ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

REVISED TECHNICAL SPECIFICATION PAGES

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PBAPS
LIST OF FIGURES

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LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cool-down shall not exceed 100° F increase (or decrease) in any one-hour period.

2. The reactor vessel shall not be pressurized for inservice hydrostatic testing above the pressure allowable for a given temperature by Figure 3.6.1.

The reactor vessel shall not be pressurized during heatup by non-nuclear means, during cooldown following nuclear shut down or during low level physics tests above the pressure allowable by Figure 3.6.2, based on the temperatures recorded under 4.6.A.

The reactor vessel shall not be pressurized during operation with a critical core above the pressure allowable by Figure 3.6.3, based on the temperatures recorded under 4.6.A.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cool-downs, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any 2 readings taken over a 45 minute period is less than 5° F.

(a) Bottom head drain
(b) Recirculation loop A and B.

2. Reactor vessel temperature and reactor coolant pressure shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220° F and the reactor vessel is not vented.

Test specimens of the reactor vessel base, weld and heat affected zone metal were installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-66 to the degree discussed in the FSAR.

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   The reactor vessel shall not be pressurized during heatup by non-nuclear means, during cooldown following nuclear shut down or during low level physics tests above the pressure allowable by Figure 3.6.2, based on the temperatures recorded under 4.6.A.

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LIMITING CONDITIONS FOR OPERATION

3.6.A Thermal and Pressurization Limitations (Cont'd)

3. The reactor vessel head bolting studs shall not be under tension unless the temperatures of the closure flanges and adjacent vessel and head materials are greater than 70° F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50° F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145° F.

SURVEILLANCE REQUIREMENTS

4.6.A Thermal and Pressurization Limitations (Cont'd)

Selected surveillance specimens shall be removed and tested in accordance with 10 CFR 50, Appendix H, to experimently verify or adjust the calculated values of integrated neutron flux and irradiation embrittlement that are used to determine the RT_{NDT} for Figures 3.6.1, 3.6.2 and 3.6.3, and the figures shall be updated based on the results.

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

LIMITING CONDITIONS FOR OPERATION

3.6.A Thermal and Pressurization Limitations (Cont'd)

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SURVEILLANCE REQUIREMENTS

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5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

3.6.A & 4.6.A BASES (Cont'd)

Operating limits on the reactor pressure and temperature were developed after consideration of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G to 10 CFR Part 50. These considerations involved the reactor vessel beltline and certain areas of discontinuity (e.g. feedwater nozzles and vessel head flange). These operating limits (Figures 3.6.1, 3.6.2 and 3.6.3) assure that a postulated surface flaw can be safely accommodated. Figure 3.6.3 includes an additional 40° F margin required by 10 CFR 50 Appendix G.

The fracture toughness of the vessel low alloy steel in the core region, referred to as beltline, gradually decreases with exposure to neutrons, and it is necessary to account for this change. Regulatory Guide 1.99, Revision 2 provides methods for predicting decreased fracture toughness, in terms of shift in reference temperature of nil-ductility (RT_{NDT}). Generic methods are used until two surveillance capsules are removed and tested, at which time the surveillance test results may be used to develop plant-specific relationships of RT_{NDT} shift versus fluence.

Three capsules of neutron flux wires and samples of vessel material were installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The first capsule of wires and samples was removed and tested in 1989 to experimentally verify the irradiation shift in RT_{NDT} predicted by Regulatory Guide 1.99, Revision 2 methods. The results of the testing are documented in GE Report SASR 90-50 of DRF B11-00494. The results of vessel material testing will not be factored into Figures 3.6.1, 3.6.2 and 3.6.3 until the second capsule is tested. However, the flux wire results were used to predict the design fluence (valid to 32 effective full power years (EFPY)).

The flux wire test results provide the flux at one location in the vessel. The flux distribution can be determined analytically from the core physics data. The ratio of the flux at the peak vessel location to that at the flux wire location, known as the lead factor, was calculated to relate the flux wire test results to the maximum value for the vessel. In developing Figures 3.6.1, 3.6.2 and 3.6.3, the shift predicted by Regulatory Guide 1.99, Revision 2 methods for 32 EFPY of fluence was taken into account. However, in comparing the beltline operating limits (with 32 EFPY shift) to the feedwater nozzle limits, it was determined that the feedwater nozzle was more limiting. Since the feedwater nozzles do not experience significant changes in fracture toughness due to irradiation, the pressure-temperature limits in Figures 3.6.1, 3.6.2 and 3.6.3 apply, without any RT_{NDT} shifting, through 32 EFPY of operation.

