3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing, the reactor vessel shell temperatures specified in 4.6.B.1, except for the reactor vessel bottom head, shall be at or above the temperatures shown on the two curves of Figure 3.6.2, where the dashed curve, "RPV Core Beltline," is increased by the core beltline temperature adjustment from Figure 3.6.1. The reactor vessel bottom head temperature shall be at or above the temperatures shown on the solid curve of Figure 3.6.2, "RPV Remote from Core Beltline," with no adjustment from Figure 3.6.1.

2. During heatup by non-nuclear means (except with the reactor vessel vented), cooldown following nuclear shutdown, or low level physics tests the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.3 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

3. During all operation with a critical reactor, other than for low level physics tests or at times when the reactor vessel is vented, the reactor vessel shell and fluid temperatures specified in 4.6.A shall be at or above the higher of the temperatures of Figure 3.6.4 where the dashed curve, "RPV Core Beltline," is increased by the expected shift in RT_{NDT} from Figure 3.6.1.

4.0 SURVEILLANCE REQUIREMENTS

B. Reactor Vessel Temperature and Pressure

1. During in-service hydrostatic or leak testing when the vessel pressure is above 312 psig, the following temperatures shall be recorded at least every 15 minutes.

   a. Reactor vessel shell adjacent to shell flange.
   b. Reactor vessel bottom head.
   c. Reactor vessel shell or coolant temperature representative of the minimum temperature of the beltline region.

Bases 3.6/4.6 (Continued):

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this in the operation of the reactor pressure vessel. Two types of information are needed in this analysis: 1) A relationship between the changes in fracture toughness of the reactor pressure vessel steel and the neutron fluence (integrated neutron flux), and 2) A measure of the neutron fluence at the point of interest in the reactor pressure vessel wall.

The relationship of predicted adjustment of reference temperature versus fluence and the copper and nickel content of the core beltline materials given in Regulatory Guide 1.99, Revision 2, was originally used to define the core beltline temperature adjustment versus fluence shown on Figure 3.6.1.

A relationship between full power years of operation and neutron fluence has been experimentally determined for the reactor vessel. The vessel pressurization temperatures at any time period can be determined from the thermal energy output of the plant and Figure 3.6.1 used in conjunction with Figure 3.6.2 (pressure tests), Figure 3.6.3 (mechanical heatup or cooldown with a noncritical core), or Figure 3.6.4 (operation with a critical core). During the first fuel cycle, only calculated neutron fluence values were used. At the first refueling, neutron dosimeter wires which were installed adjacent to the vessel wall were removed to experimentally determine the neutron fluence versus full power years of operation. This experimental result was updated by testing additional dosimetry removed with the first surveillance capsule.

Reactor vessel material samples are provided, however, to verify the relationship expressed by Figure 3.6.1. Three sets of mechanical test specimens representing the base metal, weld metal, and weld heat affected zone (HAZ) metal have been placed in the vessel and can be removed and tested as required. Two sets of specimens were contained in the first surveillance capsule which was removed from the vessel in 1981. One set of specimens was tested at this time. The second set was later inserted into a new capsule, and installed in the Prairie Island Nuclear Generating Plant RPV for accelerated irradiation. This capsule was removed and tested in 1996. NSP performed calculations per the requirements of Regulatory Guide 1.99, Rev. 2, Position 2.1 to develop new pressure/temperature (P-T) curves. Results of Charpy V-notch impact tests for the two sets of data and from 1997 non-irradiated material test data were used in developing the revised Figures 3.6.1, 3.6.2, 3.6.3, and 3.6.4.