U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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EXECUTIVE SUMMARY

Indian Point Unit 3
NRC Inspection Report 50-286/98-81

The Engineering Inspection Team judged the engineering activities at IP-3 to be generally good and to have improved somewhat from the weak performance of years prior. The engineering methods and results were generally sound and correct, and engineering backlogs have been reduced. The team's extensive review found problems, but the frequency, nature, and significance of the problems were diminished when compared to prior engineering performance. For example, oversights and errors still existed and had resulted in improper engineering actions associated with control room habitability analyses and a change to specified flow in an emergency operating procedure (EOP) change. Also, the team identified problems regarding surveillance tests, safety evaluations, qualification of engineering reviewers, and administrative control of calculations.

The conduct of engineering was generally sound, including modifications, and modification, surveillance, and post-maintenance testing. One oversight in surveillance testing was identified by the team and was being corrected by NYPA.

- Modifications associated with EQ component replacements were implemented appropriately and in accordance with NYPA's EQ upgrade procedures. Modification acceptance tests reviewed were acceptable. (E1.1)

- The surveillance test program and post maintenance testing were generally acceptable for the sample of surveillance tests and work orders reviewed. The team identified a minor error in several surveillance tests and contacts which were not being tested; however, the safety consequences were minimal and NYPA was correcting the test procedures. (E1.2)

Engineering support of facilities and equipment was generally good, including design calculations, system monitoring, implementation of the Maintenance Rule, and operability determinations. However, inadequate evaluations had been performed for control room habitability and a flow change for an EOP, and resulted in violations. The backlog of engineering work has been declining.

- The quality of engineering work on the control room habitability and ESF leakage issues was weak. While these issues have a complicated regulatory and technical history, NYPA did not demonstrate a thorough, effective approach to resolving these issues, and the team concluded that four violations existed regarding design control, 10 CFR 50.59 review, reporting to NRC, and engineering administrative control. Often, corrective actions were implemented piecemeal and were not consistent with the licensing bases. Nonetheless, in many instances on these issues, NYPA engineering demonstrated good quality in calculations and effective insights into technical concerns. (E2.1)

- The team concluded that NYPA had an inadequate evaluation as part of changing an EOP flow and that the inadequate regulatory review had violated 10 CFR 50.59, in that no safety evaluation was performed for the change. Also, the team concluded that the inadequate design review had violated 10 CFR, Appendix B, Criterion III (Design Control), in that significant aspects of the design (ability to operate, recirculation pump NPSH, and radiological consequences) were not evaluated prior to the change and the flow chosen had no basis (other than historical) or calculation to support it. The team noted that the NYPA effort to originally address the EOP flow concern had been well-intentioned and that the subsequent design evaluations had been thorough and accurate. (E2.2)

- Design calculations had good quality, including technically acceptable assumptions and design inputs, and appropriate reviews and verifications. A violation was identified concerning two instances of non-compliance with administrative design control procedures. (E2.3)

- System monitoring and implementation of the Maintenance Rule for the component cooling water, containment isolation, containment recirculation fans and main steam systems was acceptable. System engineers were knowledgeable of their responsibilities regarding the rule. (E2.4)

- The quality and depth of operability determinations were generally acceptable. Operability conclusions were correct. (E2.5)

- The backlog of engineering work was being managed effectively, and the amount of backlogged work has been declining. (E2.6)

Engineering procedures and documentation were acceptable, including procedures for modifications, safety evaluations, UFSAR changes, and temporary modifications. The team identified concerns regarding review guidance and documentation for margin of safety, and the safety evaluation qualifications at the corporate office.

- The team concluded that the NYPA modification procedures were clearly written and provided appropriate guidance. (E3.1)

- The team concluded that the three modifications selected for review were properly prepared in accordance with appropriate procedures and engineering and demonstrated sound technical analyses. (E3.2)

- The team concluded that NYPA's safety evaluations of modifications and UFSAR changes were acceptable. The team noted that documentation of the margin of safety was sometimes weak, in that it was insufficient to enable an independent determination, and a deficiency in the review guidance regarding margin of safety was identified by the team and addressed by NYPA. (E3.3)

- Currently installed temporary modifications including documentation were acceptable. (E3.4)

- NYPA had a satisfactory program for training employees on the requirements of 10 CFR 50.59 and preparing safety evaluations. The contents of the training were appropriate and supported the 10 CFR 50.59 evaluation process. The 50.59 training and qualification appeared to have been properly applied for the plant engineering staff; however, the 50.59 training qualification process for the engineering staff at NYPA's White Plains Office was found to need improvement. (E3.5)

Quality assurance of engineering activities was appropriate and acceptable. Problem identification and root cause evaluation were effective. Also, review committees, self-assessments, and audits were contributing to the effectiveness of engineering activities. The team did not identify any additional examples of ineffective corrective action, but NRC and NYPA have previously identified weaknesses that are being addressed.

- The overall material condition of plant equipment and problem identification was good. The equipment appeared to be properly maintained and in adequate working condition. Deficiencies were being appropriately identified and planned corrective actions tracked to correct the problems. Problems associated with the effectiveness of corrective actions had been identified by the Safety Review Committee and QA audits, a good indication of facility self-assessment. (E7.1)

- DERs were appropriately categorized. Evaluations were technically sound, including human performance, proposed corrective actions were adequately developed, and actions to resolve the problems were being properly tracked. (E7.2)

- Problems regarding corrective action effectiveness and timeliness have previously been identified by NYPA and NRC, but the team's review of corrective action program items did not find additional problems. (E7.3)

- The PORC and SRC were appropriately carrying out their assigned oversight functions. Both committees displayed a strong questioning attitude. (E7.4)

- The completed NYPA self assessments were critical and self-probing. NYPA showed good self-assessment in self-identifying problems related to the effectiveness of corrective actions. (E7.5)

- NYPA's technical evaluations of industry operating experience information. The information were timely, detailed and effective. (E7.6)

- NYPA was processing and resolving employees safety concerns. (E7.7)

- NYPA's self assessment of selected system engineering functional areas was self critical and comprehensive. The QA audit of Design Control was effective and consistent with team findings. (E7.8)

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Report Details

III. Engineering

E1 Conduct of Engineering

E1.1 Plant Modifications

a. Inspection Scope (IP 37550)

The team reviewed three modification packages which were associated with environmental qualification (EQ) component replacements, and two modification acceptance tests associated with the emergency diesel generators (EDGs) and the service water system.

b. Observations & Findings

The modification packages had appropriately upgraded EQ equipment to 10 CFR 50.49 requirements in accordance with the governing NYPA procedure. The EQ program group had reviewed the above modifications and the EQ documentation reflected the changes. The modification acceptance tests were acceptable and included appropriate tests and acceptance criteria.

c. Conclusions

Modifications associated with EQ component replacements were implemented appropriately and in accordance with NYPA's EQ upgrade procedures. Modification acceptance tests reviewed were acceptable.

E1.2 Surveillance/IST Test Program

a. Inspection Scope

The team reviewed surveillance test procedures associated with the recirculation pump and auxiliary feedwater (AFW) systems to assess technical adequacy and procedure clarity with regard to inservice testing (IST) requirements. A sample of engineered safeguards features (ESF) and other safety related logic signals, associated with the AFW/steam generator and safety injection (SI) systems, were reviewed. Engineering personnel were interviewed regarding their respective roles regarding the test program. In addition, several post maintenance tests were reviewed for technical adequacy.

b. Observations & Findings

The team identified that the AFW test 3PT-M20A incorrectly used reference flow value ranges within plus or minus 5% of the reference value. NUREG-1482 guidance allows for variations of plus or minus 2% from the reference value. If the ranges are outside 2%, then documented justification is needed in the IST program documents. NYPA subsequently performed an extent of condition review for this concern and identified that the containment spray pump test 3PT-M17 also fell outside the acceptable range of plus or minus 2%. The team determined that the discrepancies had minimal significance because the test operated the pumps within the relatively flat regions of the pump curves, and therefore the flow reference value differences resulted in minimal impact to the testing results. NYPA initiated procedure changes to correct the flow reference values to the NUREG-1482 standards during the inspection period. The team concluded that the incorrect reference value constituted a violation of minor significance and is not subject to

formal enforcement action.

