ATTACHMENT . to JPN-92-002

PROPOSED TECHNICAL SPECIFICATION CHANGES
REACTOR VESSEL HYDROSTATIC TESTING

(JPTS-91-014)

New York Power Authority

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

3.5 (cont'd)

a. From and after the date that the HPCI System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the Automatic Depressurization System, the Core Spray System, LPCI System, and Reactor Core Isolation Cooling System are operable.

b. If the requirements of 3.5.C.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hrs.

2. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $\leq$212°F with an inoperable component(s) as specified in 3.5.C.1 above.

3. The HPCI system is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.5 (cont'd)

a. When it is determined that the HPCI subsystem is inoperable the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuation logic shall be verified to be operable immediately. The RCIC system and ADS subsystem logic shall be verified to be operable daily thereafter.

3.5 (cont'd)

during such time, the HPCI System is operable.

2. If the requirements of 3.5.D.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 100 psig, within 24 hr.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.1.a and 3.5.1.b above, provided that reactor coolant temperature is $\leq$212°F and the reactor vessel is vented or reactor vessel head is removed.

4. The ADS is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.5 (cont'd)

2. A logic system functional test.

a. When it is determined that one valve of the ADS is inoperable, the ADS subsystem actuation logic for the operable ADS valves and the HPCI subsystem shall be verified to be operable immediately and at least weekly thereafter.

b. When it is determined that more than one relief/safety valve of the ADS is inoperable, the HPCI System shall be verified to be operable immediately.

3.5 (Cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. The RCIC System shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 150 psig and reactor coolant temperature is greater than 212°F except from the time that the RCIC System is made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding 7 days unless the system is made operable earlier provided that during these 7 days the HPCI System is operable.

2. If the requirements of 3.5.F cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hours.

3. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in 3.5.E.2 above, provided that reactor coolant temperature is $\leq$212°F.

4. The RCIC system is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.5 (Cont'd)

E. Reactor Core Isolation Cooling (RCIC) System

1. RCIC System testing shall be performed as follows provided a reactor steam supply is available. If steam is not available at the time the surveillance test is scheduled to be performed, the test shall be performed within ten days of continuous operation from the time steam becomes available.

Item	Frequency
a. Simulated Automatic Actuation (and Restart*) Test	Once/operating cycle
b. Pump Operability	Once/month
c. Motor Operated Valve Operability	Once/month
d. Flow Rate	Once/3 months
e. Testable Check Valves	Tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.
f. Logic System Functional Test	Once/operating cycle

* Automatic restart on a low water level signal which is subsequent to a high water level trip.

Amendment No. 40, 107, 130,

3.6 (cont'd)

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

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

Specifications 3.5.C, 3.5.D, 3.5.E and 3.6.E which would become effective because of an increase in reactor coolant temperature above 212°F or pressures above 100 and 150 psig are not required while conducting the RCS hydrostatic pressure and leakage tests between 212°F and 300°F provided all control rods are fully inserted.

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.

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 (cont'd)

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

   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.

4.6 (cont'd)

5. Not Used

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.

3.6 (cont'd)

B. Deleted

C. Coolant Chemistry

1. The reactor coolant system radioactivity concentration in water shall not exceed the equilibrium value of 3.1 μCi/gm of dose equivalent I-131. This limit may be exceeded, following a power transient, for a maximum of 48 hr. During this iodine activity transient the iodine concentrations shall not exceed the equilibrium limits by more than a factor of 10 whenever the main steamline isolation valves are open. The reactor shall not be operated more than 5 percent of its annual power operation under this exception to the equilibrium limits. If the iodine concentration exceeds the equilibrium limit by more than a factor of 10, the reactor shall be placed in a cold condition within 24 hr.

4.6 (cont'd)

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

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hr and analyzed for gross gamma activity.

   b. Isotopic analysis of a sample of reactor coolant shall be made at least once/month.

   c. A sample of reactor coolant shall be taken prior to startup and at 4 hr intervals during startup and analyzed for gross gamma activity.

   d. During plant steady state operation and following an offgas activity increase (at the Steam Jet Air Ejectors) of 10,000 μCi/sec within a 48 hr. period or a power level change of $\geq$20 percent of full rated power/hr reactor coolant samples shall be taken and analyzed for gross gamma activity. At least three samples will be taken at 4 hr intervals. These sampling requirements may be omitted whenever the equilibrium I-131 concentration in the reactor coolant is less than 0.007 μCi/ml.

