ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES
REGARDING PRESSURE-TEMPERATURE LIMITS

(JPTS-89-024)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

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LIST OF FIGURES

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3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

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. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

The reactor vessel head bolting studs shall not be under tension unless the temperatures of the reactor vessel flange and the reactor head flange are at least 90°F.

2. In-Service Hydrostatic and Leak Tests

During in-service hydrostatic or leak testing the Reactor Coolant System pressure and temperature shall be on or to the right of curve A shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour period shall be:

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

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. Pressurization and Thermal Limits

1. Reactor Vessel Head Stud Tensioning

When in the cold condition, the reactor vessel head flange and the reactor vessel flange temperatures shall be recorded:

a. Every 12 hours when the reactor vessel head flange is $\leq$120°F and the studs are tensioned.

b. Every 30 minutes when the reactor vessel head flange is $\leq$100°F and the studs are tensioned.

c. Within 30 minutes prior to and every 30 minutes during tensioning of reactor vessel head bolting studs.

2. In-Service Hydrostatic and Leak Tests

During hydrostatic and leak testing the Reactor Coolant System pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

3.6 (cont'd)

a. $\leq$20°F when to the left of curve C.

b. $\leq$100°F when on or to the right of curve C.

3. Non-Nuclear Heatup and Cooldown

During heatup by non-nuclear means (mechanical), cooldown following nuclear shutdown and low power physics tests the Reactor Coolant System pressure and temperature shall be on or to the right of the curve B shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour shall be $\leq$100°F.

4. Core Critical Operation

During all modes of operation with a critical core (except for low power physics tests) the reactor Coolant System pressure and temperature shall be at or to the right of the curve C shown in Figure 3.6-1 Part 1, 2, or 3 and the maximum temperature change during any one hour shall be $\leq$100°F.

5. With any of the limits of 3.6.A.1 through 3.6.A.4 above exceeded, either

a. restore the temperature and/or pressure to within the limits within 30 minutes, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system, and determine that the reactor coolant system remains acceptable for continued operations; or

4.6 (cont'd)

3. Non-Nuclear Heatup and Cooldown

During heatup by non-nuclear means, cooldown following nuclear shutdown and low power physics tests, the reactor coolant system pressure and temperature shall be recorded every 30 minutes until two consecutive temperature readings are within 5°F of each other.

4. Core Critical Operation

During all modes of operation with a critical core (except for low power physics tests) the reactor Coolant System pressure and temperature shall be recorded within 30 minutes prior to withdrawal of control rods to bring the reactor critical and every 30 minutes during heatup until two consecutive temperature readings are within 5°F of each other.

3.6 and 4.6 BASES (cont'd)

The expected neutron fluence at the reactor vessel wall can be determined at any point during plant life based on the linear relationship between the reactor thermal power output and the corresponding number of neutrons produced. Accordingly, neutron flux wires were removed from the reactor vessel with the surveillance specimens to establish the correlation at the capsule location by experimental methods. The flux distribution at the vessel wall and 1/4 thickness (1/4T) depth was analytically determined as a function of core height and azimuth to establish the peak flux location in the vessel and the lead factor of the surveillance specimens.

Regulatory Guide 1.99, Revision 2 is used to predict the shift in RT_{NTD} as a function of fluence in the reactor vessel beltline region. An evaluation of the irradiated surveillance specimens, which were withdrawn from the reactor in April, 1985 (6 EFPY), shows a shift in RT_{NTD} less than that predicted by Regulatory Guide 1.99, Revision 2.

Operating limits for the reactor vessel pressure and temperature during normal heatup and cooldown, and during in-service hydrostatic and leak testing were established using 10 CFR 50 Appendix G, May, 1983 and Appendix G of the Summer 1984 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These operating limits assure that the vessel could safely accommodate a postulated surface flaw having a depth of 0.24 inch at the flange-to-vessel junction, and one-quarter of the material thickness at all other reactor vessel locations and discontinuity regions. For the purpose of setting these operating limits, the reference temperature, RT_{NDT}, of the vessel material was estimated from impact test data taken in accordance with the requirements of the Code to which the vessel was designed and manufactured (1965 Edition including Winter 1966 addenda). The RT_{NDT} values for the reactor vessel flange region and for the reactor vessel shell beltline region are 30°F, based on fabrication test reports. The RT_{NDT} for the remainder of the vessel is 40°F.

