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DESCRIPTION:
Ltr furnishing info re Irradiation Surveillance Program & submitting revisions to the Tech Specs......W/Attached Figure 1, Figure 3.1-1 & Figure 3.1-2.......

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PLANT NAME: H. B. Robinson Unit # 2

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Carolina Power & Light Company

September 19, 1973

File: NG-3514 Serial: NG-73-410

Mr. Robert J. Schemel, Chief
Operating Reactors Branch #1
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

50-261

RECEIVED SEP 27 1973 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Dear Mr. Schemel:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
IRRADIATION SURVEILLANCE PROGRAM

As described in Paragraph 3.1.2 of the H. B. Robinson Technical Specifications, an irradiation surveillance program is utilized to determine the nil ductility transition temperature (NDTT) shift of reactor vessel material that results from irradiation during plant operation. This information is then used in determining heatup and cooldown rates for the system. Such a program is in compliance with requirements of Title 10, Part 50, Appendix H, of the Code of Federal Regulation.

During the 1973 spring refueling outage the first H. B. Robinson Unit 2 irradiated specimen was removed for analysis. The study of the individual specimen was conducted by Southwest Research Institute in San Antonio, Texas. The following is a summary of the results of these tests. Subsequent revision of the plant Technical Specifications are required as a result of the information obtained.

Irradiated Specimen Analysis

The specimen that was analyzed was located at the 280° vessel peripheral position and was designated as capsule S. The capsule contained the following items:

1. Tensile test specimens
2. Charpy test specimens
3. Dosimeters
4. Thermal Monitors
5. Wedge Opening Loading Specimens

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The results of each surveillance is described below. (See Figure 1 for capsule location.)

Tensile Test

The tensile specimen tests indicated that the tensile properties of the material have not significantly changed from the base line data. The yield strength, ultimate tensile strength, elongation, and reduction in area for the three vessel plate samples all agree with the original specifications for the SA302, Gr.A vessel material.

Charpy Impact Test

The data obtained from the samples indicates that the radiation induced shift in transition temperature at the 30 ft-lb "fix" energy level is below the projected values of the Technical Specifications Figure 3.1-3. The three core belt line plates experienced shifts of 20^{o}F and 30^{o}F as a result of exposure. This confirms the conservative nature of the Technical Specification curve and indicates it does not require revision. Data also revealed that there was very little degradation of the upper shelf charpy V-notch properties at this time.

Dosimetry

The dosimeter analysis resulted in a calculated fast neutron fluence of 1.52×10^{18} neutrons/cm^2 > 1 MeV received by the vessel wall in the 1.33 full power years. This is approximately 75 percent of the original prediction. Based on this figure, the 5 and 40 year projected fluences at 80 percent availability will be 4.57×10^{18} neutrons/cm^2 > 1 MeV and 3.7×10^{19} neutrons/cm^2 > 1 MeV respectively.

Thermal Monitoring

The three eutechtic alloy thermal monitors possessed melting points of 579^{o}F and 590^{o}F. None of the monitors melted, thus indicating that the sampling maximum temperature was less than 579^{o}F.

Wedge Opening Loading Test

Current technology limits the testing of the fracture mechanic samples to temperatures well below the minimum service temperature. Tests were run on the samples, but no significant data was obtained. It is anticipated that continued efforts will lead to advancing the technology to the point that valid information may be obtained.

Revisions Required to Technical Specifications

As a result of the data obtained from the capsule, various changes are required to update the plant Technical Specifications to reflect the projected neutron fluence and NDTT shift for the forthcoming surveillance interval.

The next specimen is to be removed after the fifth year of operation. This corresponds to 3.35×10^6 thermal megawatt days ($\sim$1523 EFPD). The predicted neutron fast flux for this period is 4.57×10^{18} neutrons/cm^2 > 1 MeV. Using this value in conjunction with Figure 3.1-3 of the Technical Specifications yields an NDTT shift of 117°F. This figure is then used to arrive at a design transition temperature (DTT) to establish revised limit lines for Figures 3.1-1 and 3.1-2 of the Technical Specifications. The subject figures relate to system heatup and cooldown rates. The DTT obtained in this manner is 208°F and DTT +10°F is 218°F. The attached Figures 3.1-1 and 3.1-2 have been revised to reflect this shift. Also attached are revised pages to Section 3.1.2 of the Technical Specifications which incorporate the preceding changes.

