Indian Point 3
Nuclear Power Plant
P.O. Box 215
Buchanan, New York 10511
914 736.8001

NewYork Power Authority

Robert J. Barrett
Site Executive Officer

August 11, 1997
IPN-97-107

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
License No. DPR-64
Reply to Notice of Violation 50-286/97-03

Dear Sir:

This letter provides, in Attachment I, the New York Power Authority's response to the Notice of Violation contained in NRC Integrated Inspection Report 50-286/97-03. The Authority agrees with violations VIO-97-03-01, VIO-97-03-02 and VIO-97-03-03. The Authority does not agree with violation VIO-97-03-04.

The commitments made by the Authority with this letter are contained in Attachment II. If you have any questions, please contact Mr. K. Peters at (914) 736-8029.

Very truly yours,

Robert J. Barrett

Robert J. Barrett
Site Executive Officer
Indian Point 3 Nuclear Power Plant

cc: See next page
Attachments

IE01/1

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406-1415

U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

Reply to Notice of Violation 50-286/97-03

During an NRC inspection completed on May 17, 1997, violations of NRC requirements were identified. These violations and the Authority's response are as follows:

A. Violation (50-286/97-03-01)

"Indian Point 3 Technical Specification (TS) 4.5.A.6.c for the fuel storage handling building ventilation states that "prior to handling of irradiated fuel or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal absorber use since the last test, the following conditions shall be demonstrated before the system can be considered operable:

(1) Charcoal shall have a methyl iodine removal efficiency ≥90%...

(2) A halogenated hydrocarbon (freon) tests on charcoal...

(3) A locally generated DOP test of the HEPA filters...

(4) Visual inspection in accordance with the applicable sections of ANSI N 510...

Contrary to the above, the testing in sections (3) and (4) of TS 4.5.A.6.c was not demonstrated prior to declaring the system operable on April 19, 1997.

This is a Severity Level IV violation."

Response to Violation 97-03-01

The New York Power Authority agrees with this violation. The Fuel Storage Building (FSB) ventilation system was declared operable before all testing was performed as required by Technical Specification (TS) 4.5.A.6.c.

Reason for Violation

The cause of this violation was personnel error due to document use practices. The Shift Manager (SM) failed to correctly understand and therefore, to follow the phraseology in TS 4.5.A.6.c. The Shift Manager (SM), after completion of tests on the High Efficiency Charcoal Adsorbers (HECA) only, declared the Fuel Storage Building (FSB) ventilation system operable and exited the LCO.

During repairs to the roof of the FSB on April 16, 1997, fumes from a roof repair sealant were drawn into the FSB by the 31 FSB supply fan and detected by workers during fuel handling activities. At this time the FSB ventilation system was aligned to provide flow through the system's High Efficiency Particulate Adsorbers (HEPA) and HECA because of movement of the cask crane over the spent fuel pit containing irradiated fuel. With ventilation flow through the HECAs, Performance and Chemistry judged the chemical fumes had the potential to affect the HECA portion of the FSB ventilation system and notified Operations.

Reply to Notice of Violation 50-286/97-03

Operations concluded the chemical fumes could have altered the integrity of the HECAs and declared the FSB ventilation system inoperable and entered a Limiting Condition for Operation (LCO 97-0582) for TS 3.8 at approximately 1100 hours. TS 3.8.C.6 requires that the FSB ventilation system be operable whenever irradiated fuel is being handled within the FSB. The FSB ventilation system may be inoperable when irradiated fuel is not being handled and neither the spent fuel cask nor the cask crane are being moved over the spent fuel pit during the periods of inoperability. A Work Request (WR) for testing only the FSB charcoal filter (HECA) was authorized by a Control Room Supervisor (CRS) on April 18, 1997. TS 4.5.A.6.c requires "prior to handling of irradiated fuel, or at any time fire, chemical releases or work done on the filters could alter their integrity or after every 720 hours of charcoal adsorber use since the last test", that four conditions be demonstrated before the FSB ventilation system can be considered operable; a methyl iodine test and freon test of the charcoal filter (HECA), a DOP test of the HEPA filters, and a visual inspection.