3.6 (cont'd)

2.

a. From and after the date that the safety valve function of one safety/relief valve is made or found inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is made operable sooner.

b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.E.1 and 3.6.E.2 are not met, the reactor shall be placed in a cold condition within 24 hours.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is ≤212°F and the reactor vessel is vented or the reactor vessel head is removed.

5. The Safety and Safety/Relief Valves are not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted.

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

4. An annual report of safety/relief vave failures and challenges will be sent to the NRC in accordance with Section 6.9.A.2.b.

3.6 and 4.6 BASES (cont'd)

Fig. 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.

Fig. 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 Fig. 3.6-1, curves A, B and C, 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.

Specification 3.6.A.2 identifies four LCOs that become effective with increased reactor coolant temperature or pressure but are not in effect during the hydrostatic and leakage tests. This is necessary because, as reactor fluence increases, the minimum test temperature and pressure rises into ranges normally associated with startup or hot shutdown. RCS pressure and temperature are used throughout the Technical Specifications as a basis for establishing plant mode and system operability requirements. Some LCOs and restrictions cannot be satisfied during the test at elevated temperatures. For example, Specifications 3.5.C.1 and 3.5.E.1 require that HPCI and RCIC be operable when reactor pressure exceeds 150 psig and 212°F. HPCI and RCIC cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

Hydrostatic and leakage tests shall be terminated before the reactor coolant temperature exceeds 300°F. This temperature limit is based on providing a 50°F band for operating flexibility between the 300°F limit and the highest estimated minimum testing temperature at 32 EFPY (approximately 250°F).

The protection provided by LCOs applicable during cold shutdown plus the requirement that all control rods be fully inserted are adequate to ensure protection of public health and safety. The hydrostatic test is performed once every 10 years while the leakage test is performed after each refueling when conditions are similar to cold shutdown (i.e., after the reactor has been shutdown and decay heat and the energy stored in the core is very low). The consequences of accidents (small and large break LOCAs, MSLB, etc.) are bounded by analyses that assume full power operation. Specification 3.5.A requires the low pressure ECCS systems to be operable. Specifications 3.7.A, 3.7.B and 3.7.C require the containment, SGTS and secondary containment to be operable. Specifications 3.2.A, 3.2.B and Appendix B, Specification 3.8 require instrumentation that initiate containment, low pressure ECCS, SBGT and secondary containment be operable. Emergency power is required by Specification 3.9.B.

3.6 and 4.6 BASES (cont'd)

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 Section 3.2.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.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

Materials in the Reactor Coolant System are primarily 304 stainless steel and Zircaloy fuel cladding. The reactor water chemistry limits are established to prevent damage to these materials. Limits are placed on chloride concentration and conductivity. The most important limit is that placed on chloride concentration to prevent stress corrosion cracking of the stainless steel. The attached graph, Fig. 4.6-1, illustrates the results of tests on stressed 304 stainless steel specimens. Failures occurred at concentrations above the curve; no failures occurred at concentrations below the curve. According to the data, allowable chloride concentrations could be set several orders of magnitude above the established limit, at the oxygen concentration (0.2-0.3 ppm) experienced during power operation. Zircaloy does not exhibit similar stress corrosion failures.

However, there are various conditions under which the dissolved oxygen content of the reactor coolant water could be higher than 0.2-0.3 ppm, such as refueling, reactor startup, and hot standby. During these periods with steaming rates less

3.6 and 4.6 BASES (cont'd)

than 100,000 lb/hr, a more restrictive limit of 0.1 ppm has been established to assure the chloride-oxygen combinations of Fig. 4.6-1 are not exceeded. At steaming rates of at least 100,000 lb/hr, boiling occurs causing deaeration of the reactor water, thus maintaining oxygen concentration at low levels.