The first surveillance capsule containing test specimens was withdrawn in April, 1985 after 6 EFPY. The test specimens removed were tested according to ASTM E 185-82 and the results are in GE report MDE-49-0386. The next surveillance capsule will be removed after 15 EFPYs of operation and the results of the examination used as a basis for revision of Figure 3.6-1 curves A, B and C for operation of the plant after 16 EFPYs.

Figure 3.6-1 is comprised of three parts: Part 1, Part 2, and Part 3. Parts 1, 2, and 3 establish the pressure-temperature limits for plant operations through 12, 14, and 16 Effective Full Power Years (EFPY) respectively. The appropriate figure and the pressure-temperature curves are dependent on the number of accumulated EFPY. Figure 3.6-1, Part 1 is for operation through 12 EFPY, Figure 3.6-1, Part 2 is for operation at greater than 12 EFPY through 14 EFPY, and Figure 3.6-1, Part 3 is for operation at greater than 14 EFPY through 16 EFPY. The curves contained in Figure 3.6-1 are developed from the General Electric Report DRF 137-0010, "Implementation of Regulatory Guide 1.99, Revision 2 for the James A. Fitzpatrick Nuclear Power Plant," dated June, 1989.

Figure 3.6-1 curve A establishes the minimum temperature for hydrostatic and leak testing required by the ASME Boiler and Pressure Vessel Code, Section XI. Test pressures for in-service hydrostatic and leak testing are a function of the testing temperature and the component material. Accordingly, the maximum hydrostatic test pressure will be 1.1 times the operating pressure of about 1105 psig.

