ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING AUGMENTED INSERVICE INSPECTION OF MAIN STEAM AND FEEDWATER PIPING WELDS

(JPTS-89-013)

New York Power Authority

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

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

3.6 and 4.6 BASES (cont'd)

not required to be operable (reactor coolant temperature less than or equal to 212°F and the reactor vessel vented or the reactor vessel head removed). Permitting physics testing and operator training under these conditions would not place the plant in an unsafe condition.

F. Structural Integrity

A pre-service inspection of the ASME Code Class 1 components was performed after site erection to assure the system was free of gross defects. An initial inspection program as detailed in Appendix F of the FSAR was developed and based on an approved edition of the ASME Code.

The program has been expanded to include the requirements of later, approved ASME Code editions and addenda as far as practicable. The importance of these inspections is recognized, and efforts to develop practical new alternative methods of assuring plant inservice integrity will continue. This inspection program should assure the detection of problem areas well before they represent a significant impact on safety.

In addition, visual inspection in accordance with the approved ASME code will be made during periodic pressure and hydrostatic tests of critical systems. The inspection program specified encompasses the major areas of the vessel and piping system within the drywell. The inspection period is based on the observed rate of defect growth from fatigue studies sponsored by the AEC.

These studies show that thousand of stress cycles, at stresses beyond any expected to occur in a Reactor Coolant System, were required to propagate a crack. The test

ATTACHMENT II

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING AUGMENTED INSERVICE INSPECTION OF MAIN STEAM AND FEEDWATER PIPING WELDS

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I. DESCRIPTION OF THE PROPOSED CHANGES

This application for an amendment to the James A. FitzPatrick Technical Specifications deletes the augmented inservice inspection program imposed on the main steam and feedwater piping welds. The proposed changes to the Technical Specifications are:

A. Section 4.6.F.2, page 144; delete the following specification:

> 2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

B. Bases Section 3.6 and 4.6, page 153; delete the following paragraph:

> Several locations on the main steam lines and feedwater lines are not restrained to prevent pipe whip in the event of pipe failure at these locations. The physical layout within the drywell precludes restraints at these points. Unrestrained high stress areas have been identified in these lines where breaks could result in pipe whip such that the pipe could impact the primary containment wall. Augmented inservice inspection of these weld locations shall be performed during each inspection period.

II. PURPOSE OF THE PROPOSED CHANGES

Existing Specification 4.6.F.2 requires 100% of the carbon steel pipe welds inside the drywell on the Main Steam and Feedwater Systems (total of 34 welds) to be examined every 10 years. This augmented inservice inspection (ISI) frequency was required by the Atomic Energy Commission staff in their Safety Evaluation Report (Reference 1), because it was not practicable to backfit pipe whip restraints at these locations on the main steam and feedwater high energy piping. As required by the augmented ISI inspection program, all 34 welds (12 on main steam and 22 on feedwater) have been examined at least once. In all cases, no flaws were noted.

This proposed technical specification change will delete the requirement for augmented inservice inspection. Future inspections of the affected welds will be in accordance with the standard Fitzpatrick ISI program which implements the requirements of 10 CFR 50.55a and ASME Section XI (Reference 2). That is, 25% of the affected welds will be inspected every 10 years. The proposed change will reduce radiation exposures by approximately 20 rem during a 10 year ISI interval.

On October 27, 1987 (Reference 3), the NRC modified General Design Criteria 4 (GDC 4) in 10 CFR 50, Appendix A, by allowing the use of leak-before-break (LBB) technology to eliminate from plant design bases the dynamic effects associated with high energy pipe ruptures. The modified rule permits the removal of pipe whip restraints and jet impingement barriers in operating nuclear power plants. In general, the LBB technology uses fracture mechanics analyses to show that high energy pipe flaws (cracks) are detectable, either by normal ASME inservice inspection techniques or by leakage monitoring systems, long before the flaws can grow to critical or unstable sizes and lead to large break areas such as a double-ended guillotine pipe rupture.

A leak-before-break evaluation has been performed for the Authority by Structural Integrity Associates. A report of this evaluation, entitled "Evaluation of Leak-Before-Break for Feedwater and Main Steam Piping Inside Containment at the James A. Fitzpatrick Nuclear Power Plant," is enclosed as Attachment III. This evaluation concludes that the main steam and feedwater piping systems comply with the general criteria of NUREG-1061, Volume 3 (Reference 4) and that augmented inservice inspections of weld locations which are unrestrained against postulated pipe breaks are not necessary.

III. IMPACT OF THE PROPOSED CHANGES

Fracture mechanics analyses (Attachment III), coupled with leak detection systems and ASME Section XI ISI, demonstrate that the probability of a main steam or feedwater pipe rupture(s) is extremely low. The leak-before-break evaluation (Attachment III) was performed on the 24 inch main steam piping and the 12.75 inch and 18 inch feedwater piping. Weld locations with the least favorable combination of high stress and material properties were analyzed in each pipe size for bounding considerations. The leak-before-break evaluation is based on detecting pipe leaks, through leak detection systems or inservice examinations, and that detectable cracks (flaws) are inherently stable (i.e., will not rupture). Indirect pipe failure mechanisms such as water hammer, erosion/corrosion, and fire are also evaluated.

