

John C. Brons
Executive Vice President
Nuclear Generation

April 30, 1990
JPN-90-036

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Mail Stop P1-137
Washington, D. C. 20555

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Generic Letter 89-19
Safety Implications of
Control Systems in LWR Nuclear Power Plants

- References:
1. NRC Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants, Pursuant to 10 CFR 50.54(f),' dated September 20, 1989.
 2. NYPA letter, J. C. Brons to USNRC, dated March 19, 1990 (JPN-90-014/IPN-90-024) regarding Generic Letter 89-19.
 3. BWROG letter, S. D. Floyd to J. G. Partlow, (NRC) dated April 2, 1990 (BWROG-9048) transmits BWROG response to GL 89-19.

Dear Sir:

In Generic Letter 89-19 (Reference 1), the NRC recommended that the Authority reassess and, if necessary, modify the existing reactor overfill protection system at the James A. FitzPatrick Nuclear Power Plant. In Reference 2, the Authority rescheduled its response to GL 89-19 to permit our review of a generic response prepared by the BWROG (Boiling Water Reactor Owners' Group). The BWROG report was submitted to the NRC as an attachment to Reference 3.

The Authority has reviewed the BWROG report and has determined that it is applicable to the FitzPatrick plant. The Authority endorses Reference 3.

The BWROG report summarized the design of installed overfill protection at several BWRs, estimated the cost of hardware modifications recommended by GL 89-19, and reviewed relevant operational experience for BWRs throughout the world. The BWROG report concludes that the recommended modifications are expensive and do not contribute significantly to overall plant

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safety. Operational experience does not suggest that improvements to the existing system are warranted. No feedwater control system common mode failures were identified by the BWROG that might have resulted in reactor vessel overflow.

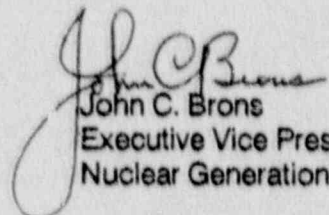
Based on this report, the Authority believes that the FitzPatrick's existing overflow protection is adequate. No hardware modifications are planned in connection with this issue.

The Authority recognizes the importance of a reliable overflow protection system. To confirm the conclusions of the BWROG report, the Authority will expand the FitzPatrick IPE/PRA to quantify the risks associated with vessel overflow and feedwater control system failures. If the results of the IPE/PRA indicate that the existing overflow protection system is a significant contributor to overall risk, modifications will be considered.

Plant shutdown and startup procedures will be changed to assure system operability as recommended by GL 89-19. Surveillance procedures already adequately assure overflow protection equipment operability. Simulator training for licensed operators includes feedwater control problem scenarios. FitzPatrick's Technical Specifications require periodic verification of overflow protection system operability. No additional overflow protection LCOs or STs are being considered.

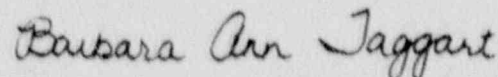
The Authority's complete response and further details are included in Attachment I. Should you or your staff have any questions regarding the Authority's plans or position, please contact Mr. J. B. Ellmers of my staff.

Very truly yours,


John C. Brons
Executive Vice President
Nuclear Generation

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and sworn to before me
this 30th day of April 1990.



Notary Public

BARBARA ANN TAGGART
NOTARY PUBLIC, State of New York
No. 4281437
Qualified in Putnam County 92
Commission Expires Jan. 27, 1991

cc: Next page

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**NEW YORK POWER AUTHORITY
James A. FitzPatrick Nuclear Power Plant**

1.0 Background and Introduction

In Generic Letter 89-19 (Reference 1), the NRC recommended that the Authority reassess and modify existing reactor overfill protection at the James A. FitzPatrick Nuclear Power Plant. These recommendations were a direct result of the staff's recent resolution of Unresolved Safety Issue (USI) A-47, "Safety Implications of Control Systems in LWR Nuclear Power Plants."