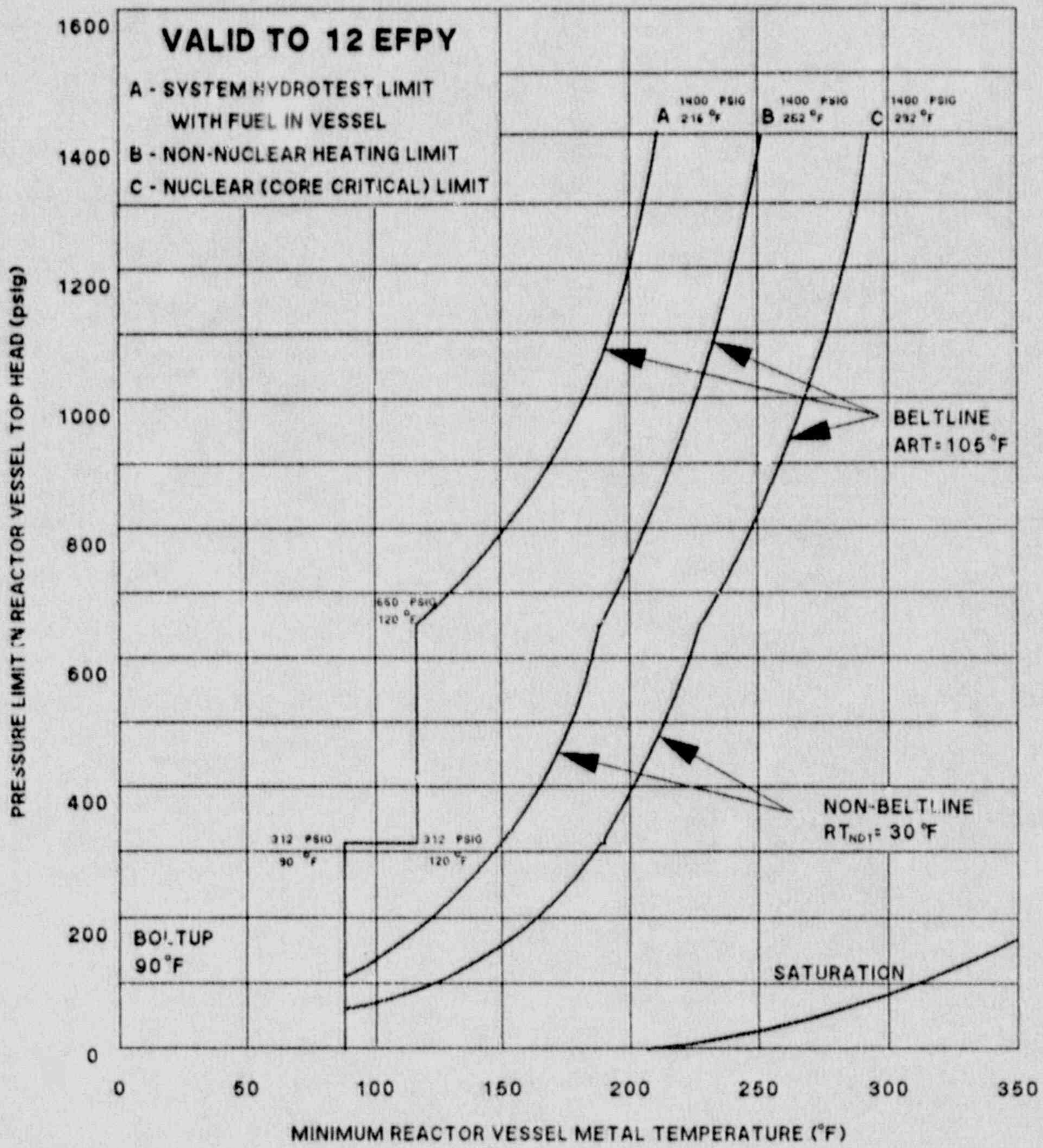

Figure 3.6-1 Part 1 Reactor Vessel Pressure-Temperature Limits Through 12 EFPY

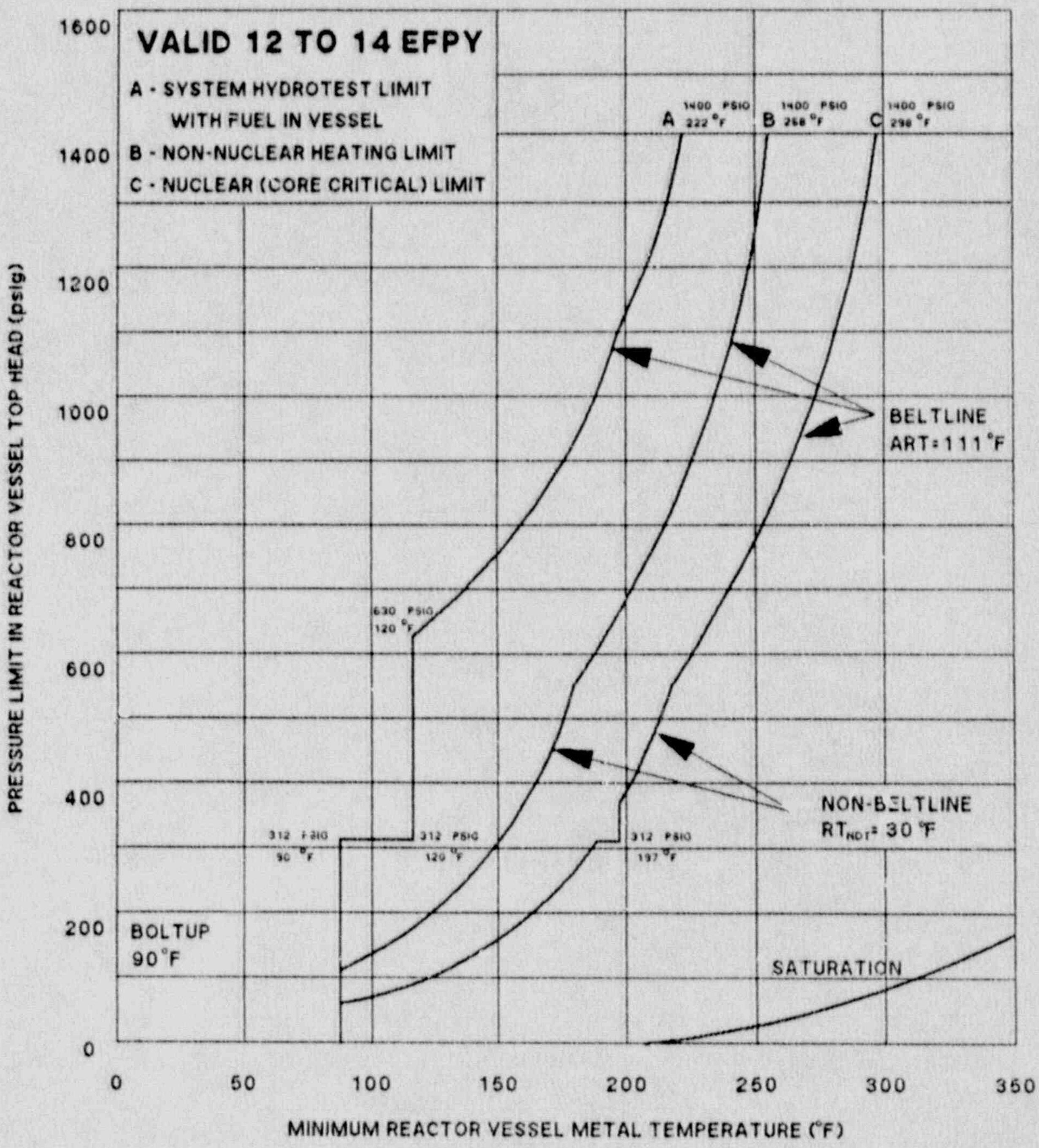

Figure 3.6-1 Part 2 Reactor Vessel Pressure-Temperature Limits Through 14 EFPY

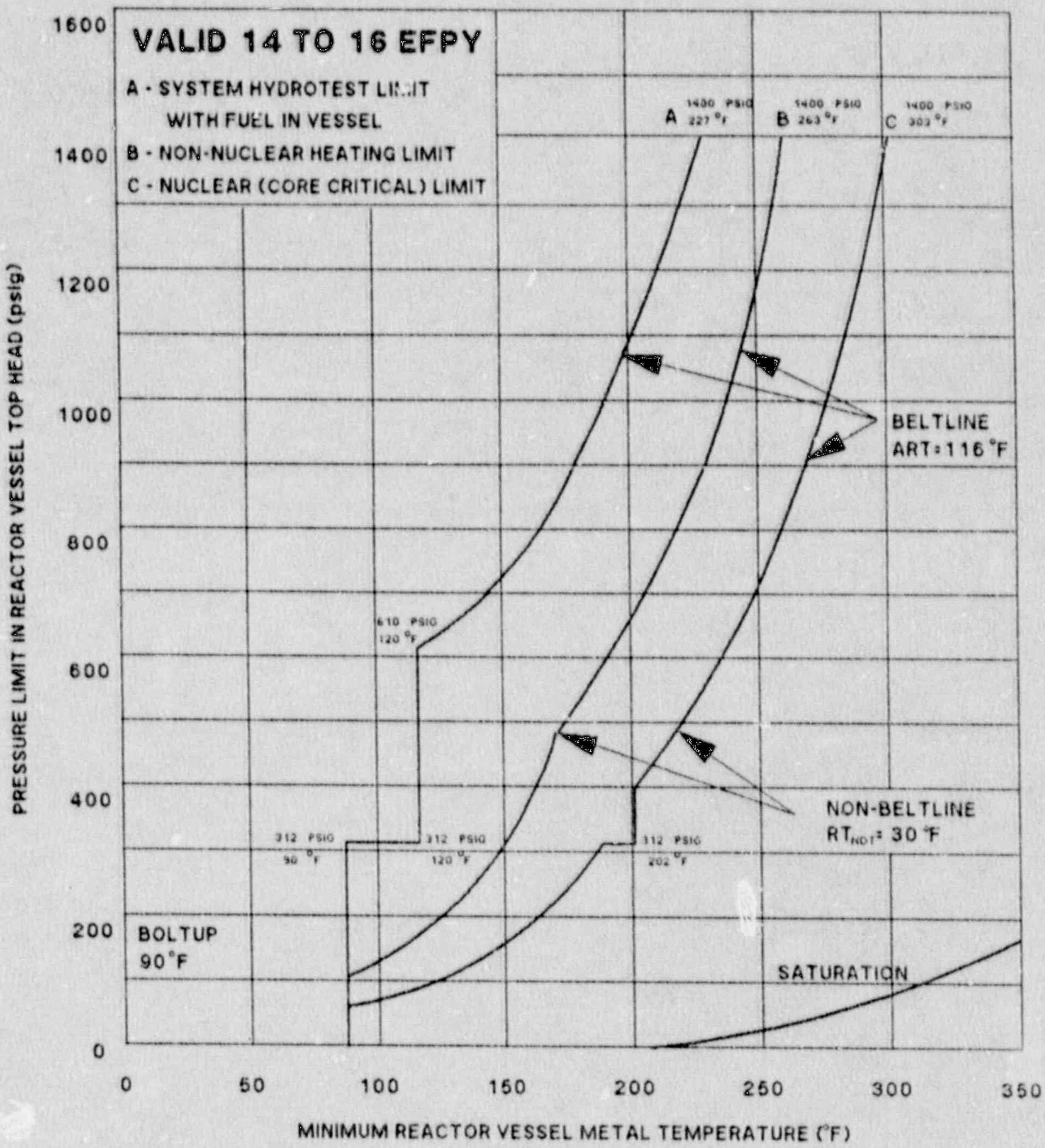

Figure 3.6-1 Part 3 Reactor Vessel Pressure-Temperature Limits Through 16 EFPY

Amendment No.

ATTACHMENT II

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING PRESSURE-TEMPERATURE LIMITS

(JPTS-89-024)

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59

I. DESCRIPTION OF THE PROPOSED CHANGES

This application for an amendment to the James A. FitzPatrick Technical Specifications revises Specification 3.6.A, "Pressurization and Thermal Limits," and its associated bases to comply with Generic Letter 88-11 (Reference 1) and