duly authorized Officer of the Company.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

DBW:cpw

CC: Mr. N. B. Bessac
Mr. T. E. Bowman
Mr. J. B. McGirt
Mr. W. E. Graham
Mr. D. V. Menscer
Mr. D. B. Waters

Sworn to and subscribed before me this 24th day of January, 1975.

Nancy A. Stephens (Yancey)

My Commission expires: June 29, 1976

NANCY A. STEPHENS NOTARY PUBLIC WAKE COUNTY, N. C.

3.1.2 HEATUP AND COOLDOWN

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2, and are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed $60^{o}F/hr$.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed $100^{o}F/hr$. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary System Hydrostatic leak tests may be performed as necessary, provided the temperature-pressure limitations as noted on Figure 3.1-1 are not exceeded. Maximum hydrostatic test pressure should remain below 2350 psia.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below $70^{o}F$.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed $200^{o}F/hr$. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than $320^{o}F$.

3.1.2.4 Figures 3.1-1, 3.1-2, and 3.1-3 shall be updated periodically in accordance with the following criteria and procedures, before the calculated exposure of the vessel exceeds the exposure for which the figures apply.

a. Figure 3.1-3, which shows the effects of neutron fluence and copper content on the shift of RT_{NDT} due to exposure at 550°F, shall be used to predict the increase in the reference nil-ductility temperature based on the expected vessel exposure. If measurements on the irradiation specimens show increases above and to the left of the predicted curve, a new curve shall be constructed such that it encompasses the applicable data points.

b. At or before the end of the integrated power period for which Figures 3.1-1 and 3.1-2 apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Summer 1972 Addenda, Non-Mandatory Appendix G. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes in Figure 3.1-3 resulting from the irradiation specimen measurement program.

c. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

Basis:

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility under certain conditions of irradiation. In pressure vessel material, the most serious mechanical property change is the reduction in the upper shelf impact strength. Accompanying the decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperatue $60^{o}F$ less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program[1] where a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. The increase in the Charpy V-notch 50 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + RT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods(2) derived from Non-Mandatory Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies explicit safety factors of 2.0 and 1.25* on stress intensity factors induced by pressure and thermal gradients, respectively. Thus, the governing equation for the heatup-cooldown analysis is:

$$2\,K_{IM} + 1.25\,K_{It} \leq K_{IR} \qquad (1)$$

where:

K_{IM} is the pressure intensity factor caused by the thermal (pressure) stresses.

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

K_{IR} is the reference stress intensity factor provided by the code as a function of temperature relative to the RT_{NDT} of the material.

During the heatup analysis, Equation (1) is evaluated for two distinct situations.

---

* The 1.25 safety factor on K_{It} represents additional conservatism above Code Requirements.

First, allowable pressure-temperature relationships are developed for steady state (i.e., zero rate of change of temperature) conditions assuming the presence of the code reference 1/4T deep flaw at the ID of the pressure vessel. Due to the fact that, during heatup, the thermal gradients in the vessel wall tend to produce compressive stresses at the 1/4T location, the tensile stresses induced by internal pressure are somewhat alleviated. Thus, 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 1/4T location is considered. The second portion of the heatup analysis concerns the calculation of pressure temperature limitations for the case in which the 3/4T location becomes the controlling factor. Unlike the situation at the 1/4T locations, at the 3/4T position (i.e., the tip of the 1/4T deep O.D. flaw) the thermal gradients established during heatup produce stresses which are tensile in nature; and, thus, tend to reinforce the pressure stresses present. These thermal stresses are, of course, dependent on both the rate of heatup and the time (or water temperature) along the heatup ramp. Furthermore, since the thermal stresses at 3/4T are tensile and increase with increasing heatup rate, a lower bound curve similar to that described in the preceeding paragraph cannot be defined. Rather, each rate of interest 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 in the following fashion. First, a composite curve is constructed 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 two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The use of the composite curve in the cooldown analysis is necessary because system control is based on a measurement of reactor coolant temperature, whereas the limiting pressure is calculated using the material temperature at the tip of the assumed 1/4T reference flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the coolant which is at the indicated temperature. This condition is, of course, not true for the steady-state situation. It follows that the ΔT induced during cooldown results in a calculated higher K_{IR} which is less limiting at a given indicated temperature for finite cooldown rates than for steady state under certain conditions. Hydrostatic (leak) test temperatures are defined by ASME Code Appendix G. For H. B. Robinson Unit No. 2 which has a reactor vessel shell thickness of 9.3125 inches and a vessel inner radius of 77.75 inches, a hydrostatic test at $\sim$2350 psi produces membrane stresses of 19,619 psi. Since bending and secondary stresses due to thermal gradients are negligible during hydrostatic test conditions, the governing equation becomes:

$$1.5\ K_{Im} < K_{IR}$$

Using the methods of ASME code Appendix G for determining stress intensity factors for the 1/4T assumed flaw, K_{Im} is 58.13 Ksi $\sqrt{in}$ and thus, 1.5 K_{Im} is 87.2 Ksi $\sqrt{in}$. In order for K_{IR} to be 87.2 Ksi $\sqrt{in}$. or greater, a temperature of RT_{NDT} + 109°F must be attained as determined from the K_{IR} curve of ASME Code Appendix G. This results in the limit shown on Figure 3.1-1 for the applicable integrated power period.