The team noted that system engineers were not routinely involved in surveillance tests. Notification of failed surveillance tests are provided to system engineers via the deviation/event report (DER) process, and IST data reflecting abnormal trends are forwarded to system engineers by memo. To evaluate whether system engineers were aware of surveillance test failures and abnormal trends, the team reviewed specific test concerns with system engineers and found them to be knowledgeable regarding the concerns.

The 50.54f project has been structured such that the system engineer is temporarily assigned to the 50.54f review project when his assigned systems are reviewed. This has enabled system engineering in identifying some discrepancies between surveillance tests and the Updated Final Safety Analysis Report (UFSAR). Examples of this included a recirculation pump test discrepancy regarding containment sump screen and an AFW test, regarding test methodology for verification of acceptable time to reach full flow.

Nonetheless, the team identified an additional example of a surveillance test discrepancy regarding testing of relay contacts within the steam generator blowdown isolation valve logic. Specifically, the contacts associated with the low generator level and main feed-pump turbine trip signals were not tested. However, the test had covered the signals for the ESF function for the Phase A isolation from safety injection (SI), anticipated transient without scram (ATWS) mitigation system actuating circuitry (AMSAC), loss of voltage, high radiation and high area temperature isolation. NYPA initiated DER 98-0637 and an associated reasonable expectation of operability basis for signals which were not tested. The interim determination of operability was adequate. Based on the operability of the other signals, which had been tested, no immediate safety concern existed.

The team determined that the contacts which were not tested were not Technical Specification test or other regulatory requirements. NYPA stated that they had intended that such contacts be routinely tested but that a review error during the Generic Letter (GL) 96-01 review effort had prevented identification of this discrepancy. NYPA stated that surveillance tests 3PT-M13A and 3PT-M13B will utilize the temporary procedure change process to test the omitted contacts.

Post maintenance tests developed by the performance group for scheduled preventive maintenance (PM) overhauls on the EDGs were found to be adequate. The tests included additional checks beyond the normal EDG Surveillance test.

c. Conclusions

The surveillance test program and post maintenance testing were generally adequate for the sample of surveillance tests and work orders reviewed. The team identified a minor error in several surveillance tests and contacts which were not being tested; however, the safety consequences were minimal and NYPA was correcting the test procedures.

E2 Engineering Support of Facilities and Equipment

E2.1 Control Room Habitability

a. Inspection Scope (IP 92903, 37550)

During design basis reconstitution of the control room heating, ventilation, and filtration

system, NYPA identified an inconsistency between the Technical Specifications and the control room booster fan design. Unresolved item 50-286/96-08-05 was opened as a result of the NRC's initial review of NYPA's finding. The team reviewed subsequent corrective actions within the larger context of conformance with the current plant licensing and design bases for control room habitability and engineered safety feature (ESF) component leakage outside of the primary containment following a loss of coolant accident.

Current Licensing Basis

10 CFR 50, Appendix A, General Design Criterion (GDC) 19 requires that adequate radiological protection shall be provided to permit occupancy of the control room during an accident without personnel receiving radiological exposures in excess of five Rem whole body or its equivalent to any part of the body for the duration of the accident. The limiting dose to the thyroid is 30 Rem. The licensing basis for control room habitability at IP3 is defined in the facility operating license and Technical Specifications (TS), docketed NYPA correspondence, NRC safety evaluation reports (SERs), and NYPA calculations, as follows:

- Operating license condition 2.L (Amendment #38), dated October 7, 1981
- TS 3.3.H.1 and 4.5.A.5, Control Room Ventilation System
- FSAR Sections 6.2, 9.9, and 14.3.5
- Letter IPN 81-20, Control Room Habitability Study, dated March 23, 1981
- Letter IPN 81-15, TMI Lessons Learned Supplemental Response, dated February 3, 1980
- Letter IPU 90219, ESF System Leakage, dated February 3, 1980
- Atomic Energy Commission SER for IP3, Section 15.4, dated September 21, 1973
- NRC SER, TMI Action Item III.D.3.4, Control Room Habitability, dated January 27, 1982
- NYPA calculation 6604-220-2, Control Room and Onsite Technical Support Center Habitability Study, dated December 1, 1980

The licensed maximum post-accident control room radiological dose to the thyroid is 28.8 Rem (FSAR Table 14.3-14G). NYPA calculated this value using the source terms and dose conversion factors in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites," and assumed: (1) 1400 cubic feet per minute (cfm) control room filtration flow rate (1000 cfm recirculation + 400 cfm outside air) to filter the air and maintain a slight positive pressure in the control room for 30 days, (2) initiation of ESF system flow outside the primary containment 29 minutes after the onset of a loss of coolant accident, and (3) an ESF component leakage rate of 0.705 gallons per hour (gph).

Control Room Filtration System Flow Rates

In September 1992, NYPA identified that the control room booster fans were sized for 2000 cfm at 3.0 inches of water column (" WC) while the Technical Specifications and surveillance procedures permitted up to 6.0" WC across the filter/charcoal adsorber unit. (Design Document Open Item IP3-CCRHV-315-12) In October 1993 NYPA used engineering judgement to determine that fan operation at 1400 cfm would permit differential pressure greater than 3.0" WC. However, the actual differential pressure at 1400 cfm was not known. The highest recorded differential pressure across the filter/charcoal adsorber train was less than 2.0" WC. NYPA took no further action to resolve the item until the NRC questioned its status in August 1996 (Inspection Report 50-286/96-08). NYPA then obtained vendor assistance to evaluate the operational limitations

of the system. Vendor calculation 83990.164-2-HVAC-092, "Control Room Air Filtration System Performance Calculation," concluded that the booster fans were capable of producing 1400 cfm at a static pressure of 3.0" WC.

In April 1997 NYPA expanded the scope of the calculation to determine the effect of operating time and filter loading on system flow rates. The calculation concluded that an initial minimum flow of 1580 cfm (1330 recirculation + 250 cfm outside air) at a maximum differential pressure of 2.98" WC was needed to maintain design basis flow rates for 31 days of continuous operation. Fan shutoff conditions were predicted to occur at a differential pressure of about 3.4" WC. In July 1997 NYPA performed a 10 CFR 50.59 safety evaluation, revised the acceptance criteria of procedure 3PT-C032, "Control Room Filtration System Functional," revised Section 9.9 of the FSAR, and verified the new system flow rates through testing.

b. Observations and Findings

NYPA's initiative that booster fan performance be evaluated for filtration train loading over time was perceptive. The vendor calculation used valid and well-documented design inputs and employed sound engineering principles. The surveillance procedure was revised promptly to incorporate the new limits. However, when accepted by NYPA on June 6, 1997, the vendor had not completed design verification and the calculation was in draft form. The final calculation (accepted by NYPA in August 1997) did not differ materially from the draft that was accepted erroneously in June. However, NYPA's acceptance and use of an unverified draft calculation was a lapse in plant configuration control and contrary to the administrative procedure for control of vendor-generated design documents (Section 6.3 of DCM-11). This is the first example of a **violation** of 10 CFR 50, Appendix B, Criterion V, which requires activities affecting quality to be accomplished in accordance with procedures. **(VIO 50-286/98-81-01)**

Prior to July 1997, the booster fan flow rate acceptance criteria of procedure 3PT-R032C did not accurately reflect the IP3 licensing basis. In addition, the TS and procedural limits on filter/charcoal adsorber train differential pressure were inconsistent with the booster fan design. TS 4.5.A.5.c requires that differential pressure across charcoal adsorber banks and filters at the design accident flow rate must be less than 6.0" WC, and that booster fan flow rate must be at least 90% of the accident design flow rate. Prior to July 1997, 90% of the fan accident design flow rate was 1260 cfm (0.9 X 1400 cfm). For five operating cycles between 1983 and 1997 one or both of the booster fan flow rates was less than 1260 cfm. The minimum 1000 cfm recirculation flow rate assumed in the design basis control room radiological dose calculation similarly was not achieved during that period. In July 1997, NYPA revised the design booster fan flow rate to 1730 cfm, making the TS-required flow rate 1557 cfm (0.9 X 1730 cfm). Failure to translate the control room filtration system design correctly into specifications and procedures was the first example of a **violation** of 10 CFR 50, Appendix B, Criterion III, "Design Control." **(VIO 50-286/98-81-02)**

On June 16, 1997, NYPA analyzed the radiological dose consequences to control room personnel assuming the worst-case historical flow rates (from December 1990 to June 1992). The analysis concluded that fan shutoff conditions would have been reached in 3.6 days, causing a loss both of filtered air flow and positive pressure in the control room. Using dose conversion factors from International Commission on Radiological Protection Factors Publication 30 (ICRP-30), "Limits for Intake by Workers," and the system hydraulic performance predicted in the draft vendor calculation, NYPA concluded that GDC-19 dose limits would have been met. The preliminary results subsequently were confirmed in NYPA

Radiological Engineering Technical Information Document TID-97-004, "Control Room Habitability Under Degraded HVAC Flow Conditions," dated August 7, 1997. NYPA concluded that the control room filtration system had been degraded but operable, and thus that the condition was not reportable to the NRC.