Testing of only the HECA was performed using the applicable sections of test procedure 3PT-R32A, "Fuel Storage Building Filtration System," which is used to perform all four tests. Upon satisfactory completion of only the HECA test, the Shift Manager (SM) declared the FSB ventilation system operable and exited the TS LCO at approximately 1138 hours on April 19, 1997. On April 25, 1997 an NRC resident informed IP3 management that all four tests had not been performed. The Operations Department declared the FSB ventilation system inoperable. All four tests were subsequently performed satisfactorily and the FSB ventilation system was declared operable on April 27, 1997.

Corrective Actions Taken

- The FSB ventilation system was tested satisfactorily in accordance with TS 4.5.A.6.c and declared operable on April 27, 1997.

- Test procedure 3PT-32A was revised to note that when declaring the FSB ventilation system inoperable for the conditions listed in TS 4.5.A.6.c, then each of the four tests identified in TS 4.5.A.6.c shall be performed.

- Operations issued a Shift Order notifying the operations staff of the issue and the requirement to perform all the tests of TS 4.5.A.6.c when conditions (fire, chemical release or work on the filters could alter their integrity) invoke TS 4.5.A.6.c.

- The Operations Manager instructed licensed operators on the use of TS section 4 before declaring a system operable.

- Test procedures for other Air Filtration Systems subject to similar TS were revised to note that when declaring a ventilation system inoperable for the conditions listed in a TS, then each of the tests identified in the TS shall be performed.

Reply to Notice of Violation 50-286/97-03

Corrective Actions to be Taken to Avoid Further Violations

The Authority believes the actions taken should prevent further recurrence.

Date When Full Compliance Will Be Achieved

Compliance was achieved on April 27, 1997, when the FSB ventilation was satisfactorily tested per TS 4.5.A.6.c.

Reply to Notice of Violation 50-286/97-03

B. Violation (50-286/97-03-02)

"10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires in part that measures shall be established to assure that conditions adverse to quality such as deviations and non-conformances are promptly identified and corrected. Administrative Procedure (AP)-8...requires that an operability determination shall be initiated for degraded and non-conforming conditions for which the operability status is indeterminate.

Contrary to the above, from February 7, 1997 through February 10, 1997, the licensee identified that the plant was being operated with a design feature defeated in that the residual heat removal (RHR) suction valves were deenergized and open, but did not promptly correct the non-conforming condition.

This is a Severity Level IV violation."

Response to Violation 97-03-02

The Authority agrees with this violation. Appropriate corrective action was not taken when the RHR suction valves were found in a condition that was in disagreement with the Final Safety Analysis Report (FSAR).

Reason for Violation

The cause of this violation was personnel error due to work practices. The work practices did not promptly correct the non-conforming condition because of habit intrusion, wrong assumptions, and a lack of perceived need for prompt action.

On February 7, 1997, with the plant depressurized and RCS temperature less than 200 degrees F, Operations Department staff identified that a step in plant operating procedure POP-4.1, Operation at Cold Shutdown, directed RHR suction valves AC-MOV-730 and 731 to be de-energized open when the RCS was depressurized and vented. This was not consistent with the FSAR which states that AC-MOV-730 and 731 are automatically closed when the RCS is above a certain pressure. At the time valves AC-MOV-730 and 731 were procedurally opened and deenergized. This discrepancy was identified in Deviation Event Report (DER) 97-0292. The shift manager determined that this condition was acceptable and a written evaluation was not necessary. A 10 CFR 50.59 evaluation for this configuration could not be located. This condition defeated a feature designed to isolate the RHR system from the RCS contrary to the FSAR. This placed the RHR system in a non-conforming condition which required further evaluation and documentation in accordance with administrative procedure AP-8, Deviation & Event Reporting and Operability Determination Procedure. AP-8 requires that an operability determination be performed for non-conforming conditions for systems whose operability status is indeterminate. AP-8 also requires that an engineering evaluation based on engineering judgment should be documented within 24 hours and a final evaluation within 30 days.

