October 22, 1987

Docket No. 50-333

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Mr. John C. Brons
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

The Commission has issued the enclosed Amendment No. 113 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 16, 1987.

The amendment would change the Technical Specifications to include revised limits that restrict operating pressures and temperatures to assure that brittle fracture of the reactor vessel cannot occur and that vessel integrity is maintained.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

Harvey Abelson, Project Manager
Project Directorate I-1
Division of Reactor Projects, I/II

Enclosures:
1. Amendment No. 113 to DPR-59
2. Safety Evaluation

cc: w/enclosures
See next page

PDI-1	PDI-1	OGC	PDI-1
CVogan	HAbelson		RCapra
10/9/87	10/8/87	10/15/87	10/22/87

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

POWER AUTHORITY OF THE STATE OF NEW YORK

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. DPR-59

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

   A. The application for amendment by Power Authority of the State of New York (the licensee) dated March 16, 1987, 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 2.C.(2) of Facility Operating License No. DPR-59 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 113, 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

Robert A. Capra

Robert A. Capra, Acting Director
Project Directorate I-1
Division of Reactor Projects, I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 22, 1987

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

ATTACHMENT TO LICENSE AMENDMENT NO. 113

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
ii	ii
vii	vii
136	136
137	137
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146	146
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LIST OF FIGURES

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 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 ≤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 ≤50°F, or

c. When only one loop is idle, the temperature difference between the idle loop and the operating loop is ≤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.

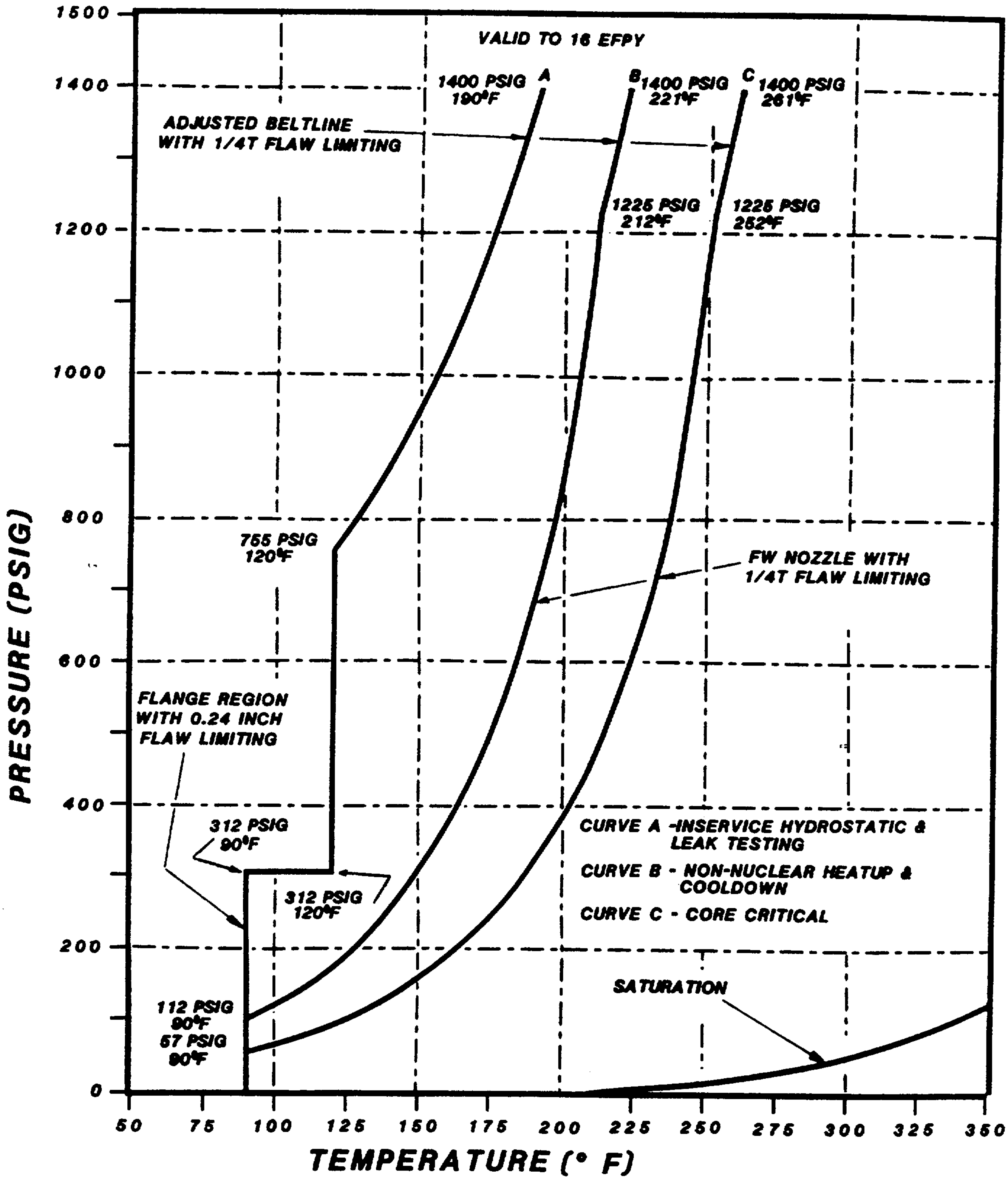

FIGURE 3.6-1
REACTOR VESSEL PRESSURE - TEMPERATURE LIMITS

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

INTRODUCTION

By letter dated March 16, 1987, the Power Authority of the State of New York proposed to revise the pressure-temperature limits in the James A. FitzPatrick Nuclear Power Plant Technical Specifications (TS), Section 3.6. The proposed pressure-temperature limits were developed from a General Electric (GE) report that the licensee submitted on April 12, 1985. The report is entitled, "James A. FitzPatrick Nuclear Power Plant Reactor Pressure Vessel Surveillance Materials Testing and Fracture Toughness Analyses," MDE-49-0386. The pressure-temperature limits are a set of curves that set minimum pressure and temperature for three reactor operating conditions: system hydrostatic and leakage tests, heatup or cooldown, and core critical operation.