As described in paragraph 4.2.5 of the Final Safety Analysis Report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50° F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

3.6.A & 4.6.A BASES (Cont'd)

The design basis event for protection from pressure in excess of vessel design pressure, as required by the ASME Boiler and Pressure Vessel Code, is the closure of all MSIVs resulting in a high flux scram (the slowest indirect scram due to high pressure). The reactor vessel pressure Code limit of 1375 psig is well above the peak pressure produced by this most limiting overpressure event. This is discussed in more detail in Section 4.4.6 of the FSAR and GE safety analyses NEDE-24011-P-A.

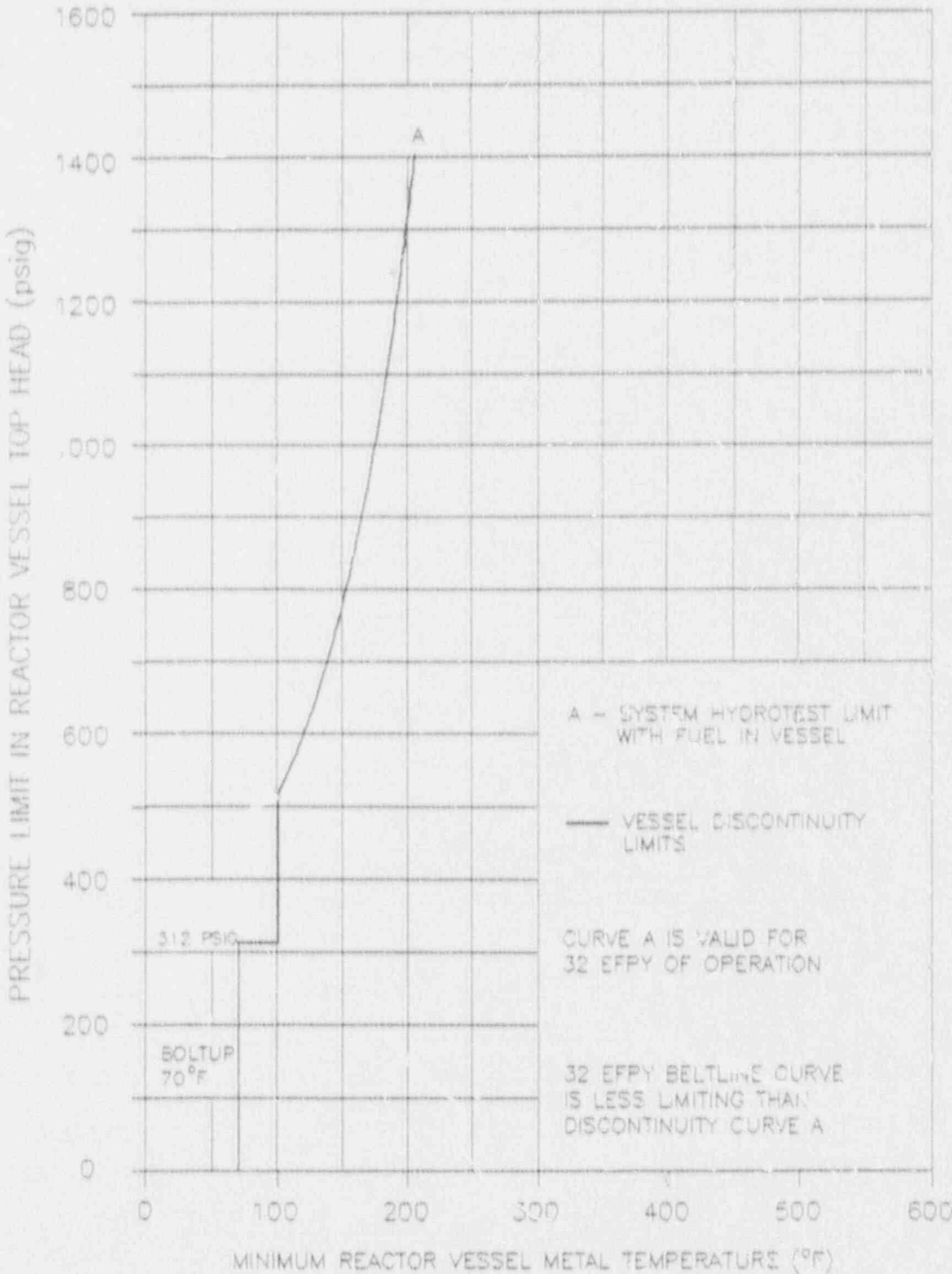

Figure 3.6.1 Peach Bottom 3 Minimum Temperature for Pressure Tests Such as Required by Section XI

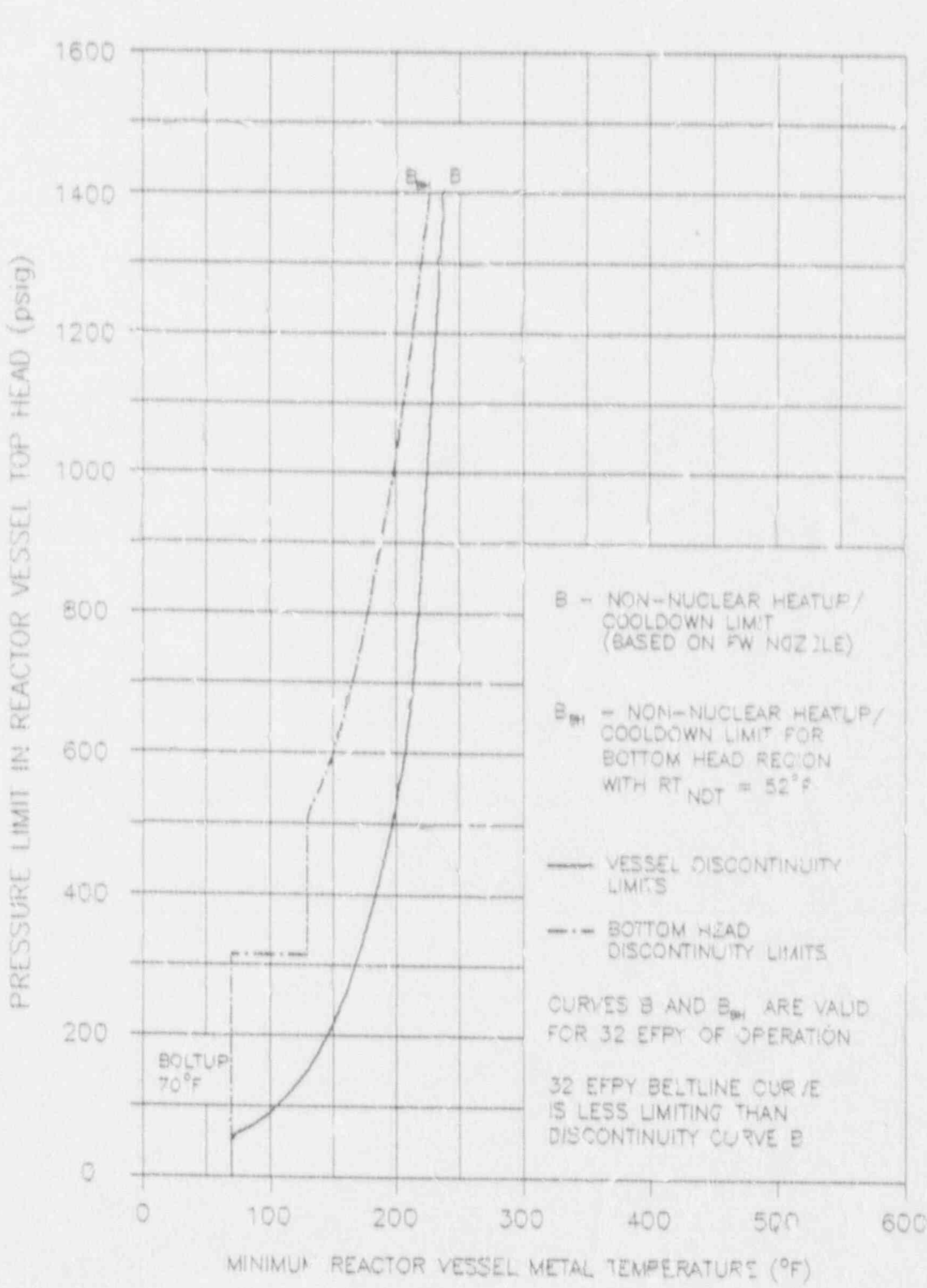

Figure 3.6.2 Peach Bottom 2 Minimum Temperature for Mechanical Heatup or Cooldown Following Nuclear Shutdown