In addition to the above change, several plant procedures will require revision to reflect the new temperature limitations. Procedures to be changed are as follows:

1. MI-4a, Reactor Coolant System Cold Leak Test
2. OP-25, Reactor Coolant System Operations
3. OP-30, Pressurizer Pressure and Spray Control
4. GP-1, Overall Plant Operating Procedure

These changes will be made contingent upon your approval of the Technical Specifications. The present limitations are valid for 610 full power days. Through July 31 the plant has been operated for 525.5 EFPD.

Your prompt consideration of this proposal is requested. It is estimated that the 610 EFPD figure will be achieved near the end of October, 1973.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

JGH:mvp

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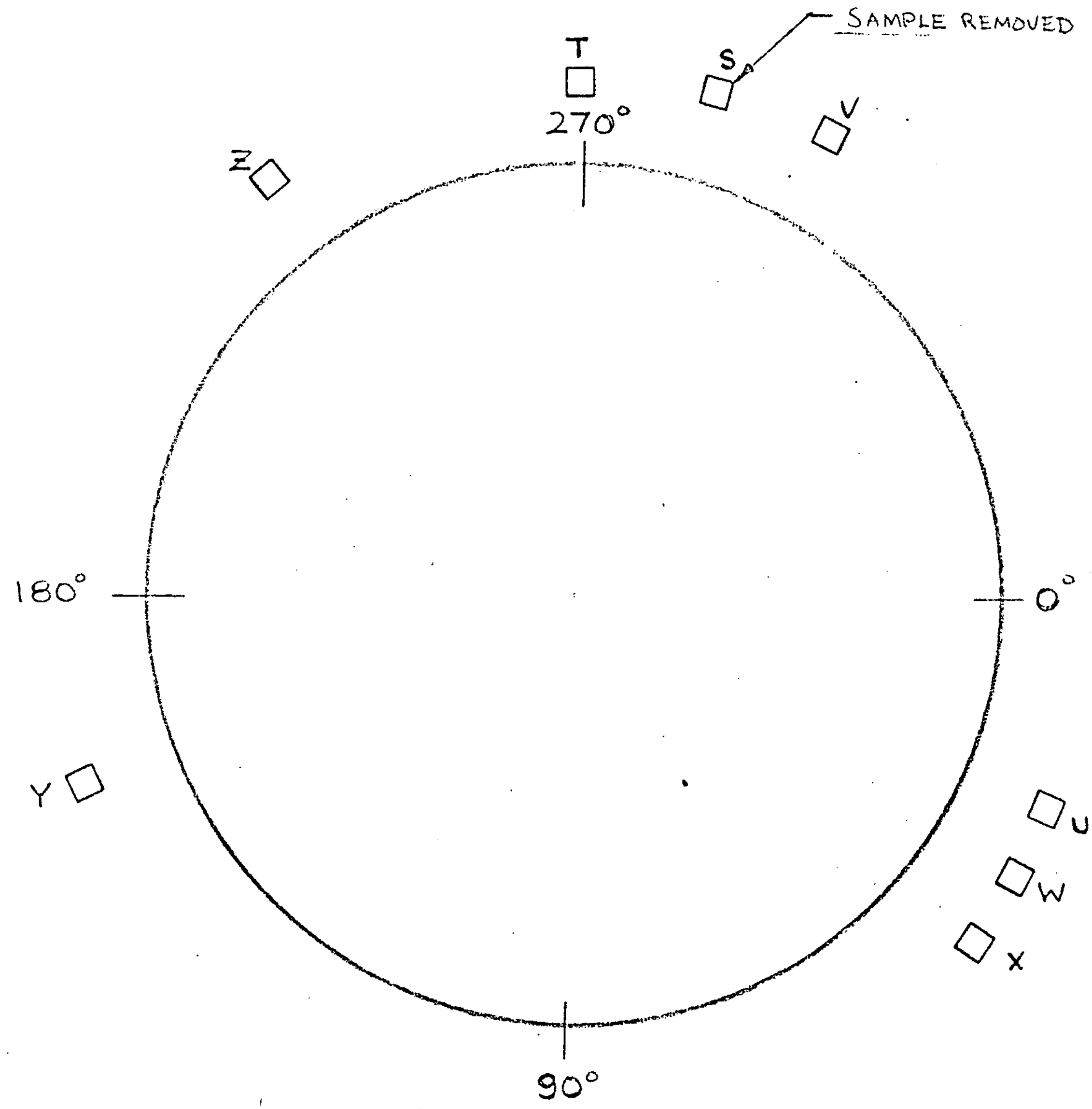

SAMPLE LOCATIONS

FIGURE 1

REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS

(Applicable to 3.35×10^6 MWt - Days)

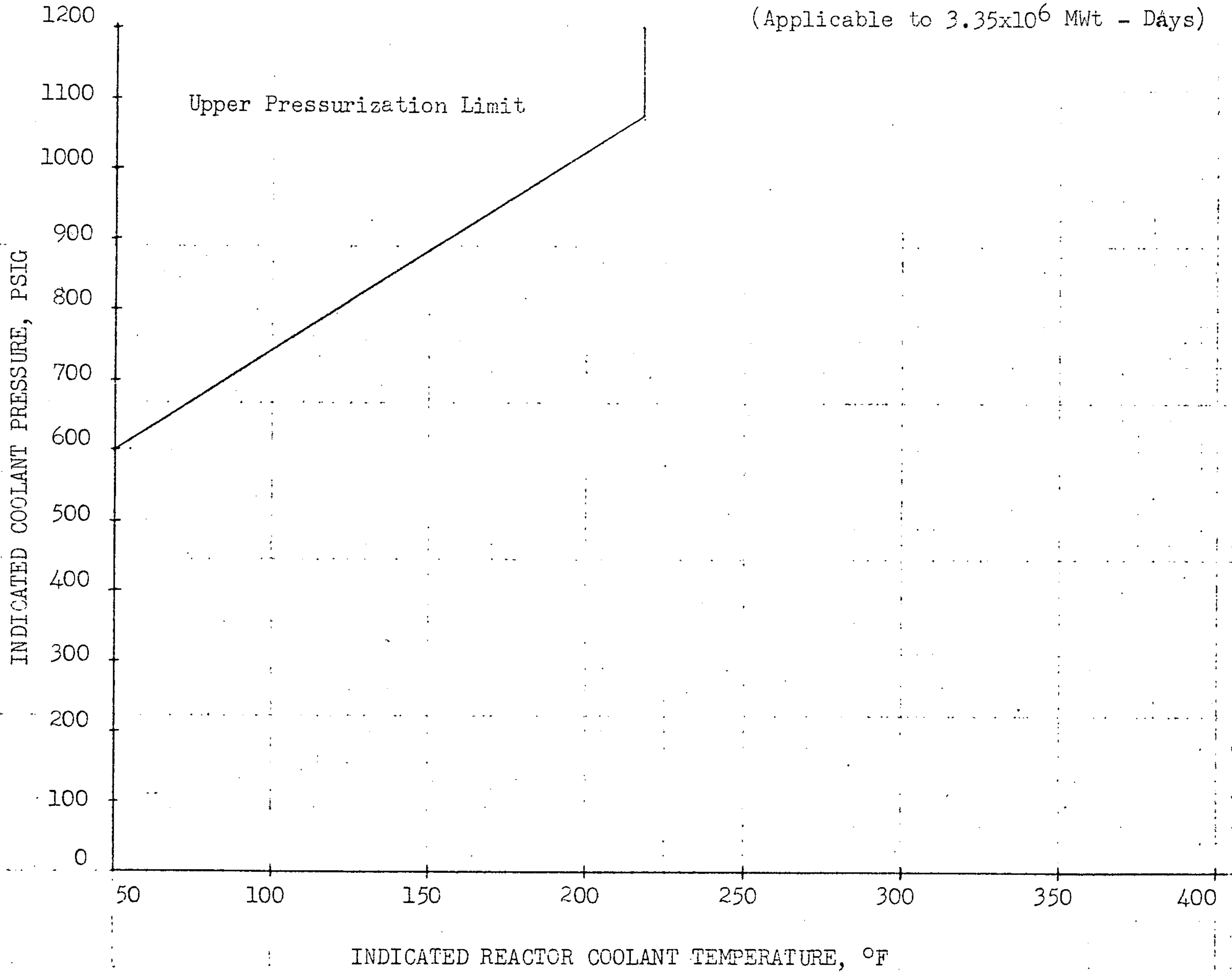

FIGURE 3.1-1

3.1-19

REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS

(Applicable to 3.35×10^6 MWt - Days)