Generic Letter 89-19 (GL 89-19) also requires that the Authority submit, under oath or affirmation, a statement to the NRC as to whether the recommendations of Enclosure 2 to GL 89-19 will be implemented, a schedule for the implementation of any modifications or changes, and the basis for that schedule.

This attachment constitutes the Authority's response.

1.1 BWROG on Hardware Change Recommendations

The Authority participated in the BWROG committee on GL 89-19. The committee prepared a report which has been submitted to the NRC staff (Reference 3). The Authority has reviewed this report and has determined that it is applicable to the FitzPatrick plant. The Authority endorses this report.

The BWROG report summarized the design of installed overfill protection at several BWRs, estimated the cost of hardware modifications recommended by GL 89-19, and reviewed relevant operational experience for BWRs throughout the world. The BWROG report concludes that the recommended modifications are expensive and do not contribute significantly to overall plant safety. Operational experience does not suggest that improvements to the existing system are warranted. No feedwater control system common mode failures were identified by the BWROG that might have resulted in reactor vessel overfill.

Based on this report, the Authority believes that the existing overfill protection system is adequate. No hardware modifications will be installed at this time.

1.2 Incorporation into FitzPatrick IPE/PRA Program

The Authority recognizes the importance of a reliable overfill protection system. To confirm the conclusions of the BWROG report, the Authority will expand the FitzPatrick IPE/PRA to quantify the risks associated with vessel overfill and feedwater control system failures. If the results of the IPE/PRA indicate that the existing overfill protection system is a significant contributor to overall risk, modifications will be considered.

Attachment I
Response to NRC Generic Letter 89-19

Although the Authority did not identify that it would resolve USI A-47 as part of the FitzPatrick IPE/PRA in Reference 6, we estimate that this additional work can be completed without delaying the IPE/PRA.

1.3 Plans and Schedules

If, as result of the IPE/PRA, the reactor overfill protection system is determined to be a significant contributor to overall risk, the Authority will submit plans and schedules for appropriate modifications one hundred and twenty days after the IPE/PRA results have been sent to the NRC. This report will include a description of the proposed modification(s), along with plans and schedules for:

- plant modification(s),
- operator training,
- surveillance procedures
- test procedures,
- plant operating procedures, and
- technical specification changes.

As described in the Authority's 60-day IPE/PRA response (Reference 6), the Authority will submit the FitzPatrick IPE/PRA by the end of this year (December 30, 1990). Based on this date, plans and schedules for any necessary overfill protection system modifications will be submitted by April 30, 1991.

The Authority's IPE/PRA program was recently approved by the NRC in Reference 10.

2.0 Line-by-Line Response to GL 89-19

The paragraphs below respond line-by-line to the five specific issues detailed in GL 89-19 for the FitzPatrick plant.

2.1 Section 1(a) - Evaluation of Overfill Protection

Group Classification (I, II, III): Class I. Automatic overfill protection trips the main feedwater pumps in a "2-out-of-3" logic system.

Significant Deviations from Recommendations: The existing overfill protection system is not electrically independent or physically separate from the main feedwater control system. One of the three reactor vessel level transmitters is devoted exclusively to overfill protection; the two remaining transmitters also provide signals to the main feedwater control system.

Implementation Schedule: No hardware modifications to the existing overfill protection system are warranted at this time.

Review of Plant Operating Procedures

Existing Procedures: Plant operating procedures for the High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC) and the Condensate/Feedwater Systems were reviewed to assure that they adequately address reactor vessel overfill protection. Based on their design and flow capabilities, only these three systems could potentially overfill the reactor vessel. (The Control Rod Drive Hydraulic System is a low-flow system and the low-pressure emergency core cooling systems are low-head systems.)

A caution statement will be added to FOP-65, "Start-up and Shutdown Procedure," (Ref. 11). This operating procedure provides detailed instructions for reactor start-up and shutdown.