Regulatory Guide 1.99, Revision 2 (Reference 2). Specifically, the pressure-temperature curves in Figure 3.6-1 are replaced with new curves for operation to 12, 14, and 16 Effective Full Power Years (EFPY). The associated Limiting Condition for Operation (LCO) and the Bases Section are revised to reflect the new pressure-temperature curves.

The specific changes to the Technical Specifications are:

A. Pressure-Temperature Limit Changes

Replace existing Figure 3.6-1, "Reactor Vessel Pressure - Temperature Limits," on page 163 with the following new figures:

Figure 3.6-1 Part 1, "Reactor Vessel Pressure - Temperature Limits Through 12 EFPY," on page 163

Figure 3.6-1 Part 2, "Reactor Vessel Pressure - Temperature Limits Through 14 EFPY," on page 163a

Figure 3.6-1 Part 3, "Reactor Vessel Pressure - Temperature Limits Through 16 EFPY," on page 163b

B. Associated Wording Changes

1. Section 3.6.A.2, "In-service Hydrostatic and Leak Tests," page 136;
Section 3.6.A.3, "Non-nuclear heatup and Cooldown," page 137;
Section 3.6.A.4, "Core Critical Operation," page 137:

   Replace "Figure 3.6-1" with "Figure 3.6-1 Part 1, 2, or 3"

2. Bases Section 3.6, "Pressurization and Thermal Limits," page 147:

   - Replace second paragraph on page 147 (begins with "A method of relating ...") with:

     Regulatory Guide 1.99, Revision 2 is used to predict the shift in RT_{NTD} as a function of fluence in the reactor vessel beltline region. An evaluation of the irradiated surveillance specimens, which were withdrawn from the reactor in April, 1985 (6 EFPY), shows a shift in RT_{NTD} less than that predicted by Regulatory Guide 1.99, Revision 2.