Use of ICRP-30 dose conversion factors was technically acceptable for the purpose of assessing system operability. However, plant operation with flow rates less than those assumed in the design basis calculation of record could have resulted in inability to maintain a positive pressure in the control room and postulated thyroid doses to control room personnel in excess of GDC-19 limits when calculated using the licensed design basis DCFs in TID-14844, "Calculation of Distance Factors for Power and Test Reactors." This constituted a condition outside the plant design basis and was reportable to the NRC per 10 CFR 50.72 and 10 CFR 50.73. Failure to notify the NRC within one hour or to submit a Licensee Event Report within 30 days was the first example of a **violation** of NRC reporting requirements. **(VIO 50-286/98-81-03)**

ESF Component Leakage

IP-3 license condition 2.L requires NYPA to reduce ESF system component leakage to as low as practical through a program of preventive maintenance, visual inspections, and integrated leak tests. The ESF component leakage program has been implemented by performance engineering through administrative procedure AP-35, "Integrity of Systems Outside Containment." Leakage limits are established in procedure 3PT-CO1, "Total Leakage Rate Monitoring Tabulation," and sub-tier surveillance procedures, and a running account of total component leakage has been maintained throughout each operating cycle. Prior to 1995 ESF component leakage was limited to 2.0 gph to be consistent with the residual heat removal (RHR) system leakage limit of TS 4.4.I.2. However, NYPA did not recognize, and the procedures did not reflect, that the TS applied only to off-site dose consequences, and that the leakage design input for control room habitability analyses and compliance with GDC-19 was 0.705 gph, based on the design basis calculation of record. As a result, the plant operated for prolonged periods with ESF component leakage rates exceeding 0.705 gph. For example, from 1987 to 1993 leakage exceeded 0.705 gph for over half the operating time. This condition would likely have resulted in control room thyroid doses higher than the plant licensing basis and GDC-19 limits using the other design inputs in the calculation of record.

NYPA missed an opportunity to have corrected the problem in 1995. During design basis reconstitution of the auxiliary building ventilation system, NYPA identified that the ESF component leakage rate used in the 1981 control room habitability analysis was inconsistent with TS 4.4.I.2. LER 95-016, dated September 11, 1995, documented two instances in 1990 and 1992 in which the TS leakage limit of 2.0 gph had been exceeded. As corrective action, NYPA performed calculation IP3-CALC-RAD-0021, "LOCA Radiological Analysis - Maximum ESF Component Leakage Requiring No PAB Exhaust Filtration," dated March 1, 1995. It concluded that the leakage rates corresponding to 30 Rem and 27 Rem to the thyroid were 1.34 gph and 1.09 gph, respectively. On June 11, 1996, NYPA submitted a TS amendment request (IPN 96-063) to the NRC proposing a leakage limit of 1.09 gph. In effect, the proposed amendment changed the scope of TS 4.4.I.2 to include control room habitability as well as off-site dose. A 10 CFR 50.92 significant safety hazards evaluation accompanied the request. Finally, as a compensatory measure, the acceptance criterion of procedure 3PT-CO1 was lowered to 1.0 gph.

Calculation IP3-CALC-RAD-0021 contained several assumptions different from those used

in the design basis calculation of record, including: (1) slightly lower initial reactor power level and source term, (2) higher containment fan cooler unit charcoal adsorber efficiency, (3) credit for containment spray, and (4) initiation of ESF flow outside the containment 14 hours following a design basis loss of coolant accident. As a result of the differences, the leakage rate corresponding to the GDC-19 limit (30 Rem) and the design basis control room dose (28.8 Rem) increased from 0.7 gph to 1.09 gph and 1.24 gph, respectively. NYPA did not perform a

10 CFR 50.59 safety evaluation for the calculation or the revised acceptance criterion in procedure 3PT-CO1. While the compensatory leakage limit of 1.0 gph was an improvement, it still permitted the licensing basis leakage assumption (0.705 gph) to be exceeded, which would likely have resulted in a post-accident thyroid dose greater than 28.8 Rem using license basis assumptions.

The NRC has not acted on the 1996 TS amendment request. Following discussions with the Office of Nuclear Reactor Regulation (NRR) in late 1997, NYPA revised calculation IP3-CALC-RAD-0021 (Revision 1, dated December 1997) to use the DCFs contained in ICRP-30. This resulted in a leakage value at the GDC-19 limit of 2.57 gph, and formed the basis for NYPA's assessment of control room ventilation system operability regarding previous high ESF component leakage rates and an intention to withdraw the 1996 TS amendment request. However, as of this inspection, NYPA had not withdrawn the TS amendment request, and no 10 CFR 50.59 safety evaluation had been performed on the revised control room habitability calculation.

On April 13, 1998, NYPA revised the component leakage acceptance criterion of procedure 3PR-CO1 back to the TS limit of 2.0 gph. The personnel who processed the change were apparently not aware that the current 2.0 gph TS limit for RHR system leakage did not apply to control room habitability. As a result, the accompanying 10 CFR 50.59 screening evaluation incorrectly concluded that no safety evaluation was required, as the revision merely reinstated the existing TS limit. Since the revision changed the leakage limit that had been imposed as a result of the 1996 TS amendment request, a 10 CFR 50.59 safety evaluation should have been performed to determine if the change created an unreviewed safety question. Failure to perform the evaluation was a **violation** of 10 CFR 50.59. **(VIO 50-286/98-81-04)**

During the first week of the inspection, on April 8-10, 1998, ESF component leakage reached about 1.5 gph due to degradation of a weld on a charging pump recirculation line. NYPA performed operability determination 98-016, which concluded that leakage less than 1.09 gph was needed to meet the design basis of the control room filtration system. The conclusion was not consistent with the 1981 design basis calculation assumption of 0.705 gph. Based on calculation IP3-CALC-RAD-0021, however, the control room ventilation system was considered operable as long as the leakage remained below 2.228 gph (corresponding to 27 Rem). NYPA concluded that the system was degraded but operable, and thus that the condition was not reportable to the NRC. Operation in excess of 0.7 gph would likely have resulted in exceeding the GDC-19 thyroid dose limit using licensing basis assumptions. Failure to report the excessive leakage as a condition outside the plant design basis was the second example of a **violation** of 10 CFR 50.72 and 10 CFR 50.73. **(VIO 50-286/98-81-03)**

Bounding post-accident control room radiological doses are calculated using a series of conservative design assumptions including release of the source term from the fuel into the containment atmosphere and recirculation sumps, and subsequently into the environment via ESF component leakage. As a result, using less conservative (and more realistic) assumptions, the actual consequences of the ESF leakage rates that exceeded the design basis value would likely not have been as severe as those predicted in the design basis analysis. NYPA's use of less conservative, but technically valid, assumptions was appropriate for the purpose of assessing control room filtration system operability. Nonetheless, failure to translate the licensed design basis leakage limit correctly into procedures initially in 1982, and subsequently in September 1995 and April 1998 was a second example of a **violation** of the design control requirements of 10 CFR 50, Appendix B, Criterion III. **(VIO 50-286/98-81-02)**

Following the inspection, on May 1,1998 NYPA's corrective actions included revising procedure 3PT-CO1 to be consistent with the licensed design basis limit of 0.705 gph, and performing a 10 CFR 50.59 safety evaluation for the change.

c. Conclusions

The quality of engineering work on the control room habitability and ESF leakage issues was weak. While these issues have a complicated regulatory and technical history, NYPA did not demonstrate a thorough, effective approach to resolving these issues, and the team concluded that four violations existed regarding design control, 10 CFR 50.59 review, reporting to NRC, and engineering administrative control. Often, corrective actions were implemented piecemeal and were not consistent with the licensing bases. Nonetheless, in many instances on these issues, NYPA engineering demonstrated good quality in calculations and effective insights into technical concerns.

