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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 62
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated September 5, 1980, 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-28 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 62 , 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

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 14, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 62

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-331

1. Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

| Remove | Insert |
|---|---|
| 111 | 111 |
| 117 | 117 |
| 118 | 118 |

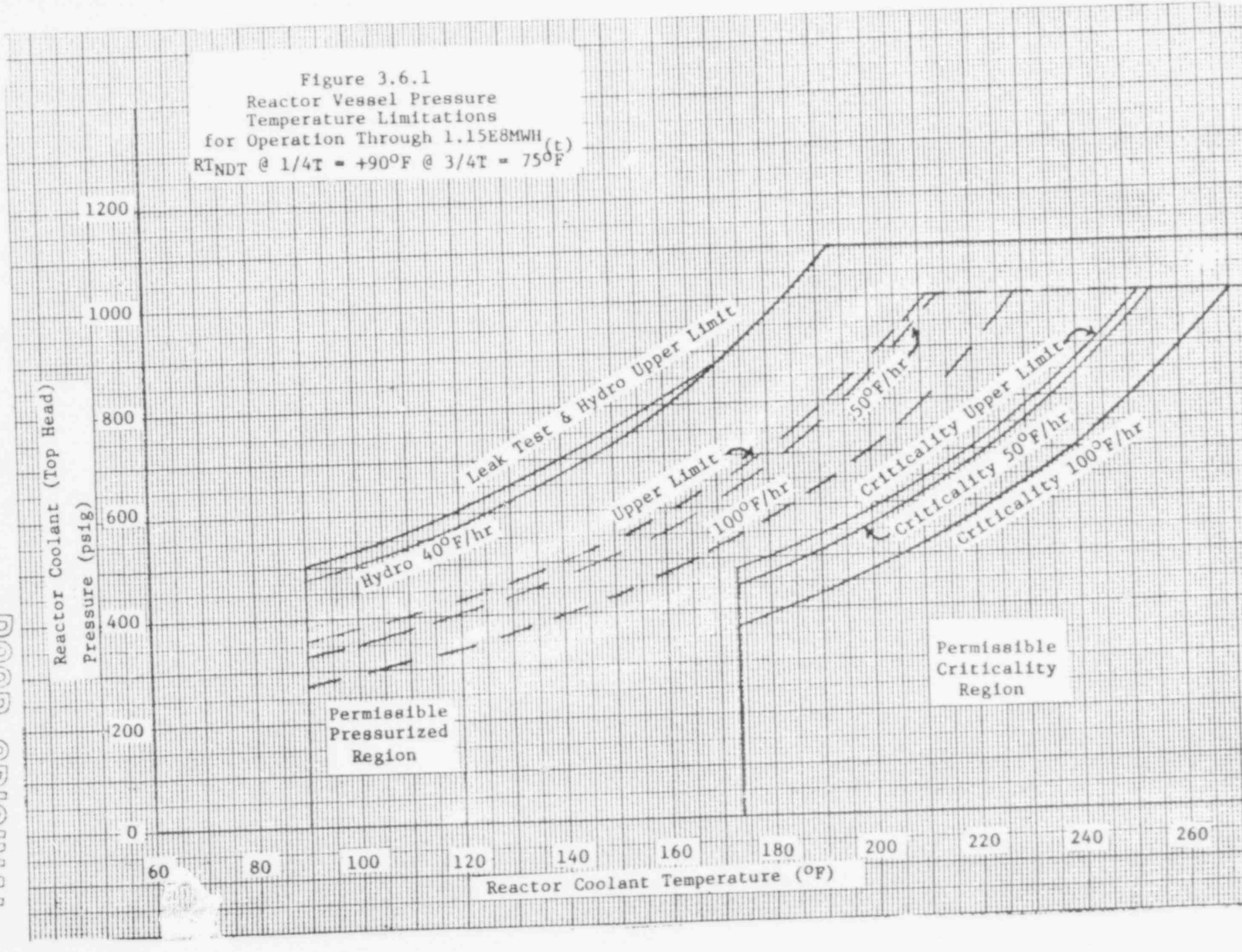


Figure 3.6.1
Reactor Vessel Pressure
Temperature Limitations
for Operation Through 1.15E8MWH$_{(t)}$
$RT_{NDT}$ @ 1/4T = +90°F @ 3/4T = 75°F

Bases:

## 3.6 & 4.6 REACTOR COOLANT SYSTEM

### A. Pressure and Temperature Limitations

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 4.2 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 cyclic operation.

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 locations.

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.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within $50^{o}F$ of each other prior to startup of an idle loop.

The reactor vessel materials have been tested to determine their initial nil-ductility transition temperature (NDTT) of $40^{o}F$ maximum. An additional margin of $20^{o}F$ has been added in order to estimate reference temperature, $RT_{NDT}$. Reactor operation and resultant fast neutron ($E > 1$ Mev) irradiation will cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature can be predicted using current industry practices (GE SIL No. 14, Supplement No. 1) based on recent GE surveillance data. The pressure/temperature limit curve Figure 3.6.1 includes predicted adjustments for this shift in $RT_{NDT}$ for operation through $1.15x10^{8}$ MWH(t), as well as adjustments for possible errors in the pressure and temperature sensing instruments.

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 and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. In order to estimate the material properties at the 1/4 and 3/4 positions in the vessel plate, the shift in NDTT is assumed to be 62% and 22%, respectively of the irradiation samples properties. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.6.1 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 for reactor criticality and for inservice leak and hydrostatic testing.

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

B. Coolant Chemistry

A steady state radioiodine concentration limit of 1.1 $\mu$Ci of I-131 dose equivalent per gram of water in the reactor coolant system can be reached if the gross radioactivity in the gaseous effluents are near the limit as set forth in Specification 3.8.C.1.a or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was