- Delete the third sentence in the fourth paragraph on page 147. The sentence to be deleted reads as follows:

  The curves of Figure 3.6-1, A through C, reflect findings in the report related to copper-phosphorus content of the reactor vessel shell beltline, flux wire testing fluence distribution analysis, and Charpy V-Notch specimen testing.

- Add the following paragraph between the fourth and fifth paragraphs on page 147.

  Figure 3.6-1 is comprised of three parts: Part 1, Part 2, and Part 3. Parts 1, 2, and 3 establish the pressure-temperature limits for plant operations through 12, 14, and 16 Effective Full Power Years (EFPY) respectively. The appropriate figure and the pressure-temperature curves are dependent on the number of accumulated EFPY. Figure 3.6-1, Part 1 is for operation through 12 EFPY, Figure 3.6-1, Part 2 is for operation at greater than 12 EFPY through 14 EFPY, and Figure 3.6-1, Part 3 is for operation at greater than 14 EFPY through 16 EFPY. The curves contained in Figure 3.6-1 are developed from the General Electric Report DRF 137-0010, "Implementation of Regulatory Guide 1.99, Revision 2 for the James A. Fitzpatrick Nuclear Power Plant," dated June, 1989.

3. List of Figures, page vii: replace "Figure 3.6-1, Reactor Vessel Pressure - Temperature Limits," page 163 with

   Figure 3.6-1, Part 1, "Reactor Vessel Pressure - Temperature Limits Through 12 EFPY," page 163

   Figure 3.6-1, Part 2, "Reactor Vessel Pressure - Temperature Limits Through 14 EFPY," page 163a

   Figure 3.6-1, Part 3, "Reactor Vessel Pressure - Temperature Limits Through 16 EFPY," page 163b

II. PURPOSE OF THE PROPOSED CHANGES

Regulatory Guide 1.99, Revision 2 (Reference 2) revised the methodology used to evaluate neutron radiation embrittlement of reactor vessel beltline materials. Generic Letter 88-11 (Reference 1) requests licensees to use Revision 2 of the regulatory guide to evaluate predicted embrittlement. The Authority has reevaluated the effect of neutron radiation on reactor vessel materials (Reference 3) and is changing the pressure-temperature limits contained in the Fitzpatrick Technical Specifications. This proposed change is consistent with the requirements of Section V of 10 CFR 50, Appendix G.

Background

Specification 3.6.A, "Pressurization and Thermal Limits," establishes, in part, pressure-temperature curves which define the minimum pressure and temperature for three reactor operating conditions: 1) system hydrostatic and leakage tests, 2) heatup and cooldown, and 3) core critical operation. These pressure-temperature curves protect the reactor pressure vessel from brittle failure by clearly identifying the regions where the vessel is subject to brittle fracture failure modes.

Amendment 113 (Reference 4) revised the pressure-temperature limits to be consistent with test results and analyses performed on the irradiated surveillance capsule removed from the Fitzpatrick reactor in April, 1985. (Surveillance capsules are installed in the reactor vessel before startup and contain test specimens that are made from the plate, weld, and heat affected zone materials of the reactor beltline.) Radiation embrittlement was calculated using the surveillance data and adjusting the nil-ductility reference temperature (RT_{NDT}) in accordance with Regulatory Guide 1.99, Revision 1 methodology.

The effect of neutron radiation on reactor vessel materials has been recalculated (Reference 3) in accordance with Generic Letter 88-11 and Regulatory Guide 1.99, Revision 2. The resultant shift in RT_{NTD} bounds the previously calculated results for the beltline region of the core.

New beltline Pressure-Temperature curves were developed for operation to 12, 14, and 16 Effective Full Power Years (EFPY). The non-beltline region curves (recirculation inlet nozzles and head flanges) are not affected by the changes to RT_{NTD} shift associated with Regulatory Guide 1.99, Revision 2.

The beltline curves apply to the vessel plates and welds and are limiting above 500 psig. For example, at a 1000 psig on the leak test curve, the required test temperature is 192°F for 16 EFPY compared to the current limitation of 157°F.