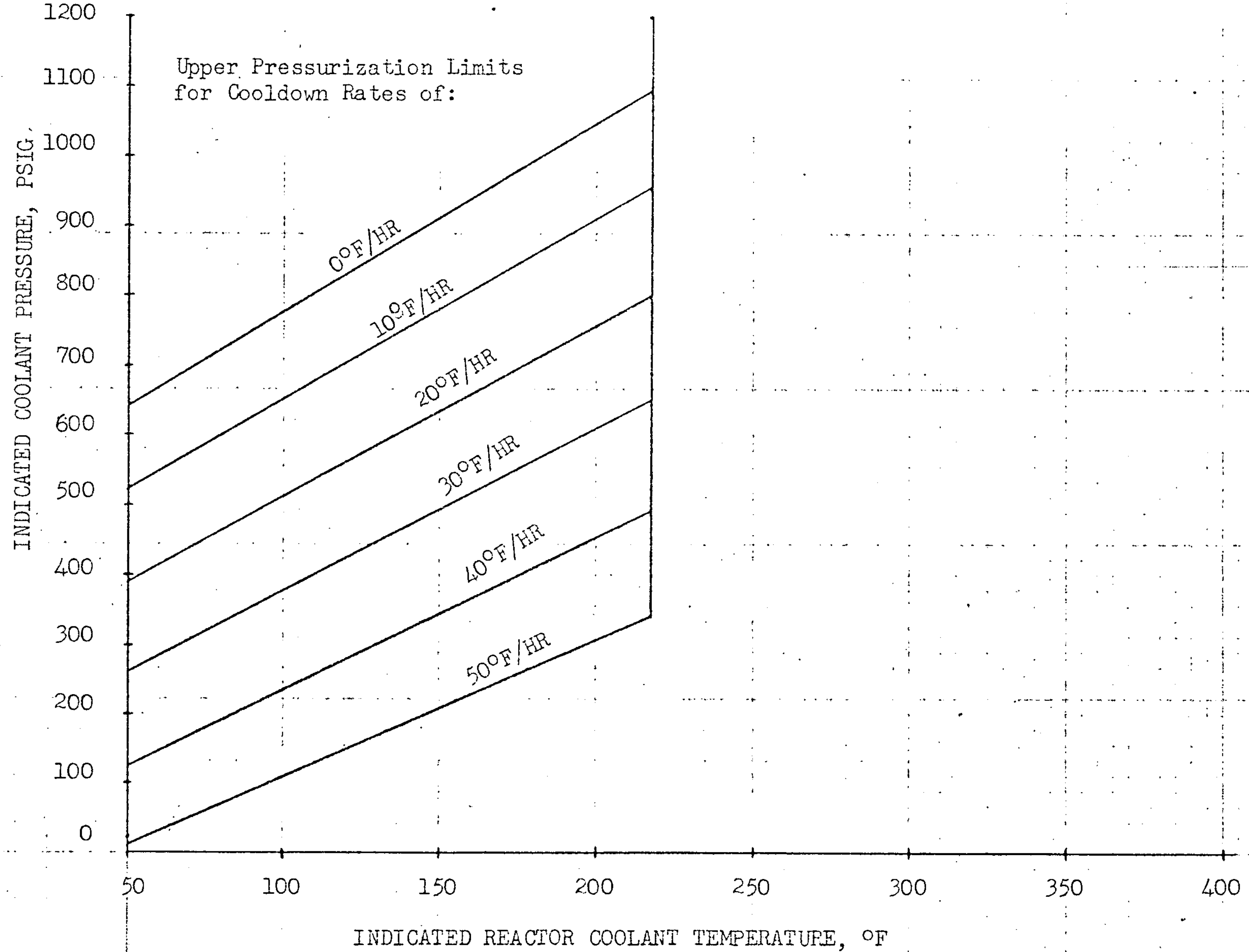

FIGURE 3.1-2

a. The "max. for 550°F" curve in Figure 3.1-3 shall be used to predict the increase in transition temperature based on integrated power unless measurements on the irradiation specimens show increases above and to the left of the predicted curve, in which case a new curve having the same slope as the original shall be constructed such that it is above and to the left of all 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 as follows. The total integrated reactor thermal power from startup to the end of the new period shall be converted to an equivalent integrated neutron exposure. For this plant, 3.35×10^6 thermal megawatt days ($\sim$1523 days at full power) of operation is equivalent to 4.57×10^{18} neutrons/cm^2 > 1 Mev. The predicted transition temperature increase for the end of the new period shall then be obtained from Figure 3.1-3 as revised by 3.1.2.3a above.

