ATTACHMENT I TO JPN-87-13

PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING PRESSURE-TEMPERATURE LIMITS (PTS-86-05)

NEW YORK POWER AUTHORITY

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
DPR-59

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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 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. ≤ 20°F when to the left of curve C.

b. ≤ 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 and the maximum temperature change during any one hour shall be ≤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 and the maximum temperature change during any one hour shall be ≤ 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.

b. be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. Idle Recirculation Loop Startup

When Reactor Coolant System temperature is $>$ 140°F an idle recirculation loop shall not be started unless:

a. The temperature differential between the reactor coolant system and the reactor vessel bottom head drain line is $\leq$145°F, and

b. When both loops are idle, the temperature difference between the reactor coolant system and the idle loop to be started is $\leq$50°F, or

c. When only one loop is idle, the temperature difference between the idle loop and the operating loop is $\leq$50°F.

B. Deleted

6. Idle Recirculation Loop Startup

Within 30 minutes prior to startup of an idle loop:

a. The differential temperature between the reactor coolant system and the reactor vessel bottom head drain line shall be recorded, and

b. When both loops are idle, the differential temperature between the reactor coolant system and the idle loop to be started shall be recorded, or

c. When only one loop is idle, the temperature differential between the idle loop and the operating loop shall be recorded.

7. Reactor Vessel Flux Monitoring

The reactor vessel Flux Monitoring Surveillance Program complies with the intent of the May, 1983 revision to 10 CFR 50, Appendices G and H. The next flux monitoring surveillance capsule shall be removed after 15 effective full power years (EFPYs) and the test procedures and reporting requirements shall meet the requirements of ASTM E 185-82.

B. Deleted

3.6 and 4.6 BASES

A. Pressurization and Thermal Limits

The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F/hr averaged over a period of 1 hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F/hr rate is applied provide for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hr applied continuously over a temperature range of 100°F to 546°F.

Calculated stresses were within 1965 ASME Boiler and Pressure Vessel Code, Section III, with 1966 addenda stress intensity and fatigue limits. The normal heating and cooling rate of 100°F/hr was also evaluated to assure protection against brittle fracture of the vessel shell remote from discontinuities. The rate meets the requirements of Appendix G to the Summer 1972 Edition of 1971 ASME III, throughout plant life, and is, therefore, satisfactory.

The limiting coolant temperature differential between the upper and lower regions of the reactor vessel, prior to recirculation pump operation, assures that the vessel bottom head region will not be warmed at an excessive rate due to rapid sweep-out of cold coolant in the vessel lower head region by recirculation pump operation (cold coolant can accumulate as a result of control rod drive inleakage and/or low recirculation flow rate during startup or hot standby). The limit on idle recirculation loop startup avoids high thermal stress effects in the pumps and piping, while also minimizing thermal stresses on the vessel nozzles.

The nil-ductility transition (NDT) temperature RT_{NDT} is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Reactor vessel flux monitoring samples are installed to conform with the 1972 draft revision of ASTM E 185. Surveillance program test results have established the magnitude of changes in the NDT temperature as a function of the integrated neutron exposure for BWR vessels. The design life of the reactor vessel is 40 years, and the maximum fast neutron exposure at 40 years was originally calculated to be 7.0×10^{17} n/cm^2. Based on the surveillance program test results, the EOL fluence is now estimated to be 1.7×10^{18} n/cm^2.

Fast neutron irradiation affects the fracture toughness of the reactor vessel material. In order to assure that non-ductile failure does not occur, two types of information are needed:

a) a relationship between the change in RT_{NDT} and the accumulated fast neutron fluence, and,

b) a relationship between the neutron fluence at the point of peak flux in the reactor pressure vessel shell and the effective full power years.

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.

A method of relating shift in RT_{NDT} to accumulated fast neutron ($>$1 MeV) fluence is contained in Regulatory Guide 1.99. Experimental results of the evaluation of the irradiated surveillance specimens taken from the reactor pressure vessel in April, 1985, show a shift in RT_{NDT} greater than predicted by Regulatory Guide 1.99. Therefore, the surveillance results were used with the methods of Regulatory Guide 1.99 to establish the RT_{NDT} shift. The shift for 16 EFPY was added to the unirradiated reactor pressure vessel shell beltline curve.