<u>References</u>:

1. S. E. Yanichko, "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program", Westinghouse Nuclear Energy Systems - WCAP-7373 (January, 1970).
2. S. E. Yanichko etal, "Analysis of Capsule S from Carolina Power & Light Company, H. B. Robinson Unit No. 2, Reactor Vessel Radiation Surveillance Program", Westinghouse Nuclear Energy Systems - FP-RA-2 (December 18, 1973).

3.1.3 MINIMUM CONDITIONS FOR CRITICALITY

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is positive.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is positive, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis:

During the early part of the initial fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below the power operating range.(1) (2) The moderator coefficient at low temperatures will be most positive at the beginning of life of the initial fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the initial fuel cycle and during subsequent reload fuel cycles, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative in the power operating range.(1) (2) The maximum teperature at which the moderator coefficient is positive, at the beginning of life of the initial fuel cycle, with all control rods withdrawn, will be determined during preoperational physics tests. When control rods are inserted, the temperature at which the moderator coefficient becomes negative is lower so that at the temperature determined during the physics tests with the operational control rod program, the temperature coefficient is expected to be negative. The requirement that the reactor is not to be made critical when the moderator coefficient is positive has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50 Appendix B paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods, are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as calculated by ASME Code Appendix G methods, and an additional safety margin of $40^{\circ}F$ must be maintained above the applicable heatup curve at all times.

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-9

(3) FSAR Figure 3.2.1-10

3.1.4 MAXIMUM REACTOR COOLANT ACTIVITY

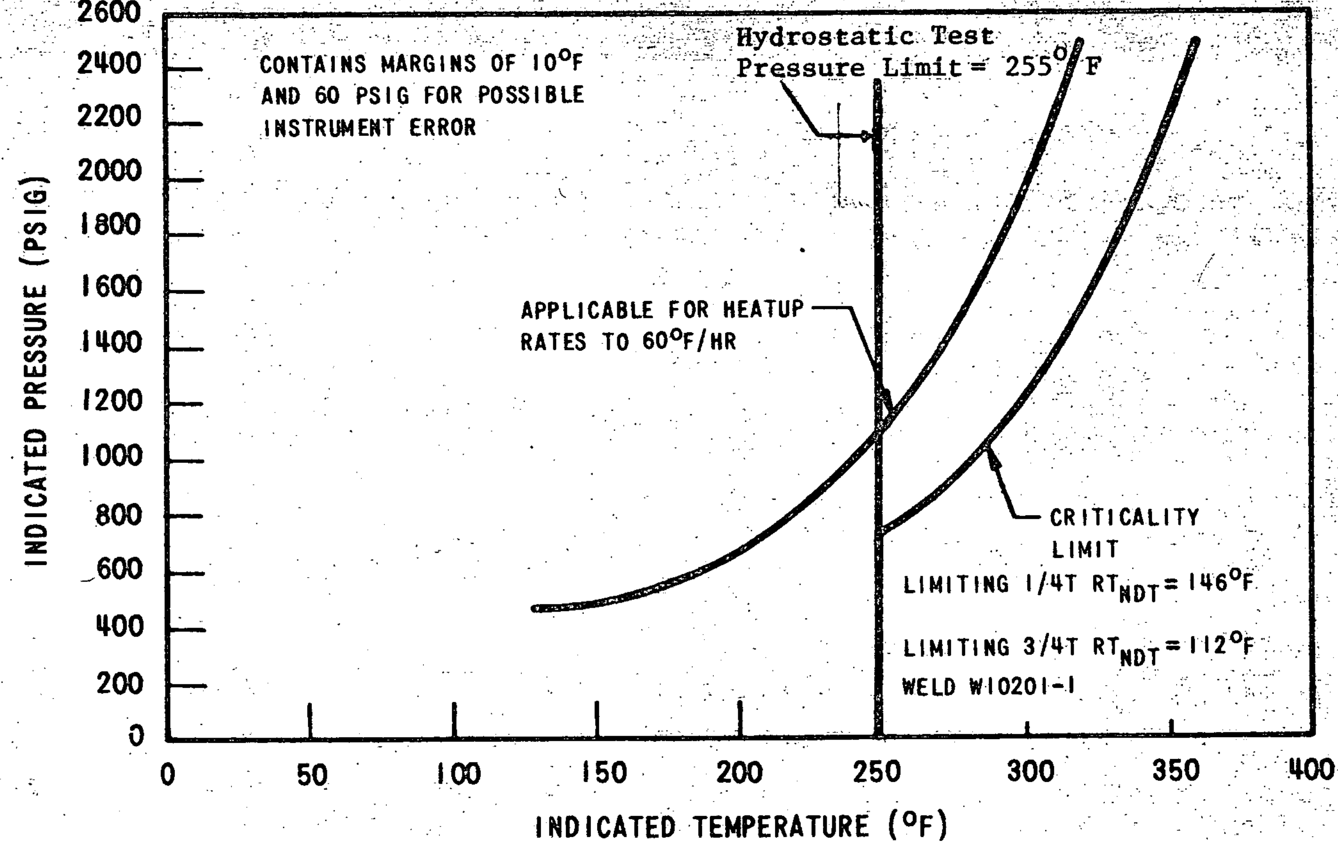
Specification:

The total specific activity in μCi/cc of the reactor coolant due to nuclides with half-lives of more than 30 minutes shall not exceed the number equivalent to $50/\bar{E}$ whenever the reactor is critical or the average reactor coolant temperature is greater than $500^{\circ}F$. ($\bar{E}$ is the average of beta and gamma energy (Mev) per disintegration of the specific activity).

Basis

The above specification is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is per the allowable limit. The potential release of activity to the atmosphere has been evaluated to assure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the Secondary System. The major portion of this activity entering the Secondary System is noble gasses[2] and could be released to the atmosphere from the air ejector or a relief valve. Activity release could continue until the operator could reduce the primary system pressure below the set point of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam



H. B. Robinson Unit No. 2 Reactor Coolant System Heatup Limitations Applicable For the First 4.25 Effective Full Power Years

Figure 3.1-1

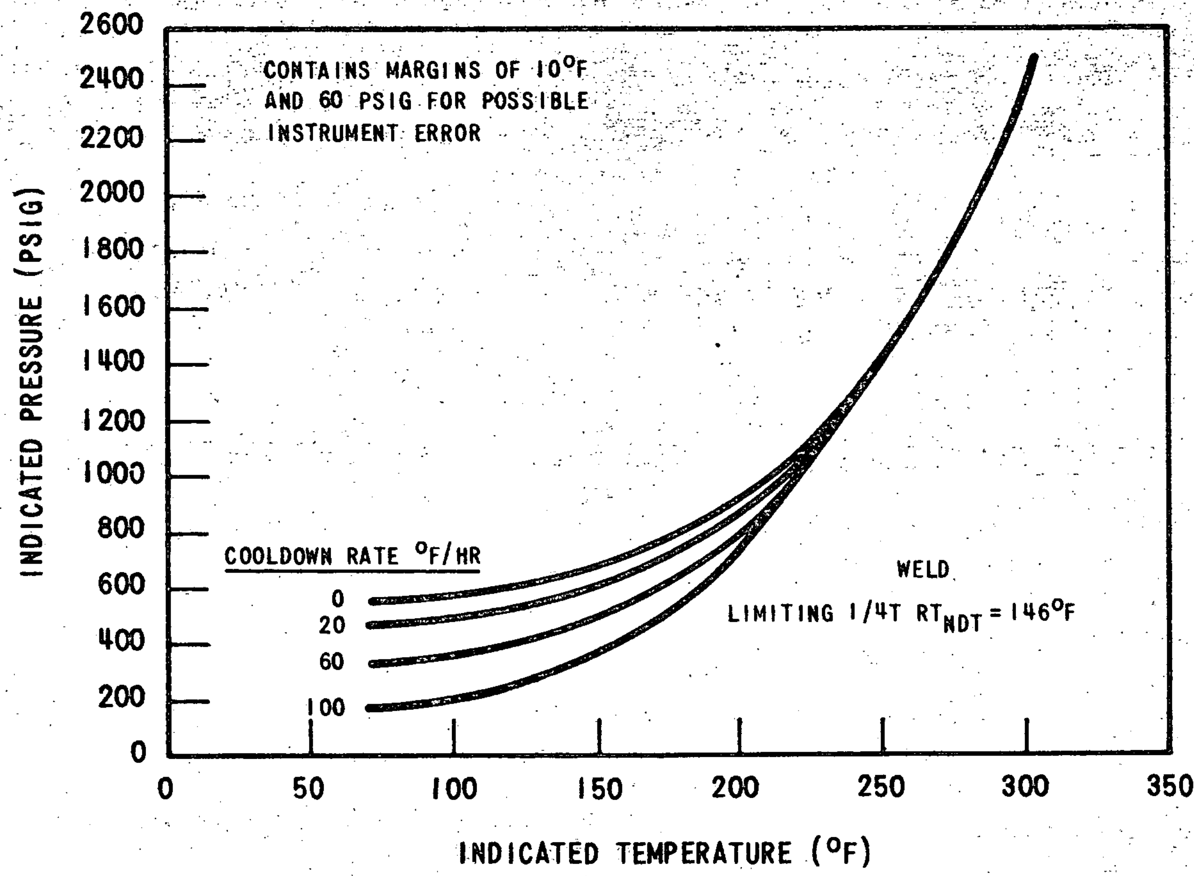


Figure 3.1-2

H. B. Robinson Unit No. 2 Reactor Coolant System Cooldown Limitations Applicable for the First 4.25 Effective Full Power Years

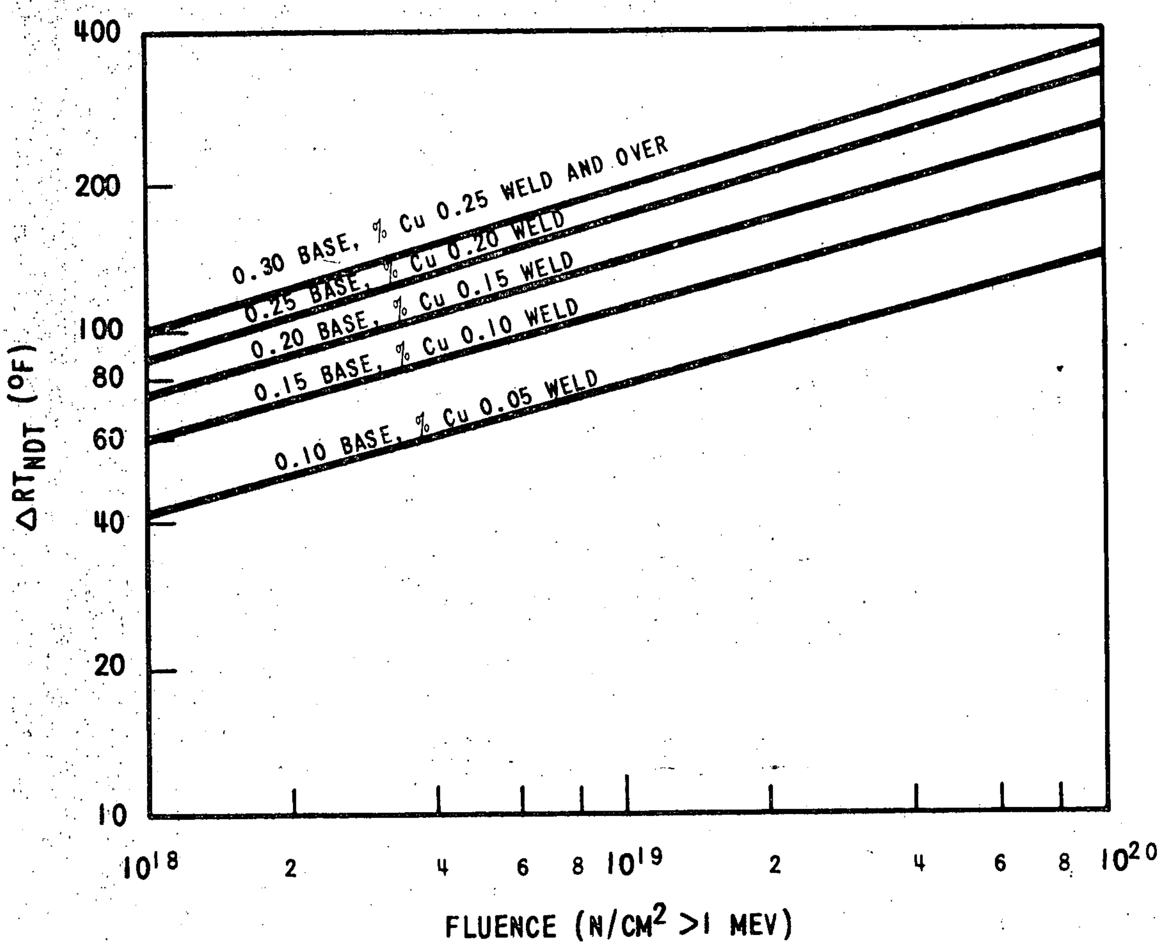


Figure 3.1-3

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