Leak detection

The calculations assumed a leak detection limit of 5 gpm which is consistent with the design basis of the Fitzpatrick plant's leak detection systems (FSAR Section 4.10 and Technical Specification 3.6.D). The lower bound leak detection capability is generally considered to be 0.5 gpm which provides a leak detection margin of 10 as specified in NUREG-1061, Volume 3.

A long part-through-wall flaw which is detectable by ultrasonic means is bounded by a 5 gpm leaking crack. Fatigue crack growth analyses of 360° part-through-wall cracks was performed to assess the margin against rupture for pipes with long subsurface flaws. The results show that an assumed initial flaw with a depth of 15% of the pipe wall will not grow to a depth exceeding the critical flaw depth in 40 years.

Crack stability

A postulated through-wall flaw which will leak at 5 gpm can be doubled in length and will remain stable under normal operating plus safe shutdown earthquake loads.

A loading safety factor of 1.4 is maintained for the highest stressed (limiting) location in each pipe size as specified in NUREG-1061, Volume 3.

Other mechanisms

Water hammer is not expected to have an adverse effect on the integrity of the main steam and feedwater piping inside the containment. A review of past ISI inspection reports notes that there have been no water hammer induced pipe support deficiencies on the main steam and feedwater systems inside containment.

An erosion/corrosion inspection program has been implemented at the Fitzpatrick plant. The feedwater system is included in this program to detect wall thinning.

The drywell is inerted with nitrogen gas during power operation. Postulated fires can not occur on or near the main steam and feedwater piping within the containment.

The carbon steel welds on the main steam and feedwater piping inside containment are not susceptible to intergranular stress corrosion cracking. All 34 welds have been examined at least once, and no flaws or defects have been noted.

The proposed technical specification changes are administrative in nature. They do not involve any physical modification to the plant; nor do they introduce any new failure modes. The changes reduce the frequency of weld inspections and eliminate dynamic effects from the design basis of main steam and feedwater high energy piping.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

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

1. involve a significant increase in the probability or consequences of an accident previously evaluated. A leak-before-break evaluation was performed on the 24 inch main steam piping and the 12.75 inch and 18 inch feedwater piping. Weld locations with the least favorable combination of high stress and poor material properties were analyzed in each pipe size for bounding considerations. This evaluation concludes that the main steam and feedwater piping systems comply with the general criteria of NUREG-1061, Volume 3 and that augmented inservice inspections of the carbon steel pipe welds inside the drywell on the main steam and feedwater piping are not necessary. These fracture mechanics analyses, coupled with leak detection systems and ASME Section XI ISI, demonstrate that the probability of main steam and feedwater piping rupture(s) remains extremely low. The proposed changes are administrative in nature and can not increase the consequences of postulated accidents.

2. create the possibility of a new or different kind of accident from those previously evaluated. The proposed changes are administrative in nature. They do not involve any physical modification to the plant; nor do they introduce any new failure modes. The changes reduce the frequency of weld inspections and eliminate dynamic effects from the design basis of main steam and feedwater high energy piping.

3. involve a significant reduction in the margin of safety. The application of leak-before-break technology to exclude dynamic effects from the design basis of main steam and feedwater high energy piping is allowed by 10 CFR 50, Appendix A, General Design Criterion 4. Fracture mechanics analyses, coupled with leak detection systems and ASME Section XI ISI, demonstrate that the probability of main steam and feedwater piping rupture(s) is extremely low. Any uncertainties associated with flaw geometry, analytical procedures, or leak detection are conservatively accounted for in accordance with NUREG-1061, Volume 3. Significant safety margins are applied to leak detection, piping loads, and leakage crack sizes, such that the margin between the leakage crack size and the critical (unstable) crack size is a factor of two.

   The proposed technical specification changes significantly reduce worker radiation exposures with an insignificant impact on offsite risk.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

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

VI. CONCLUSION

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

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

b. will not create the possibility of 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. involve no significant hazards consideration, as defined in 10 CFR 50.92.

VII. REFERENCES

1. USAEC "Safety Evaluation of the James A. FitzPatrick Nuclear Power Plant" (SER), dated November 20, 1972, pages 5-4 and 5-5.

2. ASME Boiler & Pressure Vessel Code, Section XI, 1974 Edition through the Summer 1975 Addenda.

3. Federal Register, Volume 52, No. 207, pages 41288 - 41295, Final rule amending General Design Criterion 4, October 27, 1987.

4. NUREG-1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," November, 1984.

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

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

7. Structural Integrity Associates, Inc. Report No. SIR-86-033, Revision 1, "Evaluation of Leak-Before-Break for Feedwater and Main Steam Piping Inside Containment at the James A. Fitzpatrick Nuclear Power Plant," April, 1988.

8. NSAC 141, Nuclear Safety Analysis Center, "Lead Plant Application of Leak-Before-Break to High Energy Piping," January, 1989.

9. NSAC 114, Nuclear Safety Analysis Center, "Applying Leak-Before-Break to High Energy Piping," November, 1987.

10. EPRI NP-4991, Electric Power Research Institute," Application of the Leak-Before-Break Approach to BWR Piping," December 1986.