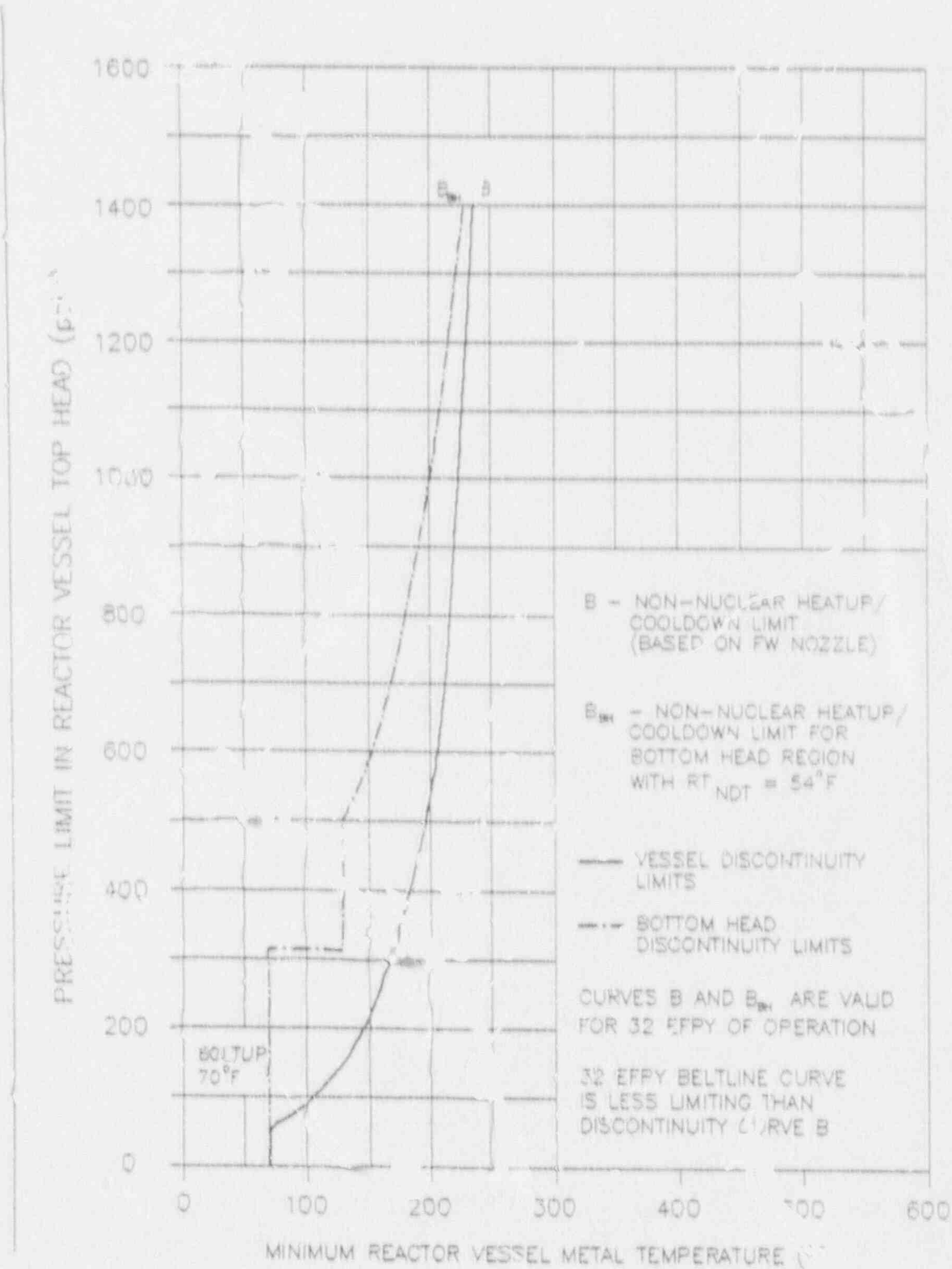

Figure 3.6.2 Peach Bottom 3 Minimum Temperature for Mechanical Heatup or Cooldown Following Nuclear Shutdown