E2.2 Post LOCA Recirculation Flow Capacity

a. Inspection Scope

The team reviewed the engineering response to DER 98-0281, associated with minimum post-LOCA (loss of coolant accident) recirculation flow, including Emergency Operating Procedure (EOP), ES-1.3, revision 13, "Transfer to Cold Leg Recirculation, Temporary Procedure Change (TPC) 98-0053, dated February 23, 1998, Operability Determination 98-015, dated April 10, 1998 and April 23, 1998, and the applicable Nuclear Safety and Environmental Impact Screen.

Background

During a review of post-LOCA long term cooling, system engineering questioned the adequacy of the long term core cooling flow parameters found in EOP ES-1.3. NYPA requested clarification regarding cold leg recirculation emergency core cooling system (ECCS) flow requirements from Westinghouse, with reference to its Nuclear Safety Advisory Letter (NSAL) 95-001. The NSAL had resolved a potential discrepancy relating to the amount of ECCS flow needed for core cooling during the cold leg recirculation period for a hypothetical large break LOCA. Analysis had shown that due to effects of extended periods of boiling that can occur in both the downcomer and lower plenum during injection and recirculation, a new criteria was established for minimum ECCS flow of 1.2 times the decay heat boiloff flow. Westinghouse concluded in the NSAL that IP-3 could not meet the 1.2 times decay heat boiloff requirement; however, detailed analysis at the time had shown that all plants were expected to have adequate core cooling during cold leg recirculation. Therefore, the issue was not considered to be a substantial safety hazard.

On February 25, 1998, Westinghouse clarified this issue in a new recommendation of flow, including a shorter possible switchover time from the injection phase to the recirculation phase of 13.1 minutes, in contrast to the 20.1 minutes previously assumed. In addition, a flowrate of 1.09 times decay heat boiloff was utilized to ensure adequate core cooling. A new minimum flowrate requirement of 638 gpm was established.

To change the flow requirement on February 23, 1998, NYPA issued a TPC to EOP ES-1.3, which increased the flow rate needed to 830 gpm, and bounded the minimum flowrate required by the Westinghouse letter, with additional margin for allowance of instrument error. The 830 gpm value had been a setpoint number, S.7, which previously existed in the EOP Setpoint Manual/Database.

The change was classified as an Advance Change TPC, a non intent change to be incorporated during the next procedure revision. The TPC had been approved via a Nuclear Safety and Environmental Impact Screen, which had concluded that the change was administrative in nature and did not require a Nuclear Safety Evaluation. The 50.59 screen focused on the problem at hand, or the need for an immediate increase from the existing 460/530 gpm flowrates described in ES 1.3. Because the setpoint of 830 gpm already existed in the EOP setpoint database, and it appeared to be conservative to increase the minimum specified flow in this case, NYPA concluded the change was acceptable without the need for performing a 50.59 safety evaluation.

b. Observations & Findings

The team identified several concerns regarding the EOP flow change from the regulatory change and design change points of view and judged both to represent violations. Specifically, from the regulatory change perspective, the UFSAR, Section 6.2, specifically stated the 530 gpm flow rate to the core in the recirculation mode. The TPC to ES-1.3 changed the operator action setpoint number discussed in the UFSAR (which addresses whether the low head recirculation flow is acceptable). Therefore, the change in the flow value created an inconsistency between the manual switchover process described in the UFSAR and the implementing EOP, ES-1.3. However, no 10 CFR 50.59 safety evaluation had been performed. The root cause of this error appeared to be the inadequate Nuclear Safety and Environmental Impact Screen. In addition, the screening error, and its failure to fully evaluate the change via the 50.59 safety evaluation process, prevented Plant Operating Review Committee (PORC) review of the issue. The failure to perform a 50.59

safety evaluation represented a **violation** of 10 CFR Part 50.59. **(VIO 50-286/98-81-05)**

The design change concerns on the revised EOP flow change related to the ability of operators to perform the specified procedural steps, recirculation pump net positive suction head (NPSH), increased radiological impact, and uncertain design basis, as discussed below.

1. The TPC had changed minimum low head flow from 460/530 gpm to 830 gpm. Considering a design basis event, i.e., large break LOCA concurrent with a loss of offsite power (LOOP) and a failure of an EDG, only one recirculation pump could potentially be available. In this event, ES 1.3 would specify containment spray and core flow via one recirculation pump.

   Limitations of 3000 gpm had been previously established due to cavitation concerns for the pump. The team calculated that the 1300 gpm of containment spray combined with 1660 gpm of injection flow would result in the operator having to maintain a narrow band of total flow between 2960 gpm and 3000 gpm.

   However, existing flow instrument loop inaccuracies, piping configurations and flow balancing considerations could prevent the operator from achieving this narrow band of total flow. For this situation, the procedure assumes that not enough low head flow is achievable and directs the operator to isolate the low head path and realign flow to the high head path, utilizing the SI pumps, which involves transfer of post accident flow outside of the containment. The previous minimum flow requirement of 460/530 gpm resulted in a potential band of control between 2200 gpm and 3000 gpm. This had created a much greater operating band, and significantly decreased the probability of exceeding 3000 gpm, while maintaining core flow and containment spray flow. Therefore, the internal recirculation mode was not threatened. While no specific design requirement prevented an operating band from 2960 gpm to 3000 gpm, the team judged that this operating band could have an adverse effect on the ability of the operators to achieve this operating configuration and consequently a reduced ability to effectively mitigate the consequences of the LOCA. As such this concern needed to be evaluated as part of the flow change.

2. Based on the above the external configuration appeared to be more likely during an event, and this configuration could have adverse radiological consequences. The high head path employs additional equipment, the SI pumps, and results in an external recirculation path versus an internal path within containment. The team questioned the effect on radiological consequences for a large break LOCA with external flowpaths, consisting of flow from a recirculation pump to the suction of the SI pumps within 30 minutes after the large break LOCA. The team recognized that the design basis included use of the residual heat removal (RHR) pumps within 30 minutes following a large break LOCA as a backup to the recirculation pumps. The RHR pumps would also use an external flowpath, being outside the containment.

   Nonetheless, the new potential external flowpath configuration had not been evaluated as an acceptable lineup or reflected in the UFSAR as part of the flow change. On April 23, 1998, NYPA provided a detailed, updated reasonable expectation of operability, (OD-98-015), which concluded the radiological consequences of the potential external recirculation flow path created by the TPC were acceptable. The April 23 OD reviewed the radiological consequences with respect to control room habitability, off-site doses, and access to plant equipment. Several areas were identified which needed access but would not be accessible due to high radiation for the external recirculation mode. The OD assessed and concluded that the consequences of the inaccessibility to these areas was acceptable.

3. The EOP flow change had been implemented without evaluating the NPSH affects on the recirculation pumps. Later, the April 23 OD evaluated NPSH considerations, and the team concurred that the detailed analysis performed had shown that a reasonable expectation of operability existed for the flow of 830 gpm. The team also reviewed the updated GL 97-04 Calculation for the NPSH, IP3-CALC-SI-02430 revision 2, which supported the operability conclusion. The calculation determined the revised NPSH margin to be 0.36 feet for single recirculation pump operation for core cooling flow concurrent with containment spray, an acceptable though marginal NPSH. The calculation assumptions were adequate.

4. The 830 gpm operator action setpoint had no retrievable calculations or analysis to support the number. NYPA stated that this number had been in the EOP setpoint database prior to 1989, but could not find any supporting analysis.