III. IMPACT OF THE PROPOSED CHANGES

The purpose of Specification 3.6.A, "Pressurization and Thermal Limits," is to establish operating limits that provide a wide margin to brittle failure of the reactor pressure vessel. The basis of the Pressure-Temperature (P-T) limits is found in Appendix G to 10 CFR 50 and in Section 4.2 of the updated FSAR. The limits are not derived from the design basis accident analyses, but are prescribed to avoid encountering pressures, temperatures, and temperature rate-of-changes which might cause undetected flaws to propagate.

The first technical specification change lowers the P-T curves (i.e., a higher temperature is required for a given pressure) which in turn "narrows" the reactor coolant system operating window and lengthens the time required for hydrostatic testing. This change is consistent with Generic Letter 88-11 to ensure a conservative margin to non-ductile failure.

The new P-T curves were developed for three service periods: $\leq$ 12 EFPYs, $\leq$ 14 EFPYs, and $\leq$ 16 EFPYs. The use of three curves instead of one lessens operational impacts by phasing the increases in minimum temperature over three distinct service periods. Each set of P-T curves is conservative, because the conditions at the end of each service period (12, 14, or 16 EFPY) yield the highest fluence and, therefore, the largest predicted shift in RT_{NTD}.

The second change revises the text of Sections 3.6.A and its associated Bases. The change also updates the List of Figures provided at the beginning of the Fitzpatrick Technical Specifications. These changes are editorial in nature and reflect the new limits on pressure and temperature.

Both proposed changes are administrative in nature. They do not involve any physical modification to the plant, nor do they introduce any new failure modes. The changes do not alter the conclusions of the safety analyses contained in the FSAR and the NRC staff's SER.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the James A. FitzPatrick Nuclear Power Plant in accordance with the proposed amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated. The effect of neutron radiation on reactor vessel materials has been recalculated using the latest NRC approved guidance (Regulatory Guide 1.99, Revision 2 methodology). The resultant changes to the pressure-temperature limits contained in Specification 3.6.A will preclude brittle fracture failure of the reactor vessel. The requirements on pressure-temperature limitations contained in FSAR Section 4.2 are unaffected.

   Changes are also proposed to Section 3.6.A and its associated Bases to reflect the new pressure-temperature curves. These changes are editorial in nature and , as such, can not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. create the possibility of a new or different kind of accident from those previously evaluated. The proposed changes revise existing limitations and are administrative in nature. They do not involve any physical modification to the plant, nor do they introduce any new failure modes.

   The changes to Section 3.6.A and its Bases Section are editorial in nature; thus, they can not create the possibility of a new or different kind of accident from those previously evaluated.

3. involve a significant reduction in the margin of safety. The safety margins are increased because the new Pressure-Temperature limitations are more conservative (restrictive) and a more accurate method is used to predict radiation embrittlement.

   The changes to Section 3.6.A and its Bases Section are editorial in nature; thus, they can not involve a significant reduction in the margin of safety.

In the April 6, 1983 Federal Register (48FR14870) NRC published examples of license amendments that are not likely to involve a significant hazards consideration. Examples (i) and (vii) are applicable to these changes,

(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature.

(vii) A change to make a license conform to the change in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes do not impact the Fire Protection Program at the FitzPatrick plant, nor will the change impact the environment.

VI. CONCLUSION

The changes, as proposed, do not constitute an unreviewed safety question as defined in 10 CFR 50.59. That is, they:

a. will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report;

b. will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report;

c. will not reduce the margin of safety as defined in the basis for any technical specification; and

d. involves no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. USNRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," dated July 12, 1988.

2. USNRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," dated May 1988.

3. NYPA letter, J. C. Brons to NRC, dated June 30, 1989, JPN-89-044, "Response to Generic Letter 88-11, Radiation Embrittlement of Reactor Vessel Materials."

4. Amendment 113 to the James A. Fitzpatrick Operating License, October 22, 1987.

5. USAEC "Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated November 20, 1972.

6. USAEC "Supplement 1 to the Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated February 1, 1973.

7. USAEC "Supplement 2 to the Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated October 4, 1974.

8. James A. Fitzpatrick Nuclear Power Plant Updated Final Safety Analysis Report, Section 4.2.