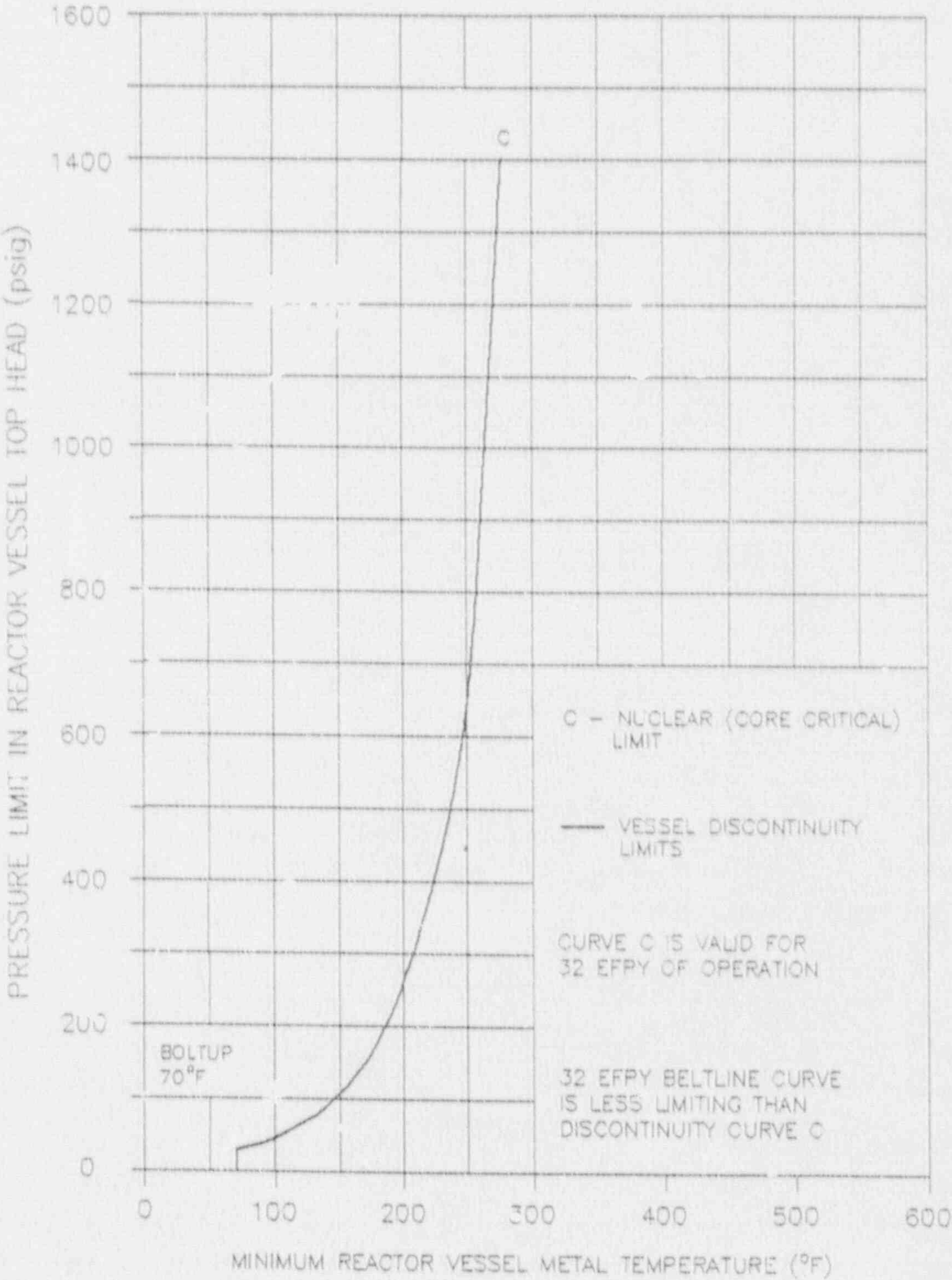

Figure 3.6.3 Peach Bottom 3 Minimum Temperature for Core Operation (Criticality)

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