   NYPA had established a setpoint control program action plan, of which actions were planned for the collection, review and basis re-confirmation of all EOP setpoints. NRC Inspection Report 97-80 documented that the setpoint project had been forecast for completion by the end of 1997. Included in the plan was an effort for the reconfirmation of all existing EOP

setpoint basis information. However, the 830 gpm setpoint had not been reconfirmed via this plan and therefore, the basis for the number had not been established, although used in the TPC performed on February 23, 1997. The 830 gpm design flow had not been established via documented analysis as a reference bounds for design.

The team concluded that although a reasonable expectation of operability had been performed per the guidance of GL 91-18, the failure to assure that the design basis was correctly translated into procedures represented a violation of 10 CFR Part 50 Appendix, B, Criterion III. **(VIO-50-286/98-81-06)**

NYPA stated that an evaluation was on-going to determine the most effective value of long term recirculation flow to utilize in the ES 1.3 procedure. A 10 CFR 50.59 safety evaluation of the flow was planned.

c. Conclusion

The team concluded that NYPA had an inadequate evaluation as part of changing an EOP flow and that the inadequate regulatory review had violated 10 CFR 50.59, in that no safety evaluation was performed for the change. Also, the team concluded that the inadequate design review had violated 10 CFR, Appendix B, Criterion III (Design Control), in that significant aspects of the design (ability to operate, recirculation pump NPSH, and radiological consequences) were not evaluated prior to the change and the flow chosen had no basis (other than historical) or calculation to support it. The team noted that the NYPA effort to originally address the EOP flow concern had been well-intentioned and that the subsequent design evaluations had been thorough and accurate.

E2.3 Design Calculations

a. Inspection Scope

The team reviewed a sample of design calculations to assess technical quality, conformance with the provisions of American National Standards Institute (ANSI) Standard N45.2.11, "Quality Assurance Requirements for the Design of Nuclear Power Plants," and compliance with NYPA's design control procedures. The following calculations were reviewed:

IP3-CALC-CCW-02487	Tube Plugging Limits For CCW Heat Exchangers, Revision 0, dated June 23, 1997
IP3-CALC-HVAC-00200	Battery Room No. 34 Ventilation, Revision 0, dated April 5, 1991
TS Calculation #178	33 Battery Hydrogen Generation and Low Temperature Evaluation, dated May 18, 1989
IP3-CALC-ED-02563	Station Batteries Hydrogen Evolution, Revision 0, dated May 4, 1997

83990.164-2-HVAC-092 Control Room Air Filtration System Performance Calculation, Revision 3, dated September 9, 1997

IP3-CALC-RAD-0021 LOCA Radiological Analysis-Maximum ESF Component Leakage Requiring No PAB Exhaust Filtration, Revision 1, dated December 4, 1997

b. Observations and Findings

The reviewed design calculations had good quality. Assumptions were technically reasonable, appropriate design inputs from codes, standards, and relevant design criteria were identified and documented, and the required design reviews and verifications were performed.

However, two instances were identified in which NYPA's administrative design control procedures were not followed. In one case, a draft vendor-generated calculation was accepted by NYPA (see Section E2.1). In another case, partially superseded calculations were not appropriately dispositioned. Section 6.7.2 of Design Control Manual procedure DCM-2, "Preparation and Control of Calculations and Analyses (IP3)," states that as a result of revising or originating calculations, any superseded or voided calculations should be annotated. For calculations where portions are to be superseded or voided, the calculation is to be revised and the appropriate sections lined through. Calculation IP3-CALC-ED-02563 was performed to update the discussion of station battery hydrogen generation contained in Section 8 of the Final Safety Analysis Report. Previous battery hydrogen generation calculations IP3-CALC-HVAC-0200 and TS Calculation #178 were partially superseded, but NYPA did not revise or line through the appropriate sections. This is the second example of a **violation** of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which requires activities affecting quality to be accomplished in accordance with procedures. **(VIO 50-286/98-81-01)**

c. Conclusions

Design calculations had good quality, including technically acceptable assumptions and design inputs, and appropriate reviews and verifications. A violation was identified concerning two instances of non-compliance with administrative design control procedures.

E2.4 System Monitoring

a. Inspection Scope

The 1997 fourth quarter system reports for the component cooling water, containment isolation, containment recirculation fans and the main steam systems were reviewed. Procedures TSP-053, "System Performance Monitoring and Trending", TSP-050, "Systems Engineering Program" and TSP-057 associated with the Maintenance Rule were reviewed. In addition, five system engineers were interviewed pertaining to their responsibilities regarding the Maintenance Rule.

b. Observations & Findings

The Maintenance Rule procedures were adequate, responsibilities were properly defined, and engineers were cognizant of their responsibilities. Action plans associated with performance of the EDGs and the containment isolation system were technically adequate.

c. Conclusion

System monitoring and implementation of the Maintenance Rule for the component cooling water, containment isolation, containment recirculation fans and main steam systems was acceptable. System engineers were knowledgeable of their responsibilities regarding the rule.

E2.5 Operability Determinations

a. Inspection Scope

The team reviewed seven operability determinations (ODs) within the January to April 1998 time frame.

b. Observations & Findings

The technical quality of determinations was found to be generally acceptable. The team noted one example of a technically inadequate reasonable expectation of operability basis, performed on April 10, 1998, in response to concerns regarding a change made to Emergency Operating Procedure ES 1.3. The operability analysis performed within DER 98-0592 failed to address the full implications of the issue and is discussed in detail in Section E2.2.

c. Conclusion

The quality and depth of the operability determinations were generally acceptable. Operability conclusions were correct.

E2.6 Engineering Work Backlog

a. Inspection Scope

The team reviewed backlogged engineering work including performance trends, work prioritization, and the safety significance of specific work activities.

b. Observations and Findings

The following backlog engineering work was being trended: Action Commitment Tracking System (ACTS) items, overdue preventive maintenance on hold, non-outage and outage corrective and preventive maintenance on hold, temporary modifications, modification closeout, safety and engineering evaluations, and deviation/event reports (DERs). Monthly trend reports are reviewed by senior managers, and the status of key items is discussed by department managers weekly. In turn, individual engineers receive weekly assignments that are tracked by the department supervisors. The system engineers interviewed by the team knew what work was expected to be completed each week, and indicated that backlog reduction was a high management priority.

Since the last NRC review of this issue in April 1997 (Inspection Report 50-286/97-80), NYPA has reduced the backlogged engineering work. The NYPA 1997 goals for preventive and corrective maintenance items on hold were met. The trend for engineering evaluations was downward, and the individual backlogged items were not safety significant. The trend for DER closure was potentially misleading, as DERS were closed based on determination of

corrective actions and creation of corresponding ACTS items. Thus, the number of closed DERs did not indicate that the related problems were resolved or fixed. However, ACTS item closure also was trended.

While the general trend for ACTS item closure was downward, the number of high priority items was relatively static while less complex and lower priority items were being worked off. Nonetheless, the number of high priority items was relatively low and none were safety or risk significant.

Plant modifications showed the least progress and represented about one year's worth of backlogged work. Much of the work has been contracted out to free up site design engineering resources, and the modification review committee and the plant leadership team were focusing sharply on the cost/benefit aspects of plant improvement modifications with a view towards eliminating some of the proposed modifications.

c. Conclusions

The backlog of engineering work was being managed effectively, and the amount of backlogged work has been declining.

E3 Engineering Procedures and Documentation

E3.1 Modification Administrative Controls

a. Inspection Scope (37550)

The team reviewed selected modification procedures and interviewed NYPA engineering staff to verify that the proper procedural guidance had been established for implementing the modification activities and that all aspects of the modification were adequately evaluated.

b. Observations & findings

NYPA primarily controls design changes activities using station modification control manual (MCM) procedures. The team reviewed these modification procedures and determined that the procedures provided sufficient detail to ensure that permanent, minor, and temporary modifications are designed and implemented in a safe and controlled manner. The team verified that the procedures clearly prescribed post-modification testing and provided detailed guidance in the areas of 10 CFR 50.59 safety evaluation and design verification.

c. Conclusion

The team concluded that the NYPA modification procedures were clearly written and provided appropriate guidance.