Reply to Notice of Violation 50-286/97-03

Operations Department personnel documented, in DER 97-0292, that further evaluation was necessary to determine if this condition was acceptable, but they did not take the appropriate steps to address the issue promptly. The operators believed they had taken the appropriate action because the plant had been safely operated in this condition previously, the work in progress had been appropriately risk assessed and the plant was in a safe condition, and it was also believed that the condition would be exited promptly and well before the issue would be fully analyzed. During review of DER 97-0292, the DER Review Committee (DRC) recognized that the discrepancy was a significant issue that required action but failed to take action to initiate more timely corrective action.

The RHR system was subsequently placed in a condition consistent with the FSAR on February 10, 1997 when the RHR suction valves were energized as part of preparations for plant heat up. Based on the conclusion of a nuclear safety evaluation (NSE), completed on May 15, 1997, operation with the RHR suction valves deenergized open was evaluated to have been acceptable for the plant conditions that existed on February 7, 1997.

Corrective Actions Taken

- The RHR suction valves were energized as a part of preparation for plant heat up.
- An NSE was prepared by engineering which concluded that operation with the RHR suction valves deenergized open in cold shut down was acceptable.

Corrective Actions to be Taken to Avoid Further Violations

- The failure to take proper and timely corrective actions when a non-conforming condition was identified for the RHR suction valves will be addressed during regularly scheduled Operations Manager 's time as part of the continuing operator training program. To be completed by November 19, 1997.
- Procedural guidance will be enhanced to ensure that corrective actions are promptly initiated when a plant condition is identified that does not conform with the Technical Specifications or the FSAR. To be completed by November 14, 1997.

Date When Full Compliance Will Be Achieved

Compliance was achieved on February 10, 1997 when the RHR suction valves were energized.

Reply to Notice of Violation 50-286/97-03

C. Violation (50-286/97-03-03)

"Indian Point 3 Technical Specification 6.8.1 requires written procedures be implemented covering activities referenced in Appendix A of Regulatory Guide 1.33, 'Quality Assurance Program Requirements', November 1972...

1. Indian Point 3 procedure CON-AD-006, "Scaffolding Control," revision, 6, requires that scaffolds located in safety-related areas which cannot be installed to conform with the technical requirements of this procedure shall be reviewed and approved by design engineering prior to use.

   Contrary to the above, on April 29, 1997, a scaffold located in the service water pump area was not installed to conform with the technical requirements of the procedure, in that the scaffold was within one inch of the service water pumps, but was not reviewed by design engineering prior to use...

2. Maintenance procedure SYS-011-GEN, revision 2, Foreign Material and Chemical Exclusion Requirements for Controlling System Cleanliness During Maintenance Activities, step 4.2.5, requires that all system/component openings be covered anytime that work is not in progress and the work area is left unattended.

   Contrary to the above, on April 9, 1997, openings on the 31 instrument air compressor head and crankcase were not covered when the work was not in progress and the area was unattended.

3. Reactor engineering procedure RA-27.2, revision 1, Internal Transfer of Fuel Assemblies and Inserts, section 4.12 states that a Transfer Form Checklist shall be completed for revisions to a fuel handling transfer form.

   Contrary to the above, on April 17, 1997, a revision was made to transfer form (TF) 1997-001 without completing a transfer form checklist.

This is a Severity Level IV violation."

Response to Violation 97-03-03

The Authority agrees with this violation. Contrary to Technical Specification 6.8.1 the following procedural requirements were not met: scaffolding was installed that did not meet the requirements of procedure CON-AD-006, "Scaffolding Control"; 31 Instrument Air Compressor (IAC) head was left uncovered and unattended while work was not in progress contrary to SYS-011-GEN, "Foreign Material and Chemical Exclusion Requirements for Controlling System Cleanliness During Maintenance Activities" and COM-001-IAS,"Instrument Air Compressor Annual Inspection", and a revision was made to transfer form (TF) 1997-001 without completing a transfer form checklist or a Fuel Assembly Handling Deviation Report contrary to RA-27.2, "Internal Transfer of Fuel Assemblies and Inserts."

