ATTACHMENT I TO JPN-84-64

PROPOSED TECHNICAL SPECIFICATION

CHANGE

RELATED TO

REACTOR COOLANT LEAKAGE DETECTION

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333

3.6 (cont'd)

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 lb/hr and during reactor shutdowns.

Conductivity	5 μmho/cm
Chloride ion	0.5 ppm

5. If Specification 3.6.C cannot be met, the reactor shall be placed in a cold condition within 24 hours.

D. Coolant Leakage

1. Anytime irradiated fuel is in the reactor vessel and the reactor coolant temperature is above 212°F, the reactor coolant leakage into the primary containment shall be limited to:

   a. 5 gpm unidentified leakage

   b. 2 gpm increase in unidentified leakage within any 24 hour period. (This limitation shall apply only after a period of 24 hours at operating pressure.)

   c. The total reactor coolant leakage into the primary containment shall not exceed 25 gpm.

2. With any reactor coolant system leakage greater than any one of the limits specified in 3.6.D.1.a or 3.6.D.1.c above, the leakage rate shall be reduced within these limits

4.6 (cont'd)

4.6.D Coolant Leakage

Reactor coolant leakage rate inside the primary containment shall be monitored and recorded once every 4 hours utilizing the Primary Containment Sump Monitoring System (equipment drain sump monitoring and floor drain sump monitoring).

3.6 (cont'd)

4.5 (cont'd)

within 4 hours or the reactor shall be in at least the hot standby condition within the following 12 hours and in cold condition within the next 24 hours.

3. If the increase in unidentified leakage as specified in 3.6.D.1.b is exceeded, the source of the leakage shall be identified within 4 hours or the reactor shall be in at least hot standby condition within the next 12 hours and in cold condition within the following 24 hours.

4. The primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring and Floor Drain Sump Monitoring) and the Primary Containment Atmosphere Monitoring System (Gaseous and Particulate) shall be operable during reactor power operation.

3.6 (cont'd)

5. With the Primary Containment Sump Monitoring System (Equipment Drain Sump Monitoring or Floor Drain Sump Monitoring) inoperable, restore the system to operable status within 24 hours or immediately initiate an orderly shutdown and be in at least hot standby condition within the next 12 hours and in cold condition within the following 24 hours.

6. With the Primary Containment Atmosphere Radioactivity Monitoring System (gaseous) or the Primary Containment Atmosphere Radioactivity Monitoring System (particulate) inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours. Otherwise be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

4.6 (cont'd)

3. Drywell Continuous Atmosphere Radioactivity Monitoring System instrumentation shall be functionally tested and calibrated as specified in Table 4.6.2.

ATTACHMENT II TO JPN-84-64

SAFETY EVALUTION

RELATED TO

REACTOR COOLANT LEAKAGE DETECTION

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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Section I - Description of Changes

The proposed changes to the James A. FitzPatrick (JAF) Technical Specifications are included as Attachment I, and amend Section 3.6.D (Coolant Leakage) on pages 141, 141a and 142. The changes impose stricter limitations on leakage rates from unidentified sources inside the primary containment.

Section II - Purpose of Changes

During a recent FitzPatrick plant outage (started March 1, 1984), the Authority treated eleven recirculation system welds with Induction Heating Stress Improvements (IHSI) methods to demonstrate the benefits of IHSI as an IGSCC (Inter Granular Stress Corrossion Cracking) countermeasure. During pre-IHSI inspections, a single weld showed linear crack indications (Reference g).

In Reference (f), the NRC staff informed the Authority that stricter limitations were being imposed on leakage rates from unidentified sources for boiling water reactors where inspections have uncovered evidence of IGSCC. Reference (f) further requested that the Authority submit an updated version of proposed Technical Specifications previously proposed to address Revision 1 to NUREG-0313 (Reference e).

The changes included as part of this amendment application respond to this request.

Section III - Impact of the Change

The changes to the FitzPatrick Technical Sepcifications will not alter the conclusions of either the FSAR or SER accident analysis.

The Authority considers that this proposed amendment can be classified as not likely to involve significant hazards considerations since the change constitutes additional limitations not presently included in the Technical Specifications. In particular, these new limitations will impose additional limiting conditions for operation. This is clearly in keeping with example (ii) included in Federal Register, Vol. 48 No. 67 dated April 6, 1983 page 14870, (Examples of Amendments that are Considered not Likely to Involve Significant Hazards Consideration which states: "A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.")

Section IV - Implementation of the Change

Implementation of these changes, as proposed, will not impact the ALARA or fire protection programs at FitzPatrick, nor will the changes impact the environment.

Section V - Conclusion

The incorporation of these modifications: a) will not change the probability nor the consequence of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report; b) will not increase the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report; c) will not reduce the margin of safety as defined in the basis for any Technical Specification; d) does not constitute an unreviewed safety question, and e) involves no significant hazards consideration, as defined in 10 CFR 50.92.

Section VI - References

(a) James A. FitzPatrick Nuclear Power Plant Final Safety Analysis Report

(b) Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission in the Matter of Power Authority of the State of New York, James A. FitzPatrick Nuclear Power Plant dated March 4, 1970 as amended.

(c) PASNY July 31, 1981 letter (JPN-81-54), J.P. Bayne to T.A. Ippolito, regarding implementation of NUREG-0313, Revision 1.

(d) NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", July 1980.

(e) PASNY September 28, 1981 letter (JPN-81-76), J.P. Bayne to T.A. Ippolito, regarding proposed changes to the Technical Specifications related to the implementation of NUREG-0313, Rev.1.

(f) NRC April 13, 1984 letter, D.B. Vassalo to J.P. Bayne, regarding intergranular stress corrosion cracking (IGSCC) in the recirculation piping system.

(g) NYPA March 9, 1984 letter (JPN-84-16), J.P. Bayne to D.B. Vassallo, regarding recirculation piping flaw indication.

(h) NYPA letter (JAFP-83-0769), C.A. McNeill to T.E. Murley, dated July 22, 1983.

(i) PASNY January 6, 1978 letter (JNRC-78-1), G.T. Berry to R.W. Reid, regarding review of reactor coolant system pressure boundary piping susceptible to stress corrosion cracking.

(j) NRC February 26, 1981 letter, D.G. Eisenhut to all BWR licensees regarding implementation of NUREG-0313, Revision 1 (Generic Letter No. 81-04).

(k) NRC September 29, 1977 letter regarding use of type or 304 and 306 austenitic stainless steel in reactor coolant pressure boundary.