E3.2 Plant Design Change Modification Review

a. Inspection Scope (37701)

The team reviewed three design changes implemented at IP3 during the last refueling outage. The team verified that the appropriate design criteria, design input requirements, and design evaluations were accurate and correct.

b. Observations and Findings

Minor modification, MMP97-3-293SGBD, was to prevent/mitigate water hammer in the steam generator blowdown (SGBD) system. The team reviewed the technical evaluation of this minor modification along with the design input summary evaluations. The writeup was technically sound and clear. The team also reviewed calculation No. IP3-CALC-SGD.-02506, "Steam Generator Blowdown Line Re-pressurization for Water Hammer Mitigation." This review consisted of the design basis, assumptions, and design inputs to the calculation. All calculation assumptions were taken from Table 9-4.2 of IP3 FSAR. The final calculation showed that the piping volume was 11.1 cubic feet requiring one hour and 23 minutes to fill the piping at 1 GPM fill rate. The team verified that the system line up for this 1 GPM flow rate was incorporated into operating procedure SOP-SG-1, Rev, 8/26/97. The team found that this calculation was technically sound and was accurately reflected in the operating procedure.

Modification MMP-94-3-250 was to replace the Masoneilan displacer type level transmitter with a new differential type transmitter installed below the pressurizer relief tank above El. 46'-0". This eliminated higher radiation exposure to personnel working inside the containment due to the longer time involved in the calibration and/or maintenance work. The team reviewed work order No. 94-04251-03 for the post modification testing of the pressurizer relief tank transmitter. The team verified the testing calibration results with acceptance criteria delineated in the work order and found it to be technically correct.

Modification MMP-97-3-299SI was to address the difference between the installed and the replacement motors of the residual heat exchanger outlet isolation valves, (SI-MOV-899A/B) in the SI system and to ensure that the replacement motors were suitable for the application. The team reviewed the design inputs and assumptions to the modification and the test requirements, a functional performance test (3PT-Q22, Rev. 12), including the test results and acceptance criteria.

Engineers and engineering managers interviewed by the team were knowledgeable with regard to their assigned modifications and the modification process.

c. Conclusion

The team concluded that the three modifications selected for review were properly prepared in accordance with appropriate procedures and engineering and demonstrated sound technical analyses.

E3.3 Safety Evaluation (10 CFR 50.59) Review

a. Inspection Scope (37001)

The team reviewed selected modification packages and Updated Final Safety Analysis Reports (UFSAR) changes to assess NYPA's compliance with 10 CFR 50.59.

The team reviewed Modification Control Manual (MCM) Procedure No. 4, which NYPA has used for conducting safety evaluations (SEs). The following sample of safety evaluations were reviewed:

NSE 92-3-264RES, Rev.1, Core Reloads for Cycle 10
NSE 95-3-275RCS, Rev.0, In Vessel and Ex-Vessel Dosimetry
NSE 96-3-250FHS, Rev.0, Manipulator Crane Upgrade
NSE 96-3-289FHS, Rev.0, Fuel Transfer System Upgrade
NSE 97-3-207RPI, Rev.0, Upgrade/Change CRMD/RPI Connectors
NSE 96-3-057PZR, Rev. 0, Replacement of Pressurizer PORVs
NSE 96-3-146CCW, Rev. 0, Replacement of Temperature Control Valve
NSE 97-3-299SI, Rev. 0, Replacement of Motor for SI-MOV-899A
NSE 97-3-293SGBD, Rev. 0, Steam Generator Blowdown Water Hammer
NSE 96-3-275ED, Rev. 1, Realignment of Power to Auxiliary Component Cooling Water Pump 32 & 33

Also, the team reviewed four of NYPA's annual report submittals for safety evaluations per 10 CFR 50.59(b)(2) from January 22, 1992 to January 23, 1996.

b. Observations and Findings

For the modifications reviewed, NYPA had prepared acceptable 10 CFR 50.59 safety evaluations to reflect the changes to the affected operating plant systems. Each modification involving a change to the facility as described in the UFSAR had been evaluated against the criteria defined in 10 CFR 50.59 to determine if the margin of safety was reduced.

The team noted weak guidance in addressing the last criterion of 10 CFR 50.59, the margin of safety, in that the MCM 4 procedure was unclear when the pre-existing margin of safety for a design change is not stated in the licensing bases (i.e., Technical Specification and FSAR). Further, NYPA's safety evaluations did not provide a description of the margin of safety changes in sufficient detail for a reviewer to independently conclude that the margin of safety had not been reduced, though the team did not find any safety evaluations that were likely to have reduced the margin of safety.

During the subsequent discussion with NYPA's engineering staff, the team noted that if the Technical Specification did not specifically address the margin of safety for the design changes, a determination should be made to establish what constituted the "pre-design change" condition or basis (as discussed in NRC Inspection Manual, Part 9900, "10 CFR 50.59 Guidelines"). NYPA initiated the following two Action Commitment Tracking System (ACTS) items that were scheduled to be completed before June 30, 1998.

1. ACTS item No. 34076 to enhance MCM 4 training to include a more detailed discussion of operating margins, margin of safety, and application to 10 CFR 50.59 safety reviews.

2. ACTS item No. 34124 to revise MCM 4 NSE Question 7 to add the UFSAR and SER.

The team found that NYPA's annual reports of changes under 10 CFR 50.59 were acceptable.

c. Conclusion

The team concluded that NYPA's safety evaluations of modifications and UFSAR changes were acceptable. The team noted that documentation of the margin of safety was sometimes weak, in that it was insufficient to enable an independent determination, and a deficiency in the review guidance regarding margin of safety was identified by the team and addressed by NYPA.

E3.4 Temporary Modification Review

a. Inspection Scope (37702)

The team reviewed the temporary modification (T-mod) process, which is controlled by NYPA administrative procedure, AP-13, Rev. 21, and a sample of T-mods, involving three T-mods in plant and documentation for six older T-mods.

b. Observations and Findings

Procedure AP-13, Rev. 21 clearly defined responsibilities and provided sufficient detailed instructions for proper implementation.

The three T-mods reviewed in the plant were properly installed in accordance with established administrative procedure AP-13 and had appropriate 10 CFR 50.59 applicability reviews. The status of these three T-mods installed in IP3 have been reviewed and evaluated per AP-13 administrative procedure requirements.

The team found that NYPA had periodically reviewed the status of the six old (as far back as 1990), primarily nonsafety-related temporary modifications in PORC meetings. There was evidence that the system engineering manager and plant management had placed emphasis on managing and minimizing T-mods.

c. Conclusion

Currently installed T-mods including documentation were acceptable.

E3.5 Safety Evaluation Program Training and Qualifications

a. Inspection Scope (37001)

The team interviewed some engineering staff members to assess their understanding of safety evaluations preparation. In addition to the discussions with the training department manager and training supervisors, the team reviewed training materials for 10 CFR 50.59 evaluators and reviewers.

b. Observations and Findings

The lesson plans and classroom handouts were of appropriate scope and depth. The plant system engineering staff members interviewed by the team were found knowledgeable of the 50.59 processes and the scope required for safety evaluations.

The team identified a potential problem related to control of qualified evaluators. Specifically, since April 3, 1996, administrative procedure "Review and Approval Authorizations," AP-2.1, has specified that individuals had to complete 3-day classroom training and an actual 10 CFR 50.59 safety evaluation proficiency demonstration before the PORC to be qualified as a 50.59 evaluator or reviewer. However, several of the reactor engineering staff at the White Plains office were not familiar with the nuclear safety evaluation process nor did they understand the 10 CFR 50.59 criteria. The last requalification for safety evaluation or 50.59 training for this engineering staff was over four years ago, prior to use of the new proficiency demonstration. Yet, they were listed under the "10 CFR 50.59 Qualified Reviewer/PORC Member" administrative approval

matrix. This matrix is intended to show the engineering staff who are prepared to perform safety evaluations or act as an independent reviewer (ANSI 45.2.11) for safety related activities at IP3.

NYPA determined that none of the involved engineers had performed any safety evaluations for which they were not qualified and initiated steps to have the list of qualified 50.59 reviewers include only those individuals who meet the current procedure conditions.

c. Conclusion

NYPA had a satisfactory program for training employees on the requirements of 10 CFR 50.59 and preparing safety evaluations. The contents of the training were appropriate and supported the 10 CFR 50.59 evaluation process. The 50.59 training and qualification appeared to have been properly applied for the plant engineering staff; however, the 50.59 training qualification process for the engineering staff at NYPA's White Plains Office was found to need improvement.