Reply to Notice of Violation 50-286/97-03

Reason for Violation

1. The cause of not complying with the requirements of CON-AD-006 was Personnel Error due to work practices. The work supervisor and craft workers did not correctly follow the requirements of CON-AD-006.

   During construction of scaffolding in the Service Water Pump Pit a scaffolding supervisor did not identify that this area was safety related, even though it was listed as a safety related area in the procedure. As a result the required items were not checked on the Scaffold Installation Checklist therefore a required review by Operations Department Personnel was not performed as required by CON-AD-006. In a separate event, scaffold construction was started in the containment spray pump area. Scaffold construction completion was contingent upon receiving metal planks as is preferred in the Radiologically Controlled Area (RCA). Three days later, scaffold construction had not yet been completed. As a result, reviews by engineering and operations had not been performed, contrary to the requirements of CON-AD-006 that erection, support and removal of scaffolding in safety related areas of the plant be performed in a timely manner in order to minimize the time that partially installed scaffolding is near safety related equipment.

2. The cause of not complying with the requirements of SYS-011-GEN and COM-001-IAS was Personnel Error due to work practices. The work supervisor and craft workers did not follow the requirements of SYS-011-GEN and COM-001-IAS.

   On April 9, 1997 preventive maintenance was being performed on 31 Instrument Air Compressor (IAC). It was noted by an NRC resident that, during a break period, not all component openings were closed as required by SYS-011-GEN and COM-001-IAS. Upon notification, Maintenance Department workers took action to restore proper FME controls in accordance with the procedural guidance, which required personnel to cover openings when the work area is left unattended.

3. The cause of failure to follow RA-27.2 was personnel error due to lack of self checking. If the reactor engineer ensured that the proper revision of RA-27.2 was being used, a Fuel Assembly Handling Deviation Report (FAHDR) entry would have been made when making changes to TF 1997-001. If the reactor engineer who made additional changes to TF 1997-001 ensured that all appropriate steps of RA-27.2 had been completed, the Transfer Form Checklist for TF 1997-001 would have been completed.

   An NRC resident identified discrepancies with a revision to a fuel transfer form (TF 1997-001). Revision 1 to RA-27.2, the procedure that governs changes to fuel transfer forms, was to take effect on April 14 but had not been distributed. Although not a requirement, IP3 administrative procedures allow for expedited distribution of procedures. Expedited procedure distribution was not used for Revision 1 of RA-27.2. On April 14, 1997 it became necessary to place a fuel assembly in a different location in the Spent Fuel Pool (SFP) than originally planned. Changes were made to TF 1997-001, as required when placing an assembly in a different location. However, Revision 0 to RA-27.2 did not include the requirement to generate a FAHDR when revising transfer forms, and this requirement of the effective revision (revision 1) was not met.

Reply to Notice of Violation 50-286/97-03

Fuel transfer form (TF) 1997-001 was again changed on April 17,1997 to delete six steps from the fuel transfer sequence, as allowed by RA-27.2. The reactor engineer did not complete the Transfer Form Checklist as required by RA-27.2. The engineer incorrectly believed that a Transfer Form Checklist was only required when adding information to a Transfer Form and was not required when only a deletion was being made, and that the Transfer Form Checklist completed for the Transfer Form to which the steps were being added was adequate for both actions.

Corrective Actions Taken

1.
- Scaffolding activities were stopped and personnel involved with scaffolding construction were briefed on the incident and reviewed CON-AD-006. In addition, the expectations of self-checking and procedure adherence were reviewed with scaffolding personnel.
- Disciplinary actions involving suspensions of direct supervision were implemented.
- A walkdown of other scaffolding in the plant was performed to ensure compliance with scaffolding installation requirements in CON-AD-006. One additional deficiency was noted and corrected.
- Maintenance Department has taken ownership of the scaffolding program and revised the scaffolding procedure to clarify requirements for timely inspection of partially completed scaffolding. This procedure will be in effect on August 25, 1997. In the interim, administrative controls have been established by Maintenance Department management delineating additional requirements for scaffolding.