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 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. 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 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 or about 1105 psig.

3.6 and 4.6 BASES (cont'd)

Figure 3.6-1, curve B, provides limitations for plant heatup and cooldown when the reactor is not critical or during low power physics tests. The thermal limitation is based on maximum heatup and cooldown rates of 100°F/hr in any one-hour period.

Figure 3.6-1, curve C, establishes operating limits when core is critical. These limits include a margin of 40°F as required by 10 CFR 50 Appendix G.

The requirements for cold boltup of the reactor vessel closure are based on NDT temperature plus a 60°F factor of safety. This factor is based on the requirements of the ASME Code to which the vessel was built. For Figure 3.6-1, curves A, B and C, 60°F margins are only added to the low temperature portion of the curve where non-ductile failure is a concern. The closure flanges have an NDT temperature not greater than 30°F and are not subject to any appreciable neutron radiation exposure. Therefore, the minimum temperature of the flanges when the studs are in tension is 30°F plus 60°F, or 90°F.

B. Deleted

C. Coolant Chemistry

A radioactivity concentration limit of 20 μCi/ml total iodine can be reached if the gaseous effluents are near the limit as set forth in Radiological Effluent Technical Specification 2.3.A if there is a failure or a prolonged shutdown of the cleanup demineralizer.

In the event of a steam line rupture outside the drywell, with this coolant activity level, the resultant radiological dose at the site boundary would be 33 rem to the thyroid, under adverse meteorological conditions assuming no more than 3.1 μCi/gm of dose equivalent I-131. The reactor water sample will be used to assure that the limit of Specification 3.6.C is not exceeded. The total radioactive iodine activity would not be expected to change rapidly over a period of 96 hr. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant. Also during reactor startups and large power changes which could affect iodine levels, samples of reactor coolant shall be analyzed to insure iodine concentrations are below allowable levels. Analysis is required whenever the I-131 concentration is within a factor of 100 of its allowable equilibrium value. The necessity for continued sampling following power and offgas transients will be reviewed within 2 years of initial plant startup.

The surveillance requirements 4.6.C.1 may be satisfied by a continuous monitoring system capable of determining the total iodine concentration in the coolant on a real time basis, and annunciating at appropriate concentration levels such that sampling for isotopic analysis can be initiated. The design details of such a system must be submitted for evaluation and accepted by the Commission prior to its implementation and incorporation in these Technical Specifications.