This new caution will alert operators of the potential of over-filling the vessel with the condensate booster pumps during startup and shutdown while the reactor vessel is at reduced pressure. The remaining plant procedures adequately address existing reactor vessel overfill protection.

Implementation Schedule: This caution statement will be added to FOP-65 not later than June 30, 1990.

Operator Training

Existing Training Program: Existing operator training programs adequately address reactor vessel overfill protection. Simulator training for licensed operators already includes feedwater control system problem scenarios. Emergency Operating Procedure training scenarios include situations when the reactor is at reduced pressure and the condensate booster pumps are operating. During EOP training, operators practice controlling reactor level and compensating for increasing level.

Implementation Schedule: No changes to the operator training program are required.

2.2 Section 1(b) - Surveillance and Test Procedures

Existing Procedures: Plant surveillance and test procedures already include provisions to periodically demonstrate overfill protection operability. Procedures are also already in place to assure that the existing overfill protection system is operable during power operation.

Implementation Schedule: No changes to the existing plant surveillance and test procedures are required.

Technical Specifications

Existing Specifications: The LCOs and surveillance requirements identified below periodically verify the operability of the existing overfill protection and assure that automatic overfill protection is available to mitigate main feedwater overfill events during reactor power operation.

Attachment I
Response to NRC Generic Letter 89-19

The following portions of the FitzPatrick Technical Specifications address existing automatic overfill protection:

Limiting Conditions for Operation

Limiting Conditions for Operation (LCO) 3.2.B, "Core and Containment Cooling Systems - Initiation and Control," page 49.

Table 3.2-2, "Instrumentation That Initiates or Controls the Core and Containment Cooling Systems", Item 3, "High Reactor Water Level," page 66.

Surveillance Tests

Surveillance Requirement 4.2.B, "Core and Containment Cooling Systems - Initiation and Control," page 49.

Table 4.2-2, "Minimum Test and Calibration Frequency for Core and Containment Cooling Systems," Item 1, "Reactor Water Level," page 79.

Implementation Schedule: No changes to the existing specifications are necessary at this time.

Attachment I
Response to NRC Generic Letter 89-19

3.0 References

1. NRC Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants,' Pursuant to 10 CFR 50.54(f)," dated September 20, 1989.
2. NYPA letter, J. C. Brons to USNRC, dated March 19, 1990 (JPN-90-014/INP-90-024) regarding Generic Letter 89-19.
3. BWROG letter, S. D. Floyd to J. G. Partlow, dated April 2, 1990 (BWROG-9048) transmits BWROG report EDE 07-0390/DRF A00-03773, "BWROG Response to GL89-19, Enclosure 2, Hardware Change Recommendations," Rev. 0.
4. NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," dated November 23, 1988.
5. Supplement 1 to GL 88-20, dated August 29, 1989 regarding initiation of the individual plant examination for severe accident vulnerabilities - 10 CFR 50.54(f).
6. NYPA letter, J. C. Brons to USNRC, (JPN-89-069/IPN-89-066) dated October 27, 1989. Submits 60-day response to GL 88-20 for both FitzPatrick and Indian Point 3.
7. NUREG-1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants, Technical Findings Related to USI A-47, Final Report," published June 1989.
8. NUREG-1218, "Regulatory Analysis for Proposed Resolution of USI A-47, Safety Implications of Control Systems, Draft Report for Comment," published April 1988.
9. NUREG/CR-4262, "Effects of Control System Failures on Transients and Accidents at a General Electric Boiling Water Reactor," including both Vols. 1 and 2, main report and appendices, respectively.
10. NRC letter, D. E. LaBarge to J. C. Brons, dated January 30, 1990 regarding review of 60-day response to GL 88-20. Concludes that the FitzPatrick IPE/PRA approach, methodology and schedule are acceptable.
11. FitzPatrick Operating Procedure FOP-65, "Start-up and Shutdown Procedure."