2.
- FME controls were restored for 31 IAC.
- Maintenance Department personnel were briefed on the FME event during department meetings.
- The Maintenance Department established an FME self assessment program for the current refueling outage.

3.
- The reactor engineers responsible for not complying with RA-27.2 were counseled by the Site Reactor Engineer on procedure adherence and the use of self-checking.
- The errors were corrected and reviewed for completeness in accordance with the requirements of RA-27.2, revision 1.
- A review of previous and subsequent fuel and core component sequences were reviewed to ensure procedural compliance. No other discrepancies were noted.

Reply to Notice of Violation 50-286/97-03

Corrective Actions to be Taken to Avoid Further Violations

Actions taken should prevent further occurrence.

Date When Full Compliance Will Be Achieved

Compliance was achieved on May 9, 1997, in that by this date the scaffolding was installed in accordance with the requirements of CON-AD-006, the FME requirements for the IAC were restored per SYS-011-GEN and COM-001-IAS, and Transfer Form 1997-001 was corrected.

Reply to Notice of Violation 50-286/97-03

D. Violation (50-286/97-03-04)

"10 CFR 50.55a(f) requires that inservice testing (IST) of certain ASME Code Class 1,2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (the Code) and applicable addenda, except where specific written relief has been requested by the licensee and granted by the Commission. Section XI of the Code (1983 Edition), IWV-1100 Scope, requires IST for valves that are required to perform a specific function in shutting down a reactor to the cold shutdown condition or in mitigating the consequences of an accident.

Contrary to the above on May 16, 1997, the NRC identified that the inservice testing of the residual heat removal suction valves, AC-MOV-730 and 731, which are Class I valves that perform a specific function in mitigating the consequences of an accident, were not performed in accordance with the Code. Specifically, these valves have a pressure isolation function to isolate the residual heat removal system from the reactor coolant system, but the closing function of these residual heat removal suction valves was not included within the scope of the IST program.

This is a Severity Level IV violation."

Response to Violation 97-03-04

The Authority does not agree with this violation. The basis for not agreeing with this violation is discussed below.

Basis

Testing is performed on valves AC-MOV-730 and 731, for certain functions, to meet the Code IST and Technical Specification (TS) requirements. The IST testing includes the open stroke timing of valves AC-MOV-730, 731 to perform a specific function in shutting down the reactor to the cold shutdown condition. In addition, LT-2 leakage testing (as defined in the IP3 IST program) in the closed position is performed as a part of the IST program to address inter-system LOCA concerns as detailed in the Authority's response to Generic Letter (GL) 87-06. The interlock function and automatic isolation function of valves AC-MOV-730, 731 are tested as required by IP3 Technical Specifications. Valves AC-MOV-730 and 731 are also included in the GL 89-10 required motor operated valve (MOV) program and are further tested in accordance with this program to assure the valves will perform their intended function. The close stroke time of these valves is recorded and trended.

IST criteria for the operability of valves AC-MOV-730 and 731 as it related to time to close was not assigned by the Authority because of the Authority's understanding of the requirements of the FSAR, the Code and NUREG 1482. The automatic isolation function was not a specific function in mitigating the consequences of an accident nor was this function necessary in order to shutdown the reactor to the cold shutdown condition.

Reply to Notice of Violation 50-286/97-03

The Authority's conclusion is based on the following:

- Valves AC-MOV-730 and 731 are required to be opened in order to shutdown the reactor to a cold shutdown condition, and as such the open stroke time is tested in accordance with the Code. The close function of these valves is not required to shutdown the reactor to a cold shutdown condition.

- As required by plant procedures, AC-MOV-730 and 731 are closed during plant operation prior to exceeding 400 psig or 350 degrees F. Testing in the closed direction is not required by the Code when operating under these conditions because these valves are already in the position necessary to mitigate an accident (i.e. inter-system LOCA.) Protection from inter-system LOCA is assured by LT-2 leak testing of AC-MOV-730 and 731 in accordance with the Code.

