UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

By letter dated January 9, 1992, the Power Authority of the State of New York (the licensee) submitted a request for changes to the James A. FitzPatrick Nuclear Power Plant, Technical Specifications (TS). The requested changes would 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 at RCS temperatures exceeding 212 degrees F. During this testing, the High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), and the Automatic Depressurization System (ADS)/Safety Relief Valves (SRV) are not required to be operable.

2.0 BACKGROUND

Hydrostatic testing and system leakage testing of the Reactor Coolant System is required by Section XI of the ASME B&PV code. NRC Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," 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 require that these tests be conducted at RCS temperatures approaching 190 degrees 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 degrees F. Furthermore, in the future, these curves will be revised to require temperatures that exceed 212 degrees 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 degrees F at 32 EFPY.

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 TS as a basis for establishing system operability requirements. However, some Limiting Conditions for Operation (LCO) cannot be satisfied during hydrostatic and leak tests at elevated temperatures. Specifically, certain LCOs for HPCI, RCIC, ADS, and the SRVs cannot be satisfied during these tests for reasons discussed below:

1. TS 3.5.C: Requires that the 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 degrees F. HPCI cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

2. TS 3.5.D: Requires that the 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.

3. TS 3.5.E: Requires that the RCIC 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 degrees F. RCIC cannot be made operable during the test because piping normally filled with steam is filled with water during the test.

4. TS 3.6.E: Requires the SRVs to be operable when the reactor coolant system exceeds atmospheric pressure and temperature is greater than 212 degrees F. The SRVs will have to be gagged closed when test pressures exceed the SRV setpoints thus rendering them inoperable.

As stated above, the required hydrostatic pressure and inservice leak testing cannot be conducted without making HPCI, RCIC, ADS, and SRVs inoperable. The proposed changes to the TS will allow testing to be conducted at elevated temperatures with these systems inoperable.

3.0 EVALUATION

As outlined in Chapter 6 of the Updated Final Safety Analysis Report (UFSAR), "Emergency Core Cooling System (ECCS)," in the event of a Loss of Coolant Accident (LOCA), the ECCS is designed to remove residual heat including stored heat and heat generated due to radioactive decay, such that excessive fuel clad temperature is prevented. The objective of the ECCS is to limit, in conjunction with primary and secondary containments, the release of radioactive materials to the environs following a LOCA so that resulting radiation exposures are kept within the guideline values given in 10 CFR Part 100. In order to satisfy the Safety Design Bases, four systems are provided for emergency core cooling:

1. HPCI System
2. Automatic Depressurization System (ADS)
3. Core Spray System
4. LPCI, an operating mode of the RHR System

These are, in addition to the other systems which supply core coolant, feedwater, control rod drive (CRD) hydraulic pumps, and RCIC.

The manner in which the ECCS operate to protect the core is a function of the rate at which coolant is lost from the break in the Reactor Coolant Pressure Boundary. The HPCI System is designed to operate while the Reactor Coolant System is at high pressure. The Core Spray System and LPCI are designed for operation at low pressures. If the break in the Reactor Coolant Pressure Boundary is of such a size that the loss-of-coolant exceeds the capacity of the HPCI System, Reactor Coolant System pressure drops at a rate fast enough to allow the Core Spray System and LPCI to pump additional coolant into the reactor vessel in time to cool the core.

Automatic depressurization is provided to automatically reduce Reactor Coolant System pressure if a break has occurred and vessel water level is not maintained by the HPCI System and other water addition systems. Rapid depressurization of the Reactor Coolant System is desirable to permit flow from the Core Spray and LPCI Systems to enter the vessel, so that the temperature rise in the core is limited.

During hydrostatic testing and system leakage testing of the RCS, the Recirculation pumps are in operation and a water-solid condition is maintained to control the necessary pressure and temperature. Reactor water makeup, pressure, and level is controlled using the Control Rod Drive and Reactor Water Cleanup systems. During the tests, all control rods are inserted to ensure the core remains subcritical and adequate subcriticality margins are maintained. Furthermore, the decay heat level is minimized following the refueling or maintenance activities, and the reactor is maintained at or near cold shutdown conditions.

During the hydrostatic pressure and leak test conditions, the postulated worst case accident is a LOCA. The effects of a small or large break LOCA are bounded by the existing plant analyses. This is assured by the following test conditions: the control rods are maintained fully inserted to maintain subcriticality margins, the reactor coolant inventory is large, the reactor coolant energy (enthalpy) is significantly less than that during power operation, and the decay heat is low. With a small break LOCA, the RCS will depressurize while the operator terminates the test and initiates RHR cooling and/or low pressure ECCS, as necessary. With a large break LOCA, the reactor will rapidly depressurize and all low pressure ECCS with their initiating instrumentation will be available. The operability of these low pressure ECCS Systems is assured by the requirements of TS 3.5, "Core and Containment Cooling Systems."

Primary containment integrity will be maintained during the hydrostatic testing and system leakage testing of the RCS. Furthermore, other systems designed to restrict radiological release (i.e., Secondary Containment and the Standby Gas Treatment System) will also be available. The availability of these systems will assure that offsite releases remain within the guideline values of 10 CFR Part 100.

For the above reasons, the NRC staff finds that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 4494). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
Brian McCabe

Date: March 9, 1992

Docket No. 50-333

March 9, 1992

Mr. Ralph E. Beedle
Executive Vice President - Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Beedle:

SUBJECT: ISSUANCE OF AMENDMENT FOR JAMES A. FITZPATRICK NUCLEAR POWER PLANT (TAC NO. M82630)

The Commission has issued the enclosed Amendment No. 179 to Facility Operating License No. DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated January 9, 1992.

The amendment revises 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 at RCS temperatures exceeding 212 degrees F. During this testing, the High Pressure Coolant Injection, Reactor Core Isolation Cooling, and the Automatic Depressurization System/Safety Relief Valves are not required to be operable.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original Signed By
Brian C. McCabe, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 179to DPR-59
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDI-1:LA	PDI-1PM	SRXB	OGC	PDI-1:D
NAME	CSVogan	BCMcCabe:pc	RJones		RACapra
DATE	02/18/92	02/18/92	02/19/92	02/19/92	03/09/92

OFFICIAL RECORD COPY
FILENAME: C:\WP\WPDOCS\M82630.AMD

DATED: March 9, 1992

AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-59-FITZPATRICK

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
B. McCabe
C. Cowgill
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9-A-1
R. Jones
G. Hill (4), P-137
Wanda Jones, P-130A
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFMB
PD Plant-specific file

cc: Plant Service list