ATTACHMENT I TO JPN-85-36

Proposed Changes to the Technical Specifications Regarding Reduction of Emergency Diesel Generator Tests (PTS-84-23)

New York Power Authority

James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333

3.5 (cont'd)

2. From and after the date that one of the Core Spray Systems is made or found inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the system is made operable earlier, provided that during the 7 days all active components of the other Core Spray System and the LPCI System shall be operable.

3. The LPCI mode of the RHR System shall be operable whenever irradiated fuel is in the reactor and prior to reactor startup from a cold condition, except as specified below.

   a. From the time that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the pump is made operable earlier provided that during such 7 days the remaining active components of the LPCI, containment spray mode, and all active components of both Core Spray Systems are operable.

4.5 (cont'd)

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, and the LPCI System, shall be demonstrated to be operable immediately. The remaining Core Spray System shall be demonstrated to be operable daily thereafter.

3. LPCI System testing shall be as specified in 4.5.A.1a, b, c, d, f and g except that three RHR pumps shall deliver at least 23,100 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

   a. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI, containment spray subsystem and both Core Spray Systems required for operation shall be demonstrated to be operable immediately, and the remaining RHR pumps shall be demonstrated to be operable daily thereafter.

3.5 (Cont'd)

b. From the time that the LPCI mode is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the LPCI mode is made operable earlier provided that during these 7 days all active components of both Core Spray Systems and the containment spray subsystem (including two RHR pumps) shall be operable.

c. When the reactor water temperature is greater than 212°F, the motor operator for the RHR cross-tie valve (MOV20) shall be maintained disconnected from its electric power source. It shall be maintained chain-locked in the closed position. The manually operated gate valve (10-RHR-09) in the cross-tie line, in series with the motor operated valve, shall be maintained locked in the closed position.

4.a. The reactor shall not be started up with the RHR System supplying cooling to the fuel pool.

b. The RHR System shall not supply cooling to the spent fuel pool when the reactor coolant temperature is above 212°F.

4.5 (Cont'd)

b. When it is determined that the LPCI mode is inoperable, both Core Spray Systems, and the containment spray subsystem shall be demonstrated to be operable immediately and daily thereafter.

c. The power source disconnect and chain lock to motor operated RHR cross-tie valve, and lock on manually operated gate valve shall be inspected once each operating cycle to verify that both valves are closed and locked.

3.5 (Cont'd)

5. All recirculation pump discharge valves and bypass valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

6. If the requirements of 3.5.A cannot be met, the reactor shall be placed in the cold condition within 24 hrs.

B. CONTAINMENT COOLING SUBSYSTEM MODE (OF THE RHR SYSTEM)

1. Both subsystems of the containment cooling mode, each including two RHR, one ESW pump and two RHRSW pumps shall be operable whenever there is irradiated fuel in the reactor vessel, prior to startup from a cold condition, and reactor coolant temperature ≥ 212°F except as specified below:

2. Continued reactor operation is permissible for 30 days with one spray loop inoperable and with reactor water temperature greater than 212°F.

4.5 (Cont'd)

5. All recirculation pump discharge and bypass valves shall be 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.

B. CONTAINMENT COOLING SUBSYSTEM MODE (OF THE RHR SYSTEM)

1. Subsystems of the containment cooling mode are tested in conjunction with the test performed on the LPCI System and given in 4.5.A.1.a, b, c, and d. Residual heat removal service water pumps, each loop consisting of two pumps operating in parallel, will be included in testing, supplying 8,000 gpm. The Emergency Service Water System, each loop of which consists of a single operating emergency service water pump of 3,700 gpm will be tested in accordance with Section 4.11D.

During each five-year period, an air test shall be performed on the containment spray headers and nozzles.

2. When it is determined that one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above are inoperable, the remaining redundant active components of the containment cooling mode subsystems shall be demonstrated to be operable immediately and daily thereafter.

3.5 (Cont'd)

3. Should one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining active components of the containment cooling mode are operable.

4. Should one of the containment cooling subsystems become inoperable, continued reactor operation is permissible for a period not to exceed 7 days, unless such subsystem is sooner made operable provided that during such 7 days all active components of the other containment cooling subsystem are operable.

5. If the requirements of 3.5.B cannot be met, the reactor shall be placed in a cold condition within 24 hr.

6. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature < 212°F with an inoperable component(s) as specified in 3.5.B above.

4.5 (Cont'd)

3. When one containment cooling subsystem loop becomes inoperable, the operable loop shall be demonstrated to be operable immediately and daily thereafter.

3.5 (cont'd)

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C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

1. The HPCI System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition, except as specified below:

4.5 (cont'd)

C. HIGH PRESSURE COOLANT INJECTION (HPCI SYSTEM)

Surveillance of HPCI System shall be performed as follows provided 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 10 days of continuous operation from the time steam becomes available.

1. HPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, f, and g except that the HPCI pump shall deliver at least 4,250 gpm against a system head corresponding to a reactor vessel pressure of 1120 psig to 150 psig.

3.5 BASES (cont'd)

the RHR System in conjunction with the Core Spray System provides adequate cooling for break areas of approximately 0.2 sq. ft. up to and including the double-ended reactor recirculation line break without assistance from the high pressure Emergency Core Cooling Systems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 8. Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the Core Spray and LPCI Systems constitute a 1-out-of-2 systems; however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in Reference 8 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgement.

Should one Core Spray System become inoperable, the remaining Core Spray and the entire LPCI System are available should the need for core cooling arise. To assure that the remaining Core Spray and LPCI Systems are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves. Based on judgements of the reliability of the remaining systems, i.e., the Core Spray and LPCI, a seven-day repair period was obtained.