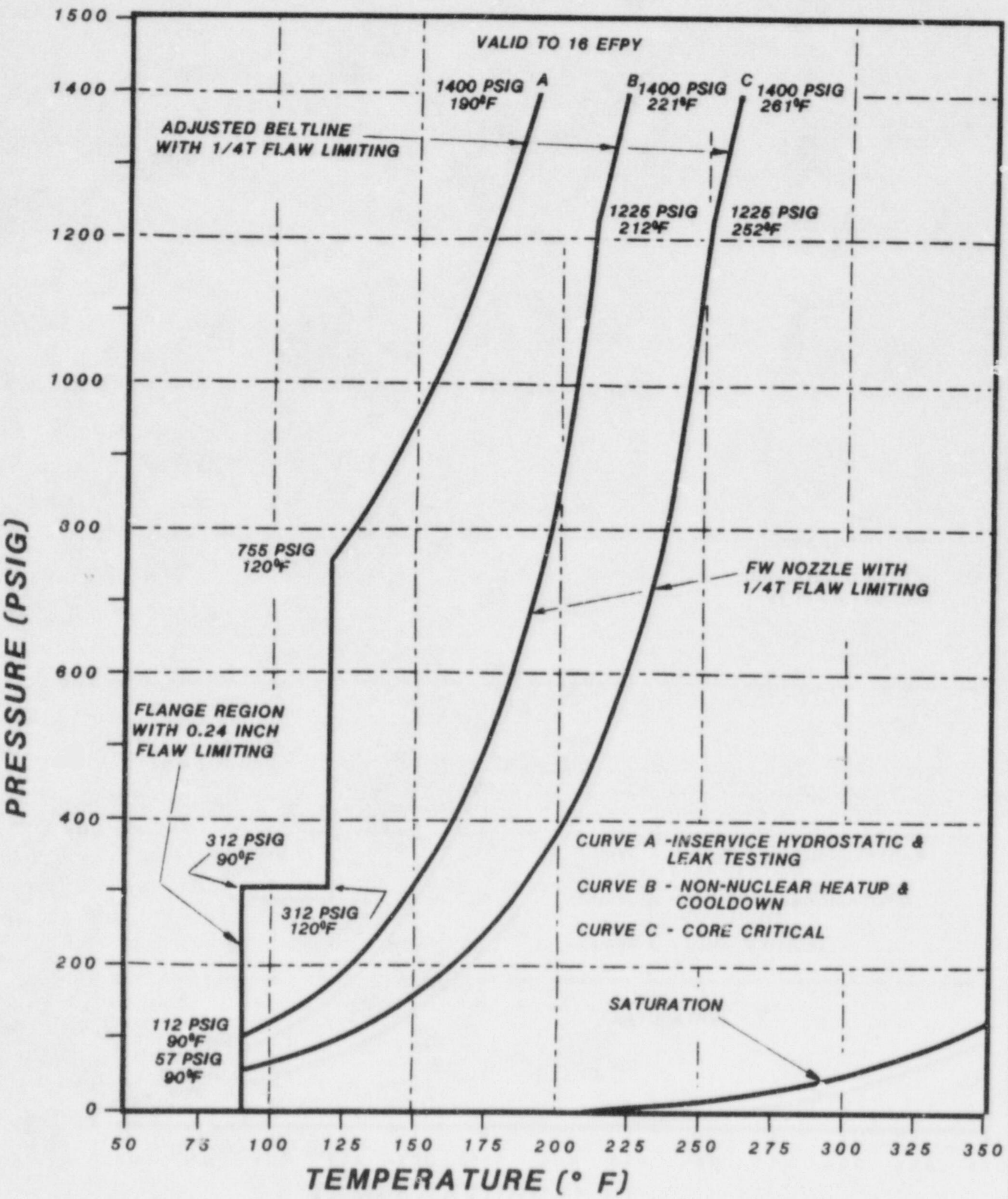

FIGURE 3.6-1
REACTOR VESSEL PRESSURE - TEMPERATURE LIMITS

Amendment No.

ATTACHMENT II TO JPN-87-13

SAFETY EVALUATION FOR THE PROPOSED CHANGE TO THE TECHNICAL SPECIFICATIONS REGARDING PRESSURE-TEMPERATURE LIMITS (PTS-86-05)

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I. Description of the Proposed Change

The proposed changes are limited to Sections 3.6, 4.6, the related Table of Contents pages, the related Bases, and Figure 3.6-1. The changes are:

Page(s)	Section	Change
ii and vii	Table of Contents	These pages are corrected to account for the changes made in this proposed amendment.
136	3.6 and 4.6	Titles are realigned.
136-138	3.6.A and 4.6.A	These sections are replaced with a new description of limiting conditions and surveillance requirements for the reactor coolant system pressurization temperature. This new description combined the current section 3.6.A/4.6.A (thermal limitations) and 3.6.B/4.6.B (pressurization temperature). This new description includes the following conditions: reactor vessel head stud tensioning, in-service hydrostatic and leak tests, non-nuclear heatup and cooldown, core critical operation, idle recirculation loop startup, and reactor vessel flux monitoring.
146-148	3.6 and 4.6.A Bases	This section is replaced with a new description of the bases used to establish operating pressure and temperature limits for the reactor vessel.
146-148	3.6 and 4.6.B Bases	This section is deleted.
148	3.6 and 4.6.C Bases	Change, "Radiological Effluent Technical Specification Section 3.2a" to "Radiological Effluent Technical Specification 2.3.A."
163	Fig. 3.6-1	Replace with new figure, "Reactor Vessel Pressure-Temperature Limits.