- As described in the FSAR, protection of the RHR system from overpressurization is provided by administrative controls and the RHR relief valve. Section 6.2.3 of the FSAR states, "The design features applied to the Residual Heat Removal System (RHRS) system Valves 730 and 731, that isolate it from the Reactor Coolant System provide a diverse combination of control interlock and mechanical limitations preventing improper opening of these valves and also pressure relief capacity capable of limiting pressure if the valves are not closed upon startup of the plant. These features are: . . .". The FSAR does describe the interlock that prevents opening the valves and generally describes part of the interlock design by saying "and also automatically closes the valve whenever the Reactor Coolant System pressure is above the setpoint." The automatic closure design feature is not credited for limiting pressure if the valves are not closed on startup. For overpressure, the FSAR states that , "4) The RHRS is equipped with a pressure relief valve ... This is a diverse backup to <u>administrative closure</u> of the isolation valves prior to startup to prevent overpressurization when returning the plant to operation."

- Section 2.2 of NUREG-1482 states, " 'Accident' refers to the accident analyses in the safety analysis reports and a broad range of possible adverse events that could occur at a nuclear power plant..." As further discussed in Question Group 104 of Appendix A to NUREG-1482 "...although most of the accidents of concern to IST are addressed in the accident analysis chapter, licensees should be aware that there may be other accident analyses in the FSAR that need to be considered." The Authority concluded that the NUREG referred to accidents or events that are sufficiently analyzed to establish the technical basis for mitigating the consequences of an accident.

- FSAR discussions include the interlocks of valves AC-MOV-730 and 731 that cause automatic closure. However, the FSAR credits other functions for providing protection to the RHR system from adverse conditions (e.g. overpressurization.) In addition, these discussions do not provide an analysis of initiators, mechanisms and sequence of events for the RHR system as it would relate to the automatic isolation function of valves AC-MOV-730 and 731. The IP3 FSAR does not provide an analysis or discussion that would establish the accident mitigating requirement of the close function of valves AC-MOV-730 and 731.

Reply to Notice of Violation 50-286/97-03

In response to a question from the NRC Inspector who raised this issue, an operability determination (OD) was performed on May 16, 1997. This OD concluded that the close stroke timing of valves AC-MOV-730 and 731 would meet the requirements of the Code. Additional review of valve test data back to 1986 concluded that the close stroke time of valves AC-MOV-730 and 731 would have met the Code requirements.

Conclusion

The Authority does not agree with the cited violation for the following reasons:

- The close stroke function of valves AC-MOV-730 and 731 was not necessary to shutdown the reactor to a cold shutdown condition.
- The close stroke function of valves AC-MOV-730 and 731 does not mitigate the consequences of an accident.
- The close stroke function was not analyzed to establish an accident mitigating function. As described in the FSAR, this specific function is not credited to prevent overpressurization when returning the plant to operation if the administrative closure of the valve did not occur on startup, nor is it credited for protection from inter-system LOCA.

In summary, the Authority does not believe that valves AC-MOV-730 and 731 have a required function to close related to accident mitigation and therefore does not agree that they are required to be stroke timed in the close direction. This position is not intended to convey that these valves do not have an important closure function, which we believe is recognized and assured; but that the inclusion of these valves in the IST Program for stroke timing the valves in the closed direction is not required. The Authority believes the testing that is currently performed on valves AC-MOV-730 and 731 is in accordance with NRC regulations and is adequate to demonstrate their operational readiness in both the open and close directions.

Actions Taken or To Be Taken

Not withstanding the Authority's belief that testing for valves AC-MOV-730 and 731 was in accordance the Code, the following actions have been or will be completed:

- An operability determination was performed to determine if the recorded close stroke time of valves AC-MOV-730 and 731 met the requirements of the Code, had the Code applied. The OD demonstrated that the valves would have met the Code requirements, had the code applied.
- The surveillance test results for valves AC-MOV-730 and 731 will be included in the IP3 IST program. We will establish augmented IST program criteria against the timing of these valves. This will be completed by December 29, 1997.
- The Authority will perform a review of valves in the IP3 IST program to identify those valves having an automatic open or close function to determine if such valves require additional acceptance criteria. To be completed by December 29, 1997.