E7 Quality Assurance in Engineering Activities

E7.1 Problem Identification Process

a. Inspection Scope (40500)

The deficiency reporting systems at IP-3 were the deviation and event report (DER) system and the problem identification description (PID) system. The team performed plant walkdowns, interviewed personnel, and reviewed QA audits and PIDs and DERs to assess the effectiveness of NYPA's problem identification.

The process for identifying equipment problems was the PIDs contained in station directive SPO-SD-01, "Work Control Process," and DERs were for problem identification as described in administrative procedure AP-8, "Deviation & Event Reporting and Operability Determination Procedure." AP-8, Revision 37 procedure had been cumbersome to use for problem identification, and the team was informed that AP-8 procedure was undergoing a revision to decentralize the procedure, which appeared to be an appropriate action. SPO-SD-01, specifies that a PID be generated when an equipment deficiency is found and that the deficiency to be entered into the reliability on-line maintenance environment (ROME) computer database.

b. Observations and Findings

Interviews of station personnel revealed that they were familiar with both the PID and DER processes for reporting problems and were not reluctant to use the systems.

During plant walkdowns of the primary auxiliary building (PAB), intake structure, EDG rooms, and turbine building, the team did not identify any equipment or plant deficiencies which had not been previously identified and appropriately documented on a PID tag. The PID tags observed in the field were verified against the ROME data base and the work request data base to ensure that corrective actions were being adequately tracked and work activities planned to resolve the identified deficiencies. The team determined that equipment deficiencies were appropriately identified and planned corrective actions were effectively tracked by the ACTS process and work requests were adequately developed to resolve identified problems. The plant equipment appeared to be properly maintained and in good working condition.

c. Conclusions

The overall material condition of plant equipment and problem identification was good. The equipment appeared to be properly maintained and in adequate working condition. Deficiencies were being appropriately identified and planned corrective actions tracked to correct the problems.

E7.2 Root Cause Analysis

a. Inspection Scope (40500)

Completed DERs were reviewed to determine the effectiveness of NYPA's root cause evaluation capabilities and proposed corrective actions to resolve identified deficiencies.

b. Observations and Findings

By reviewing a sample of completed DERs written in 1997, the team determined that DERs had been categorized appropriately. Also, 1997 root cause evaluations averaged 27 days to completion, a reasonable time frame given the categorization. The evaluations were technically sound, proposed corrective actions were adequately developed, and actions to resolve the problems were properly tracked by ACTS.

The team determined that root cause evaluations adequately assessed the contribution of human performance for the events. Human performance errors are monitored, trended and provided to station management on a monthly basis by the Operations Review Group (ORG).

c. Conclusions

DERs were appropriately categorized. Evaluations were technically sound, including human performance, proposed corrective actions were adequately developed, and actions to resolve the problems were being properly tracked.

E7.3 Corrective Action Timeliness and Effectiveness

a. Inspection Scope (40500)

QA audit reports, performance indicator quarterly trending reports and open/closed ACTS items were reviewed to assess both the timeliness and effectiveness of corrective actions.

Past NRC inspections in 1997 documented examples of inadequate corrective actions resulting in nine corrective action violations in various areas, such as operations, engineering and work controls, and the thoroughness of corrective actions and extent of condition reviews had been found to be inadequate. Also, NYPA's QA audit A98-04 "1997 QA Evaluation of IP-3 Staff Performance," the Safety Review Committee (SRC) and the recently completed self-assessment of corrective actions by the Operations Review Group in April 1998 had identified that attention was warranted in the area of effectiveness of corrective actions, in that, the timeliness and effectiveness of corrective action implementation continued to be major factors in the cause of repeat events at IP-3. These groups determined that the cause of repeat events at IP-3 was partly due to timeliness and effectiveness of corrective action implementation. In response, NYPA requested an INPO assist visit in the area of corrective action effectiveness, which took place in March-April 1998. NYPA received the results of INPO's evaluation and was assessing the report at the end of this inspection.

b. Observations and Findings

Based on review of open and closed actions items, the team found that corrective actions were generally progressing and did not identify any additional corrective action concerns beyond the issues previously identified by NYPA and NRC.

c. Conclusions

Problems regarding corrective action effectiveness and timeliness have previously been identified by NYPA and NRC, but the team's review of corrective action program items did not find additional problems.

E7.4 Onsite and Offsite Safety Review Committee

a. Inspection Scope (40500)

The team evaluated the effectiveness of the safety committees by reviewing committee minutes and attending meetings. The team also reviewed the procedures for the Plant Operating Review Committee (PORC), AP-2, Revision 20, and the Safety Review Committee (SRC) procedure manual.

b. Observations and Findings

The team attended routine and reactive Plant Operations Review Committee (PORC) meetings, and found that nuclear safety evaluations, temporary modification packages, deviation event reports, and licensee event reports (LERs) were adequately reviewed to determine if an unreviewed safety question existed. Agenda items were well presented and focused on safety and regulatory considerations. The PORC members demonstrated a strong questioning attitude during the discussion of agenda items.

The team reviewed the minutes of the Safety Review Committee (SRC) meeting for January, 1998. The SRC reviewed safety evaluations, changes to procedures, changes to Technical Specifications, reportable events, and NRC inspection reports. As part of the SRC review of safety evaluations, committee members determined that certain NSEs were weak and returned them for revision. Based on the SRC meeting minutes the SRC was concerned regarding the effectiveness of IP3 corrective action and quality of modification packages.

c. Conclusions

The PORC and SRC were appropriately carrying out their assigned oversight functions. Both committees displayed a strong questioning attitude.

E7.5 Self-Assessment Program

a. Inspection Scope (40500)

The team reviewed NYPA's Plant Standard, PS-01.07, "Site Self Evaluation", and selected self assessments of departments to determine the effectiveness of NYPA's self assessment program. The most recently completed self assessments for engineering and the operations review group (ORG) and the QA evaluation of the IP3 staff performance were reviewed. The team attended the monthly presentation of selected self assessments.

In May 1997 NYPA implemented a formal self evaluation program with a coordinator appointed in January 1998. Each month two or three departments have presented their self evaluations to the plant leadership team.

b. Observations and Findings

The completed self assessments reviewed by the team were self critical and probing. In each of the self assessments reviewed, the action commitment tracking system (ACTS) was used to track corrective actions that evolved from the self assessments.

Also, as noted in Section E7.3 above, problems associated with the effectiveness of corrective actions had been identified by the Safety Review Committee and QA audits, a good indication of facility self-assessment.

c. Conclusions

The completed NYPA self assessments were critical and self-probing. NYPA showed good self-assessment in self-identifying problems related to the effectiveness of corrective actions.

E7.6 Operating Experience Feedback

a. Inspection Scope (40500)

The team reviewed NYPA's response to several NRC information notices and Westinghouse nuclear safety advisory letters (NSAL) to evaluate the adequacy of the program and interviewed the responsible person.

NYPA's Operations Review Group (ORG) has been responsible for maintaining and distributing industry operational experience (OE) information. The OE is gathered each day from the nuclear network and has included information from INPO, NRC, and industry. Information is screened by ORG to determine if the items or issues are applicable to NYPA and distributed widely. An ACTS item is generated when a technical review is appropriate.

The team reviewed NYPA evaluations for the following NRC Information Notices (INs) to assess engineering quality and timeliness:

IN 98-07	Offsite Power Reliability Challenges From Industry Deregulation
IN 97-78	Crediting Of Operator Actions In Place Of Automatic Actions and Modifications Of Operator Actions, Including Response Times
IN 97-76	Degraded Throttle Valves In Emergency Core Cooling System Resulting From Cavitation-Induced Erosion During a Loss-Of-Coolant Accident
IN 97-21	Availability Of Alternate AC Power Source Designed For Station Blackout Event
IN 97-25	Dynamic Range Uncertainties In The Reactor Vessel Level Instrumentation

IN 97-01	Improper Electrical Grounding Results In Simultaneous Fires In The Control Room And The Safe-Shutdown Equipment Room
IN 96-13	Potential Containment Leak Paths Through Hydrogen Analyzers

b. Observations and Findings

The team determined that NYPA's review of operating experience information was generally timely and effective, and that the corrective actions in response were appropriate. Some of the technical issues discussed in the INs had been pre-dated by information from other industry sources; e.g. the "Nuclear Network" or NSALs. Thus, NYPA had initiated corrective actions well in advance of the NRC IN's issuance.