II. Purpose of the Proposed Changes

In compliance with the requirements of 10 CFR 50 Appendix H, a surveillance specimen removed from the FitzPatrick reactor was evaluated. Based on this evaluation, new operating limit curves valid up to 16 effective full power years were developed. The proposed changes reflect the new limits. The proposed change also reflects changes in the surveillance specimen withdrawal schedule.

The purpose of the new limits is to assure reactor vessel integrity. The limits accomplish this by restricting operating pressures and temperatures such that brittle fracture of the vessel cannot occur.

III. Impact of the Proposed Change

Assuring reactor vessel integrity involves evaluation of the fracture toughness of the vessel ferritic materials. The key values which characterize a material's fracture toughness are the reference temperature of nil-ductility transition (RT_{NDT}) and the upper shelf energy. These are defined in 10 CFR 50 Appendix G and in the Appendix G of the ASME Boiler and Pressure Vessel Code, Section III.

These documents contain requirements used to establish the pressure-temperature operating limits which must be met to avoid brittle fracture. The requirements of Appendix G of 10 CFR 50 include safety margins for both critical and non-critical conditions. Appendix H of 10 CFR 50 and ASTM-E185 establish the methods to be used for surveillance of the reactor vessel materials.

A method of relating shift in RT_{NDT} to accumulated fast neutron fluence is described in Regulatory Guide 1.99. In April, 1985 one of the surveillance specimen capsules required by Appendix H was removed from the James A. FitzPatrick Nuclear Power Plant reactor and evaluated. Based on the results of this evaluation, operating limit curves valid up to 16 effective full power years were developed for three reactor conditions: hydrostatic pressure testing, non-nuclear heatup and cooldown, and core critical operation. These curves provide specific guidance for each mode of operation and are included in the proposed Technical Specifications to assure that operation in the brittle fracture range is avoided.

The proposed amendment to the Technical Specifications will have little impact on the current operation of FitzPatrick.

IV. Evaluation of Significant Hazards Considerations

Operation of the FitzPatrick plant in accordance with the proposed amendment would not involve a significant hazard 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. Transient and accident analyses are based on reactor vessel integrity. When these analyses were originally done, operating limits were established to ensure that the temperature and pressure were kept in a safe range and reactor vessel integrity would be ensured. The proposed change will establish new, more conservative limits based on new calculations and on the results of evaluation of surveillance specimens. These limits were established according to the methods described in Regulatory Guide 1.99 and 10 CFR 50 Appendix G, and incorporating the safety margins included in Appendix G. Previous accident analyses are unaffected.

2. create the possibility of a new or different kind of accident from any accident previously evaluated. This change would only establish new pressure-temperature limits to ensure that operation in the brittle fracture range is avoided. The new limits were established according to NRC methodologies described in 10 CFR 50 Appendix G. The change restricts pressure at lower temperatures thus preventing operation in an unsafe region and could not create the possibility of a new type of accident.

3. involve a significant reduction in the margin of safety. Accident analyses are based on reactor vessel integrity. The analyses originally performed for the FitzPatrick plant established operating limits which ensured that temperature and pressure were maintained in a safe range and vessel integrity was assured. The proposed change will establish more conservative limits to ensure that the safety margin is maintained. The proposed limits reflect neutron flux based on testing and calculations. They incorporate safety margins as described in 10 CFR 50 Appendix G. The effect of this change will be an overall improvement in plant safety, and assurance that the margin of safety is maintained.

In the April 6, 1983 Federal Register (48FR14870), the NRC published examples of license amendments that are not likely to involve significant hazards considerations.

This proposed change most nearly corresponds to example vii, "A change to make a license conform to change in the regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations."

V. <u>Implementation of the Proposed Change</u>

The proposed change will not adversely impact the ALARA, Security, or Fire Protection programs at the FitzPatrick plant, nor will it impact the environment.

VI. <u>Conclusion</u>

The change, as proposed, does not constitute an unreviewed safety question as defined in 10 CFR 50.59, that is, it:

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

(b) will not increase the possibility of 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 bases for a technical specifications;

(d) does not constitute an unreviewed safety question;

(e) involves no significant hazards considerations, as defined in 10 CFR 50.92.

VII. References

1. James A. FitzPatrick FSAR Chapter 14 and SER.