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment are available. Three RHR pumps in conjunction with the Core Spray System will perform the core cooling function. Because of the availability of the majority of the

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3.9 (cont'd)

3. From and after the time that one of the Emergency Diesel Generator Systems is made or found to be inoperable, continued reactor operation is permissible for a period not to exceed 7 days provided that the two incoming power sources are available and connected to the emergency bus associated with the inoperable Emergency Diesel Generator System and that the remaining Diesel Generator System is operable. At the end of the 7-day period, the reactor shall be placed in a cold condition within 24 hours, unless one or both diesel generator systems are made operable sooner.

4. When both Emergency Diesel Generator Systems are made or found to be inoperable, a reactor shutdown shall be initiated within two hours and the reactor placed in a cold condition within 24 hours after initiation of shutdown.

3. The emergency diesel generator system instrumentation shall be checked during the monthly generator test.

4. Once each operating cycle, the conditions under which the Emergency Diesel Generator System is required will be simulated to demonstrate that the pair of diesel generators will start, accelerate, force parallel, and accept the emergency loads in the prescribed sequence.

5. Once within one hour and at least once per twenty-four hours thereafter while the reactor is being operated in accordance with Specifications 3.9.B.1, 3.9.B.2, and 3.9.B.3, the availability of the operable Emergency Diesel Generators shall be demonstrated by manual starting and force paralleling where applicable.

ATTACHMENT II TO JPN-85-36

<u>Safety Evaluation For
Proposed Changes To The Technical Specifications
Regarding Reduction Of Emergency Diesel Generator
Tests (PTS-84-23)</u>

New York Power Authority

James A. FitzPatrick Nuclear Power Plant

Docket No. 50-333

Section I - Description of the Changes

The proposed changes to the Technical Specifications are shown in Attachment I to the Application for Amendment. This proposed Amendment revises Sections 3.5.A.2, 3.5.A.3(a), 4.5.A.2 and 4.5.A.3(a) (page 114), 3.5.A.3(b) and 4.5.B.3(b) (page 115), 3.5.B.4 and 4.5.B.3 (page 116), 4.9.B.5 (page 217) and the Bases on page 126. The changes on these pages delete the requirement to demonstrate the operability of the emergency diesel generators when subsystems of the Emergency Core Cooling or Containment Cooling System are declared inoperable. The proposed changes also remove the diesel generators from the Limiting Conditions for Operation (LCO) for these systems and reduces the surveillance test requirements. The proposed Amendment also includes administrative changes on pages 115a, 116 and 117 of the Technical Specifications.

Section II - Purpose of the Changes

The proposed changes would delete the requirement that "the emergency diesel generators shall be operable", from the Limiting Conditions for Operation (LCO) which are applicable when the following systems are declared inoperable: Core Spray (CS); Low Pressure Coolant Injection (LPCI) mode of Residual Heat Removal (RHR); and Containment Cooling. This deletion has been proposed, because the LCO for the emergency diesel generators are specified in Section 3.9.B of the Technical Specifications under Emergency A-C Power System. This is also in accordance with the Standard Technical Specifications. In addition, Section 3.0.E of the FitzPatrick Technical Specifications states that reactor operation is governed by the time limits of the Action Statement of the LCO for the emergency power source; and, not by the Action Statement of the individual system that is determined to be inoperable due to the inoperability of its emergency power source.

The proposed changes would also delete the surveillance test requirements for the emergency diesel generators when the above mentioned systems are declared inoperable. Two of the FitzPatrick diesels have a reliability factor of 1.0 and the other two have a reliability factor of 0.99. These reliability factors have been determined in accordance with Regulatory Guide 1.108 'Periodic Testing of Diesel Generator Units used as On-Site Electric Power Systems of Nuclear Power Plants'. In addition, the NRC staff has concluded that excessive testing of diesel generators results in premature degradation of diesel engines (Reference 1). These changes have been proposed because unnecessary testing would reduce their reliability.

In the current FitzPatrick Technical Specifications, the diesels are required to be tested every eight hours when reserve power is not available from either one or both off-site sources or when one of the diesel generators is declared inoperable. The proposed amendment would also change this requirement for testing from eight hours to every twenty-four hours and thereby reduce the number of diesel surveillance tests.

Section III - Impact of the Changes

The proposed changes would reduce the number of emergency diesel generator tests by approximately 42%. This estimated reduction is based on the surveillance test data from 1981 to 1984.

The proposed changes do not change any system or subsystem and will not alter the conclusions of either the FSAR or SER accident analyses.

Based on the discussions in Section II, operation of the FitzPatrick plant in accordance with the proposed amendment would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated because:

   a) the FitzPatrick diesel generators have a high degree of reliability;

   b) a reduction in the number of surveillance tests would avoid premature degradation of the diesel generators thereby maintaining their high level of reliability;

   c) the operation of safety-related equipment is not affected by the proposed changes; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated, because the diesel generators are a part of the emergency A-C power system which is used as a back-up system and no new mode of failure is introduced; or

(3) involve a significant reduction in margin of safety because a reduction of unnecessary surveillance tests would prevent premature degradation of the diesel generators thereby maintaining their high level of reliability.

Section IV - Implementation of the Changes

Implementation of the 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 changes:

   a) Will not change the probability nor the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report;

b) Will not increase the possibility of an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report;

c) Will not reduce the margin of safety as defined in the basis for any Technical Specifications;

d) Does not constitute an unreviewed safety question as defined in 10 CFR 50.59; and

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

Section VI - References

1) NRC Generic Letter 84-15 "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability".

2) NYPA letter, C. A. McNeill, Jr. to D. G. Eisenhut, dated January 17, 1985 (JPN-85-04).

3) James A. FitzPatrick Nuclear Power Plant Final Safety Analysis Report (FSAR), Rev 2, July 1984, Section 8.6

4) James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER)