ATTACHMENT I

PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING MAIN STEAM LINE HIGH RADIATION MONITOR TRIP LEVEL SETPOINT

(JPTS-89-023)

New York Power Authority

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

TABLE 3.1-1 (cont´d)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

NOTES OF TABLE 3.1-1 (cont´d)

14. The APRM flow biased high neutron flux signal is fed through a time constant circuit of approximately 6 seconds. The APRM fixed high neutron flux signal does not incorporate the time constant, but responds directly to instantaneous neutron flux.

15. This Average Power Range Monitor scram function is fixed point and is increased when the reactor mode switch is place in the Run position.

16. *During the proposed Hydrogen Addition Test, the background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, within 24 hours prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to $\leq$ three times the anticipated radiation levels. Upon completion of the Hydrogen Addition Test, the setpoint will be readjusted to its prior setting within 24 hours.

17. This APRM Flow Referenced Scram setting is applicable to two loop operation. For one loop operation this setting becomes $S \leq (0.66W+54\%-0.66\Delta W)(FRP/MFLPD)$

    Where:

    ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

* This specification is in effect only during Operating Cycle 10.

TABLE 3.2-1 (Cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.
2. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.
   - A. Initiate an orderly shutdown and have the reactor in cold shutdown condition in 24 hours.
   - B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.
   - C. Isolate Reactor Water Cleanup System.
   - D. Isolate shutdown cooling.
3. Deleted
4. Deleted
5. Two required for each steam line.
6. These signals also start SBGTS and initiate secondary containment isolation.
7. Only required in run mode (interlocked with Mode Switch).
8. Bypassed when mode switch is not in run mode and turbine stop valves are closed.
9. The trip level setpoint will be maintained at $\leq$ 3 times normal rated full power background. See note 16 to Table 3.1-1 for re-setting trip level setpoint just prior to and following the Hydrogen Addition Test.

ATTACHMENT II

SAFETY EVALUATION FOR PROPOSED TECHNICAL SPECIFICATION CHANGES REGARDING MAIN STEAM LINE HIGH RADIATION MONITOR TRIP LEVEL SETPOINT

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I. DESCRIPTION OF THE PROPOSED CHANGES

The proposed changes to the James A. FitzPatrick Technical Specifications revise Note 16 of Table 3.1-1, "Reactor Protection System (SCRAM) Instrumentation Requirement," on page 43a and Note 9 of Table 3.2-1, "Instrumentation That Initiates Primary Containment Isolation," on page 65. These notes were incorporated into the Technical Specifications as part of Amendment 90 and were applicable only during Cycle 7. They are being revised to make them applicable during Cycle 10. The changes are as follows:

1. Table 3.1-1 on page 43a
   - Note 16 - Revise Note 16 to read as follows:

     "*During the proposed Hydrogen Addition Test, the background radiation level will increase by approximately a factor of 5 for peak hydrogen concentration. Therefore, within 24 hours prior to performance of the test, the Main Steam Line Radiation Monitor Trip Level Setpoint will be raised to $\leq$ three times the anticipated radiation levels. Upon completion of the Hydrogen Addition Test, the setpoint will be readjusted to its prior setting within 24 hours."

   - Footnote to Note 16 - Replace "Operating Cycle 7" with "Operating Cycle 10."

2. Table 3.2-1 on page 65

   Note 9 - Revise the last sentence to read as follows:

   "See note 16 to Table 3.1-1 for re-setting trip level setpoint just prior to and following the Hydrogen Addition Test."

In addition, changes incorporated as part of Amendment 90 on pages 33, 41a, 57, and 64 are to be considered applicable to this package.

II. PURPOSE OF THE PROPOSED CHANGES

The purpose of the changes is to avoid spurious trips of the main steam line radiation monitor during testing of the in-core stress corrosion monitoring system. During this test, hydrogen added to the reactor coolant reduces the concentration of oxygen in the coolant water and increases the Nitrogen-16 carryover in the steam. This results in a higher background radiation level seen by the main steam line radiation monitor, which would be above the existing trip setpoint. By raising the setpoint of the main steam line high radiation monitor trip level for the duration of the in-core stress corrosion monitoring system testing, spurious trips of the reactor can be avoided.

III. IMPACT OF THE PROPOSED CHANGES

The main steam line radiation monitors have only a single design basis which is to initiate a reactor scram and isolate the main steam lines upon detecting high radiation, caused by gross fission product release during a control rod drop accident (CRDA). FSAR Section 14.6.1.2 states the results of a CRDA are more severe at power levels less than 10%. The testing of the in-core stress corrosion monitoring system will, however, only be conducted at power levels greater than 50%. If, due to a recirculation pump trip or any other unanticipated power reduction event, the reactor power decreases to below 20% of rated power during testing, control rod withdrawal will be prohibited administratively until the necessary readjustment is made to the trip setpoint. By performing the testing only at power levels greater than 50% and by readjusting the trip setpoints should the power levels drop below 20%, the possibility of CRDA occurring which would have more severe results than those already analyzed is prevented.

Parametric studies utilizing the conservative GE excursion model provided in NEDO-10527 (Reference 5) indicate that the maximum peak fuel enthalpy for a dropped control rod of maximum worth above 20% of rated power is less than 120 calories per gram. The fuel cladding failure threshold is 170 calories per gram. Consequently, the conservatively calculated peak fuel enthalpy of less than 120 calories per gram for a CRDA above 20% of rated power provides a significant margin of safety.

IV. EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

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

1. involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change to the setpoint of the main steam line high radiation monitor trip level does not involve an increase in the probability or consequences of an accident previously evaluated, as the proposed test of the in-core stress corrosion monitoring system would be conducted only at power levels greater than 50%. Above 20% of rated power, there is significant margin between the calculated peak fuel enthalpy and the fuel cladding failure threshold enthalpy. Should power levels drop below 20%, the trip setpoint will be readjusted to the original setting. This will ensure the trip setpoint is at the original setting for power levels below 10% of rated power for which the CRDA results become more severe .

2. create the possibility of a new or different kind of accident from those previously evaluated. The changes do not create the possibility of a new or different kind of accident previously evaluated, because the only function of these monitors is to detect gross fission product release in the event of a CRDA. Below 20% of rated power, the monitors would be at their original setting. Above 20% of rated power, there will be a significant margin to the fuel cladding failure threshold.

3. involve a significant reduction in the margin of safety. The changes do not involve a significant reduction in the margin of safety because the monitor setpoint will only be changed above 20% of rated power. Above 20% of rated power, a significant margin of safety will still be provided.

V. IMPLEMENTATION OF THE PROPOSED CHANGES

Implementation of the proposed changes will not impact:

1. Radiation/ALARA considerations

   Normal radiation and ALARA practices and procedures will be in effect during the course of the test. Appropriate approved access controls will be implemented for areas subject to the higher radiation levels that result from the test. Dose rate surveys will be conducted and radiation levels will be monitored in order to comply with ALARA requirements.

2. Fire Protection

   There will be no significant impact on Fire Protection.

3. Environment

   There will be no significant impact on the environment.

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 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; and

d. involve 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 7.2.3.6, 7.3.4.8, 7.12 and 14.6.1.2.

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

3. R. C. Stirn, et al. Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 1 Multiple Enrichment Cores with Axial Gadolinium, General Electric Company, July, 1972 (NEDO-10527, Supplement 1).

4. R. C. Stirn, et al. Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores, General Electric Company, January, 1973 (NEDO-10527, Supplement 2).

5. R. C. Stirn, et al. Rod Drop Accident Analysis for Large Boiling Water Reactors, General Electric Company, March, 1972 (NEDO-10527).