When conductivity is in its proper normal range, pH and chloride and other impurities affecting conductivity must also be within their normal ranges. When and if conductivity becomes abnormal, then chloride measurements are made to determine whether or not they are also out of their normal operating values. This is not necessarily the case. Conductivity could be high due to the presence of a neutral salt; e.g., $Na_2 SO_4$, which would not have an effect on pH or chloride. In such a case, high conductivity alone is not a cause for shutdown. In some types of water-cooled reactors, conductivities are, in fact, high due to purposeful addition of additives. In the case of BWR's, however, where no additives are used and where neutral pH is maintained, conductivity provides a very good measure of the quality of the reactor water. Significant changes therein provide the operator with a warning mechanism so he can investigate and remedy the condition causing the change before limiting conditions, with respect to variables affecting the boundaries of the reactor coolant, are exceeded. Methods available to the operator for correcting the condition include operation of the Reactor Cleanup System, reducing the input of impurities and placing the reactor in the cold shutdown condition. The major benefit of cold shutdown is to reduce the temperature dependent corrosion rates and provide time for the Reactor Water Cleanup System to reestablish the purity of the reactor coolant. During startup periods, which are in the category of less than 100,000 lb/hr, conductivity may exceed 2 mho/cm because of the initial evolution of gases and the initial evolution of gases and the initial addition of dissolved metals. During this period of time, when the conductivity exceeds 2 mho/cm (other than short-term spikes), samples will be taken to assure the chloride concentration is less than 0.1 ppm.

The conductivity of the reactor coolant is continuously monitored. The samples of the coolant which are taken every 96 hr will serve as a reference for calibration of these monitors and is considered adequate to assure accurate readings of the monitors. If conductivity is within its normal range, chlorides and other impurities will also be within their normal ranges. The reactor coolant samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses of the reactor coolant required by Specification 4.6.C.1 may be performed by a gamma scan.

D. Coolant Leakage

Allowable leakage rates of coolant from the Reactor Coolant System have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to make up Reactor Coolant System leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining system

SAFETY EVALUATION FOR
PROPOSED TECHNICAL SPECIFICATION CHANGES
REACTOR VESSEL HYDROSTATIC TESTING (JPTS-91-014)

I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed technical and editorial changes to the James A. FitzPatrick Technical Specifications are as follows:

A. HYDROSTATIC AND LEAKAGE TESTING CHANGES

Page 118, Specification 3.5.C.3

Insert this new specification with the following words:

> "The HPCI system is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted."

Page 120, Specification 3.5.D.4

Insert this new specification with the following words:

> "The ADS is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted."

Page 121, Specification 3.5.E.4

Insert this new specification with the following words:

> "The RCIC system is not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted."

Page 137, Specification 3.6.A.2

Insert the following sentence at the end of the existing Limiting Condition for Operation:

> "Specifications 3.5.C, 3.5.D, 3.5.E and 3.6.E which would become effective because of an increase in reactor coolant temperature above 212°F or pressures above 100 and 150 psig are not required while conducting the RCS hydrostatic pressure and leakage tests between 212°F and 300°F provided all control rods are fully inserted."

Page 143, Specification 3.6.E.5

Insert this new specification with the following words:

> "The Safety and Safety/Relief Valves are not required to be operable during hydrostatic pressure and leakage testing with reactor coolant temperatures between 212°F and 300°F and irradiated fuel in the reactor vessel provided all control rods are inserted."

Page 148, BASES 3.6 and 4.6 A

Insert the following paragraphs at the end of Bases A on page 148:

> "Specification 3.6.A.2 identifies four LCOs that become effective with increased reactor coolant temperature or pressure but are not in effect during the hydrostatic and leakage tests. This is necessary because, as reactor fluence increases, the minimum test temperature and pressure rises into ranges normally associated with startup or hot shutdown. RCS pressure and temperature are used throughout the Technical Specifications as a basis for establishing plant mode and system operability requirements. Some LCOs and restrictions cannot be satisfied during the test at elevated temperatures. For example, Specifications 3.5.C.1 and 3.5.E.1 require that HPCI and RCIC be operable when reactor pressure exceeds 150 psig and 212°F. HPCI and RCIC cannot be made operable during the test because piping normally filled with steam is filled with water during the test.
>
> Hydrostatic and leakage tests shall be terminated before the reactor coolant temperature exceeds 300°F. This temperature limit is based on providing a 50°F band for operating flexibility between the 300°F limit and the highest estimated minimum testing temperature at 32 EFPY (approximately 250°F).
>
> The protection provided by LCOs applicable during cold shutdown plus the requirement that all control rods be fully inserted are adequate to ensure protection of public health and safety. The hydrostatic test is performed once every 10 years while the leakage test is performed after each refueling when conditions are similar to cold shutdown (i.e., after the reactor has been shutdown and decay heat and the ener stored in the core is very low). The consequences of accidents (small and large b LOCAs, MSLB, etc.) are bounded by analyses that assume full power operation. Specification 3.5.A requires the low pressure ECCS systems to be operable. Specifications 3.7.A, 3.7.B and 3.7.C require the containment, SGTS and secondary containment to be operable. Specifications 3.2.A, 3.2.E and Appendix B, Specification 3.8 require instrumentation that initiate containment, low pressure ECCS, SBGT and secondary containment be operable. Emergency power is required by Specification 3.9.B.