The limit lines in Figures 3.1-1 and 3.1-2 shall be moved parallel to the temperature axis (horizontally) in the direction of increasing temperature a distance equivalent to the transition temperature increase obtained from Figure 3.1-3 as revised less the increment used for the end of the present period. (For the end of the first 3.35×10^6 thermal megawatt days of operation, the predicted transition temperature increase is 117°F). This will result in the constant temperature (vertical) portion of the limit lines being at the projected DTT + 10°F. The sloping portions of the limit lines shall extend at constant slope to a temperature 190°F below DTT. At still lower temperatures, the limit lines shall be parallel to the temperature axis (horizontal) and shall intersect the sloping portions of the limit lines at DTT - 190°F.

The maximum integrated fast neutron (E > 1 Mev) exposure of the vessel is computed to be 5.1 x 10^{19} n/cm^2 for 40 years operation at 80 percent load factor.(3) The predicted NDTT shift for an integrated fast neutron (E > 1 Mev) exposure of 5.1 x 10^{19} n/cm^2 is 303^{o}F, the value obtained from the curve shown in Figure 3.1-3 for 550^{o}F irradiation.(3)

The actual shift in NDTT will be established periodically during plant operation by testing of vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for any increase in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the stress limits, during heatup and cooldown.

During the first five years of reactor operation, a conservatively high estimate of the energy output is 3.35 x 10^6 thermal megawatt days, which is equivalent to 1523 days at 2200 MWt. As a result of the tests performed on the irradiated specimen(6) removed during the spring 1973 refueling outage, the fast neutron exposure of the vessel for this five-year interval of operation is projected to be 4.57 x 10^{18} n/cm^2. The corresponding NDTT shift is 117^{o}F, based on the curve shown in Figure 3.1-3 for 550^{o}F irradiation. Thus, for this interval, the upper limit to the NDTT is 148^{o}F. The corresponding Design Transition Temperature, defined as NDTT + 60^{o}F, (4) is 208^{o}F.

The stress allowed in the vessel in relation to operation below NDTT and DTT (NDTT + 60) to preclude the possibility of brittle failure are:

1. At DTT; a maximum stress of 20% yield
2. From DTT to DTT minus 200^{o}F; a maximum stress decreasing from 20% to 10% yield
3. Below DTT minus 200^{o}F; a maximum stress of 10% yield

These limits are based on the data reported by Kibara and Masubichi (Effect of Residual Stress on Brittle Fracture, April 1959, Welding Journal Volume 38) and Robertson (Propagation from Brittle Fracture in Steel, Journal of the Iron and Steel Institute, 1953), which show that if the stresses are maintained within the above limits, brittle fracture does not occur.[5] The limit lines in Figures 3.1-1 and 3.1-2 are based on these stress limits and contain allowances for a $10^{\circ}F$ margin between actual and measured temperature and 60 psi margin between actual and measured pressure.

During cooldown, the thermal stress varies from tensile at the inner wall to compressive at the outer wall. The internal pressure superimposes a tensile stress on this thermal stress pattern, increasing the stress at the inside wall and relieving the stress at the outside wall. Therefore the limiting stress always appears at the inside wall, so the limit line has a direct dependence on cooldown rate. This leads to a family of curves for cooldown, as shown on Figure 3.1-2.

For heatup, the thermal stress is reversed and the location of the limiting stress is a function of the heatup rate. The limit lines no longer bear the simple relationship to heatup as they do to cooldown rate. The single line shown on Figure 3.1-1 bounds all limit lines for heatup rates up to $40^{\circ}F$ per hour for temperatures at or below $140^{\circ}F$, and $100^{\circ}F$ per hour above $140^{\circ}F$.

Further operation of the reactor beyond 3.35×10^{6} MWt days will result in further shift of the NDTT. The basis for revision of the curves will be as noted in Specification 3.1.2.3.

The basis for revising these figures, is in accordance with the stress limits defined above and inclusion of a margin for a $10^{\circ}F$ error in measured temperature, and a 60 psi error in measured pressure.

Figures 3.1-1 and 3.1-2 define stress limitations only. For normal operation other inherent plant characteristics, e.g., pump parameter and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure ranges.

The heatup and cooldown rate of 100°F per hour for the steam generator is consistent with the remainder of the Reactor Coolant System, as discussed in the first paragraph of the Basis. The stresses are within acceptable limits for the anticipated usage. The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

Temperature requirements for steam generator correspond with the measured NDT for the shell.

References:

(1) FSAR, Section 4.1.5

(2) ASME Boiler and Pressure Vessel Code, Section III, N-415

(3) FSAR, Section 4.2.5

(4) ASME Boiler and Pressure Vessel Code, Section III, N-331

(5) FSAR, Section 4.3.1

(6) Letter, Carolina Power & Light Company to DRL, September 19, 1973