DISCUSSION

Part of the NRC's effort to ensure integrity of the reactor vessel is to periodically evaluate the reduction in fracture toughness of the vessel material due to neutron irradiation damage. The effort consists of three steps. First, the licensee is required to establish a surveillance program in accordance with Appendix H of 10 CFR 50, which requires periodic withdrawal of surveillance capsules from the reactor vessel. The capsules are installed in the vessel prior to startup and contain test specimens that are made from the plate, weld, and heat affected zone materials of the reactor beltline. Secondly, the licensee is required to perform Charpy impact tests, tensile tests, and neutron fluence measurements of the specimens. These tests provide data for the actual neutron irradiation damage to the reactor vessel in terms of the reference temperature, RT_{NDT} and the upper shelf energy (USE). The neutron damage is indicated by the decrease in USE and temperature shift in RT_{NDT}. The USE is the average energy value for all specimens whose test temperature is above the upper end of the transition temperature region. The USE decreases as a function of neutron fluence and copper content in the irradiated material. The shift of the adjusted reference temperature is the temperature shift in the Charpy curve for the irradiated material relative to that for the unirradiated material.

According to Appendix G of 10 CFR 50, the USE must not be less than 50 ft-lb, and the adjusted RT_{NDT} not more than 200°F. Thirdly, the licensee is required to calculate the adjusted RT_{NDT} for a postulated flaw in the vessel wall. Regulatory Guide 1.99, Revision 1, may be used for this calculation. To calculate the corresponding pressure, the relationship between RT_{NDT} and the stress intensity factor of the postulated flaw as described in Appendix G of ASME Code, Section III may be used.

EVALUATION

The FitzPatrick reactor vessel is a 218-inch diameter BWR/4 series. It was constructed by Combustion Engineering to the 1965 ASME Code, Section III, including the Winter 1966 Addenda. Three surveillance capsules are located at the 30°, 120°, and 300° azimuths of the core midplane. The capsule at 30° was removed during the 1985 outage which corresponds to 5.98 effective full power years (EFPY). The removed capsule contained 36 Charpy specimens, 8 tensile specimens, and 9 flux wires. The Charpy and tensile specimens were identified to be from the beltline region of the vessel, including plate, weld and heat affected zone materials. The construction and selection of surveillance specimens satisfy the requirements of Appendix H of 10 CFR 50.

A boiling water reactor vessel has an inherent pressure-temperature limitation when the reactor water level is within the normal range for power operation and the reactor pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case, Appendix G of 10 CFR 50 states that the pressure-temperature curve is limited by the closure flange regions that are highly stressed by the bolt preload. The minimum permissible temperature is 60°F above the initial RT_{NDT} of the flange and when the test pressure is above 20% of the hydrotest pressure, the permissible temperature should be 90°F above the initial RT_{NDT}. The proposed FitzPatrick pressure-temperature curves satisfy these requirements.

Another temperature limitation in the older boiling water reactors pertains to feedwater nozzles, which are prone to cracking. In the 1970's, the feedwater nozzles in BWR plants were found to have cracks due to feedwater leaking through the thermal sleeve and sparger. The nozzle cracking problem has since been resolved. Nevertheless, a pressure-temperature limit is required for heatup, cooldown, and critical operation to eliminate future cracking. The proposed pressure-temperature limit for the feedwater nozzle case is based on the curve that is contained in the existing TS.

At higher temperature and pressure ranges, the pressure-temperature limits are based on the material that has the largest shift between the adjusted RT_{NDT} and the initial RT_{NDT}. The plate was found to be the limiting material for operation up to 16 EFPY and has an adjusted RT_{NDT} of 86°F. Beyond 16 EFPY, the weld will be the limiting material due to the higher irradiation shift. The staff has determined that the three proposed pressure-temperature curves satisfy the safety margins as required by Appendix G of 10 CFR 50.

The upper shelf energy for the plate at end-of-life is predicted to be 58 ft-lb which exceeds the required 50 ft-lb of Appendix G. The adjusted RT_{NDT} at end-of-life for the weld is 104°F which is below the required 200°F. Based on this finding, the FitzPatrick reactor vessel satisfies the applicable 10 CFR 50 requirements.

In summary, the staff has determined that the licensee has adequate surveillance capsules for the reactor beltline materials inside of the reactor vessel and has an acceptable fracture toughness testing program for the irradiated materials. The licensee has appropriately applied Appendix G of 10 CFR 50 and Regulatory Guide 1.99, Revision 1, to develop the pressure-temperature limits. These limits are valid up to 16 EFPY and may be incorporated into the FitzPatrick TS.

Revised TS sections 3.6 and 4.6, Limiting Conditions for Operation and Surveillance Requirements, respectively, as well as the revised bases for these sections, have also been reviewed. The staff finds that the changes reflect the revised pressure-temperature limits and satisfy the guidelines in the Standard Technical Specifications for General Electric Boiling Water Reactors (NUREG-0123) and are, therefore, acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 22, 1987

PRINCIPAL CONTRIBUTOR:

J. Tsao