B. EDITORIAL CHANGES

Page 137, Specification 3.6.A.2

Add a "29" to the list of amendment numbers at the bottom of this page.

Page 138, Specifications 3.6.A.5 and 4.6.A.5

Move Specification 3.6.A.5 from page 137 to the top of page 138.

Insert Specification 4.6.A.5 with the words, "Not Used," at the top of the second column.

Page 139, Specifications 3.6.B, 4.6.A.7, and 4.6.B

Move Specifications 3.6.B, 4.6.A.7, and 4.6.B from page 138 to the top of page 139.

Page 149, BASES 3.6 and 4.6 B and C

Move Bases Sections "B" and "C" from page 148 to the top of page 149.

Page 150, BASES 3.6 and 4.6 B and C

Move the last paragraph and the portion of the third paragraph beginning with "... than 100,000 lb/hr, a more restrictive ..." from page 149 to the top of page 150.

II. PURPOSE OF THE PROPOSED CHANGES

A. HYDROSTATIC AND LEAKAGE TESTING CHANGES

The proposed changes revise the Technical Specifications to permit hydrostatic pressure and leakage testing of the Reactor Coolant System (RCS) as required by Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Reference 3) at RCS temperatures exceeding 212°F. The hydrostatic test is performed every 10 years. The leakage test is performed frequently during plant life.

Specification 1.0 defines the hot shutdown mode as a condition when the reactor mode switch is in the shutdown position and the RCS temperature is above 212°F. The Technical Specifications require a number of systems, including emergency core cooling (ECCS), to be operable when the RCS temperature exceeds 212°F. The required hydrostatic pressure and inservice leak testing cannot be conducted without making some of these systems inoperable. The proposed changes will allow testing to proceed with inoperable systems.

B. EDITORIAL CHANGES

The application also makes two editorial changes. These include the addition of the number 29 to the list of amendments effecting page 137 to correct a typographical error from Amendment 113 (Reference 8) and the addition of the words "Not Used" on page 138 to clarify that there is no associated surveillance requirement.

III. SAFETY IMPLICATION OF THE PROPOSED CHANGES

A. HYDROSTATIC AND LEAKAGE TESTING CHANGES

Hydrostatic testing and system leakage testing of the Reactor Coolant System (RCS) is required by Section XI of the ASME B&PV code. NRC Generic Letter 88-11 (Reference 5) is used to calculate the reactor pressure vessel pressure and temperature (P-T) limits required for this test. The P-T curves defining these limits are periodically recalculated to consider the results of analyses of irradiated surveillance specimens to account for accumulated reactor fluence.

The current curves (Figure 3.6-1) require that these tests be conducted at RCS temperatures approaching 190°F. Because decay heat and mechanical heat used to heat the reactor coolant do not allow exact control, the operators require margin to maintain the test temperature between the minimum temperature limit and the maximum temperature limit of 212°F. That margin is small at this time. In addition, the Technical Specification curves will be revised to require temperatures that exceed 212°F as the accumulated fluence increases. An extrapolation from the minimum test temperature at 16 effective full power years (EFPY) indicates that minimum testing temperature will peak at about 250°F at 32 EFPY. The required test pressure is up to 1105 psig. These values define the conditions for hydrostatic pressure and leak testing after additional temperature margin is allowed to account for the control of heating.

Above 212°F, the Technical Specifications require a number of systems to be operable. Some systems cannot be made operable during testing. The current Technical Specifications were written in anticipation that the reactor would be going into operation when the temperature was raised to 212°F and requires the necessary complement of systems to be available. The proposed change would allow the RCS to be tested at temperatures above 212°F with a reduced complement of safety systems. The test duration, test frequency and limited system energy during testing do not require the same complement of systems as plant startup. This has been qualitatively evaluated by looking at the test conditions, technical specification requirements with the proposed change and the potential consequences of an accident during the test. This will not result in a substantial reduction in safety margin from the current Technical Specifications.

Test Conditions

The hydrostatic pressure test occurs once every 10 years. Leakage test typically occurs following a refueling outage and therefore has a frequency of about once per eighteen months. Recirculating pumps will be in operation and a water solid reactor coolant system will be maintained to control the necessary pressure and temperature. Reactor water makeup, pressure, and level will be controlled through the Control Rod Drive and Reactor Water Cleanup Systems.

During the test all control rods will be inserted to assure shutdown reactivity. The transition from below 212°F to a higher temperature represents an increase in the enthalpy of the reactor coolant while the core remains subcritical since the temperature increase is due to decay heat and mechanical heating. The decay heat in the core is at a minimum since testing is performed after refueling or maintenance activity with the reactor in a cold condition.

The pressure and temperature required for this test will remain within design limits. The required pressure is about 1105 psig (1.1 times the operating pressure) for the hydrostatic test and about 1005 psig (operating pressure) for the leakage test. At forty effective full power years, the temperature at these pressures was estimated to be about 250°F. This is increased by 50°F to account for testing margins.

Technical Specification Effects

The proposed change will revise the technical specifications to permit this testing. The technical specification requirements that will not be in effect during hydrostatic and leakage testing and their potential safety significance are discussed below:

- Specification 3.5.C: Requires that the High Pressure Coolant Injection (HPCI) System be operable when irradiated fuel is in the vessel, the reactor pressure is greater than 150 psig and the reactor coolant temperature is greater than 212°F. HPCI is not operable during the test due to the water solid condition of the plant. Since the core will be maintained subcritical and the operators can terminate the test if there is excessive leakage, which will be detected during the test, the safety function of this system will not be required during this test.

- Specification 3.5.D: Requires that the Automatic Depressurization (ADS) System be operable when irradiated fuel is in the vessel, reactor pressure is greater than 100 psig and prior to startup from the cold condition. The ADS has not been evaluated for operability in the water solid condition and may not be operable. Furthermore, safety/relief valves (SRV) may have to be gagged if test pressures exceed the SRV setpoints. Since the core will be maintained subcritical and the operators can terminate the test if there is excessive leakage, which will be detected during the test, the safety function of this system will not be required during this test.

- Specification 3.5.E: Requires that the Reactor Core Isolation (RCIC) System be operable under the same conditions as the HPCI system (Specification 3.5.C). RCIC is not operable during the test due to the water solid condition of the plant. Since the core will be maintained subcritical and the operators can terminate the test if there is excessive leakage, which will be detected during the test, the safety function of this system will not be required during this test.

- Specification 3.6.E: Requires the SRV's to be operable when the reactor coolant system exceeds atmospheric pressure and the temperature is greater than 212°F. These valves need not be operable as per Specification 3.5.D.

Potential Consequences

The consequences of plant design basis events while performing hydrostatic pressure and inservice leak testing above 212°F were qualitatively assessed. Maintaining primary containment assures that design basis events at power are more limiting. Under the hydrostatic pressure and leak test conditions, the worst case accident is a loss of coolant accident. A large break LOCA is

postulated to occur even though current industry leak before break analyses (Reference 6) have indicated that the pipe break will be preceded by a system leak. The potential consequences of the LOCA were qualitatively assessed to determine the effects on the primary system and the effects on the secondary containment environment.

The effects of a small or large break LOCA are considered to be bounded by the existing plant analyses. No core damage will result since the control rods are maintained fully inserted to maintain subcriticality margins. There will be adequate reactor coolant to remove the limited decay heat. With a small break, the RCS will depressurize while the operator terminates the test and initiates RHR cooling and or low pressure ECCS, as needed. Decay heat from the reactor core will tend to increase RCS pressure and temperature but the effect will be limited because of the decay time. With a large break, the reactor will depressurize immediately and all the low pressure ECCS systems with their initiating instrumentation will be available. The need to terminate the test due to leakage can be identified during leakage inspections.

The consequences of a break in primary containment are limited. The enthalpy added to bring the water in the RCS from 212°F and atmospheric pressure to 300°F and 1105 psig is significant. However, the primary containment is designed for the LOCA during power operation when the enthalpy is much greater. The systems to control containment atmosphere are available in both cases.

If a break were to occur in HPCI or RCIC steam lines outside of containment, the resulting localized effects are expected to be less severe than for high energy line breaks because there is much less energy. Although the low steam pressure isolation trips leads are lifted, operator action is available to terminate flow. Even if the operator did not terminate flow immediately, there is a limited amount of water available in the steam lines.

Primary containment integrity will be maintained during the test even though there is a requirement for personnel access. Personnel safety is not considered to be jeopardized. Industry leak before break analyses (Reference 6) indicates that there will be a small leak well before a crack could propagate into a large break. Inspectors will enter the containment and areas of secondary containment where pressurized piping is located in order to inspect for leakage. For the hydrostatic test, a 4 hour waiting period is provided to allow for detection of potential leakage concealed by insulation. The effects of significant steam or water leakage would be readily observable. Additionally, the duration of the tests are not sufficient for the leak to propagate into a larger break.

Radiological Consequences

In the LOCA evaluated in the FSAR, primary containment is designed to contain radioactive releases. During testing, primary containment is maintained. Other systems designed to restrict radiological release (e.g., secondary containment, SGTS) will be available. A LOCA during a hydrostatic or leakage test will result in a limited source term since there is no failed fuel. The LOCA during operation is therefore bounding.

For the steam line breaks outside containment, any release of radioactivity would be to the secondary containment. The potential radiological consequences are bounded by the existing plant analysis of a fuel assembly drop during refueling. During refueling, the primary containment is not maintained and a fuel rod drop accident, discussed in FSAR Section 14.6.1.4, is postulated

to result in the failure of 442 fuel rods for calculating dose consequences. The resultant offsite doses are less than 2% of the 10 CFR 100 limits. For the steam line break, there is no failed fuel and the same complement of safety systems used to limit dose consequences will be available.

These changes to the Technical Specifications do not alter the conclusions of the plant's accident analyses as documented in the FSAR or the NRC staff's SER.

B. EDITORIAL CHANGES

Editorial changes include the addition of a number 29 to the list of amendments and the addition of the words "Not Used" to clarify that there is no associated surveillance requirement. These changes have no safety significance.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Operation of the FitzPatrick 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 proposed amendment revisions involve no hardware changes, no changes to the operation of any systems or components, no changes to structures, and alters procedures only to the extent that the 212°F limit can be exceeded with certain systems inoperable. These systems are required for core cooling at high pressure. Any event requiring core cooling will rapidly depressurize the system. The increased temperature adds enthalpy to the reactor coolant during the test but the consequences of previously evaluated accidents envelope any potential events. The probability of an accident during testing is expected to increase by a minimal amount but this probability is still below that for operations. The test temperatures and pressures are still within system design limits. The test is required to demonstrate the pressure retaining capabilities of the RCS pressure boundary.

2. create the possibility of a new or different kind of accident from those previously evaluated.

   The proposed amendment revisions involve no hardware changes, no changes to the operation of any systems or components, no changes to structures, and alters procedures only to the extent that the 212°F limit can be exceeded with certain systems inoperable. The testing procedure will not change the test process but will allow increased temperature during testing to meet NRC guidance and allow margin to the minimum temperature limits.

3. involve a significant reduction in the margin of safety.

The proposed amendment revisions involve no hardware changes, no changes to the operation of any systems, no changes to structures, and alter procedures only to the extent that the 212°F limit can be exceeded with certain systems inoperable. Primary containment and most other systems required for plant transients and accidents are available. The core cooling function can be maintained with no change to the margin of safety. The additional enthalpy to the reactor coolant will reduce by a small amount the margin that existed during prior hydrostatic tests but remains within the envelope of previously evaluated plant conditions.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not adversely affect the ALARA or Fire Protection Program at the FitzPatrick plant, nor will the changes affect the environment. These changes are limited to increased primary system temperatures during a preexisting test. The testing process will not change and therefore can have no impact.

VI. CONCLUSION

These 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 type different from 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. James A. FitzPatrick Nuclear Power Plant Updated Final Safety Analysis Report, Sections 4.7, 6.4.1, and 14.6.1.4.

2. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER), dated November 20, 1972, and Supplements.

3. ASME Boiler and Pressure Vessel Code, Section XI, 1980 Edition through Winter 1981, Article IWB-5000.

4. Inservice Testing Program for James A. FitzPatrick Nuclear Power Plant, Second Inservice Interval, Revision 4, dated May 1, 1991.

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

6. NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Revision 0, dated August 1984.

7. NRC letter, D. LaBarge to J.C. Brons, dated February 13, 1991, (JAF-91-071) transmits Amendment 168.

8. NRC letter, H. Abelson to J.C. Brons, dated October 22, 1987, (JAF-87-252) transmits Amendment 113.

9. James A. FitzPatrick Nuclear Power Plant Operations Surveillance Test Procedure, ST-39H, "Reactor Vessel Operational Pressure Test (ISI)," Revision 15, dated June 8, 1990.

10. GE Report DRF 137-0010, "Implementation of Regulatory Guide 1.99 Revision 2 for the James A. FitzPatrick Nuclear Power Plant", June 1989.