In its response to IN 96-13, NYPA identified that containment hydrogen analyzer leakage test procedures 3PT-R69A and 3PT-R69B were inadequate in that the entire pressure boundary from the outlet of the supply containment isolation valve to the inlet of the return containment isolation valve was not tested. The untested portion of the system represented a containment bypass leakage path when the containment isolation valves are opened for post-accident atmosphere monitoring. The system had been tested correctly following initial installation in 1992, but the surveillance procedures had not been revised correctly to ensure that the two solenoid valves inside the analyzer cabinets were open during the leakage test. NYPA revised the procedures and satisfactorily performed the leakage tests promptly after having reviewed the IN. Failure to adequately test the hydrogen monitor penetrations was a violation of 10 CFR 50, Appendix J, for Type C testing. However, the failure constitutes a violation of minor significance and is not subject to formal enforcement action.

c. Conclusions

NYPA's technical evaluations of industry operating experience information were timely, detailed, and effective.

E7.7 Speakout Program

a. Inspection Scope

The employee concerns (Speakout) program was assessed by conducting interviews of individuals working at IP-3, including the Speakout program administrator and reviews of program guidance and documented employee concerns received by this program in 1997.

b. Observations and Findings

The formal systems at IP-3 for identifying plant problems or deficiencies are the DER and PID systems. However, NYPA has another system called the Speakout program which is intended to be a confidential alternative for employees to raise concerns.

All class 1 (most safety significant) Speakout concerns raised in 1997 were screened by the team to determine the safety significance of issues raised. The team selected specific safety concerns for detailed review to verify that the program was properly assessing and resolving employees safety concerns. Investigations were timely and corrective actions were consistent with the safety significance of the problems raised.

c. Conclusions

NYPA was processing and resolving employees' safety concerns.

E7.8 System Engineering Self Assessments

a. Inspection Scope

The Quality Assurance (QA) Division Audit Report for IP-3 Design Control (A98-05 I), and selected system engineer module self assessments were reviewed. System Engineer self assessment areas reviewed were maintenance rule, chronic problems, communications and nuclear safety evaluations (NSEs).

b. Observations & Findings

Self assessments were found to be self critical and comprehensive for the areas reviewed. The recently completed QA assessment of design control had identified a concern with the EOP setpoint control process, although no technical issues were identified. The team, prior to review of this assessment, independently identified a concern with the use of a setpoint in the EOP data base concerning long term recirculation flow. See Section E2.2 of the report for details. Accordingly, this demonstrated good problem self-identified action.

c. Conclusion

NYPA's self assessments of selected system engineering functional areas were self critical and comprehensive. The QA audit of design control was effective and consistent with team findings.

E8 Miscellaneous Issues

E8.1 (Closed) Inspector Followup Item 50-286/96008-05: Maximum control room booster fan differential pressure. This item involved an apparent discrepancy between the control room booster fan design differential pressure of three inches of water column (3.0" WC) and the maximum 6.0" WC permitted by the Technical Specification. Violations of requirements identified by the team concerning this item are documented in Section E2.1 of this report and subsequent NRC followup will occur under those numbers.

E8.2 (Closed) Inspector Followup Item 50-286/96012-04: Periodic testing of EDG exhaust fan flow rates. The exhaust fans are needed to support EDG operability by maintaining room temperature below the design limits of the emergency switchgear. Calculation IP3-CALC-HVAC-00408 had assumed an exhaust fan flow rate of 35,000 cubic feet per minute, but no periodic testing had been performed to verify that the fans continued to achieve the required flow rate.

NYPA determined that flow rate testing was not required based on the following: (I) Fan testing is not described in the Technical Specifications, TS bases, or the Final Safety Analysis Report; (ii) the safety-related EDG ventilation system fans, dampers, and louvers are included in the preventive maintenance program; (iii) the fans and motors are part of the vibration monitoring program; and, (iv) system performance is monitored during monthly EDG surveillance tests. The team concluded that the existing practices adequately maintained the design functionality of the EDG exhaust fans and that periodic flow rate testing was not required.

E8.3 (Closed) VIO 50-286/97080-02: The team reviewed the proposed and completed corrective actions for the failed Fuel Storage Building ventilation damper identified during the 97-80 NRC inspection. NYPA had replaced the damper motor for the 32 fresh air damper unit per DC 97-3-200. In addition, operating procedures were revised to provide direction to the operators for damper positions and corresponding light indications.

Inspection Report 97-80 had identified that the thermal overload heaters for SI pump suction valves had not been promptly replaced. NYPA changed the thermal overload heaters for both valves MOV-887A and MOV-887B per DC 97-03-206. In addition, tracking system item 26491 was initiated to revise procedure DCM-2 to prevent a repeat occurrence.

Based on the actions taken in response to the violation, the subject violation was closed.

E8.4 (Closed) Inspector Followup Item 50-286/97080-05: Component cooling water (CCW) system flow balance. The acceptance criteria specified in special engineering test procedure ENG-366, "Component Cooling System Flow Balance," dated April 7, 1989, were established by Westinghouse report WCAP-12313, "Safety Evaluation For An Ultimate Heat Sink Temperature Increase To 95 F At Indian Point 3." However, the test procedure did not state how flow instrument accuracy was factored into the acceptance criteria.

NYPA established a three percent instrument error in the test procedure to ensure that the field measurements represented best estimate flow values. Westinghouse did not consider the error in its system flow model explicitly. However, the modeling approach assumed that flow to each component was at its high and low test range (balancing range) which, combined with modeling of the component cooling water pumps at their high and low performance limits (analysis range), ensured conservative results. The team found the approach to be acceptable.

E8.5 (Closed) Inspector Followup Item 50-286/97080-06: EDG starting air receiver capacity. NYPA had no documented analysis showing that the EDG starting air system receivers were sized adequately to maintain the ventilation system fans and dampers operable and to meet EDG starting air requirements.

The team walked down the starting air systems and reviewed calculation 83990.003-2-

EDG-212, "EDG Compartment Ventilation System Operating Time Calculation Issue," dated April 1, 1998. The safety-related, seismically qualified air systems conformed to UFSAR descriptions and were in good condition. Based on historical leakage rates, NYPA calculated that about 20 hours of operation would be available before the exhaust ventilation system would begin to fail. Procedure SOP EL-15 was being revised to reenergize the starting air compressors and replenish the air supply within four hours of a design basis event. The team found NYPA's actions to be adequate.

E8.6 (Closed) Inspector Followup Item 50-286/97080-07: Component cooling water (CCW) heat exchanger performance monitoring. This item involved NYPA's Generic Letter 89-13 commitment to periodically inspect and clean the CCW heat exchangers. The heat exchangers were installed in 1992 and the inspections initially were assigned a 270-day frequency. However, following inspection in 1994, preventive maintenance was not performed again for another 31 months (July 1997).

The deferral from the original periodicity was based on the operational results of inspections and tests of other service water system heat exchangers and good operational performance of the CCW heat exchangers. This judgement was confirmed by the results of inspections and eddy current testing of the heat exchanger tubes performed during the 1997 refueling outage. The team reviewed the preventive maintenance change request dated March 27, 1998, and concluded that NYPA adequately justified changing the inspection interval to 24 months. The team also reviewed NYPA's responses to Generic Letter 89-13 documented in letters IPN-90-004, dated February 6, 1990 and IPN-92-040, dated September 9, 1992. No deviations from NRC commitments were identified.

E8.7 (Closed) Inspector Followup Item 50-286/98080-02: NYPA had not completed its evaluation of motor-operated valve (MOV) stem coefficient of friction in the open direction to validate an assumption that a value of 0.20 bounds load sensitive behavior effects.

The team reviewed report IP3-RPT-MULT-01279, "Evaluation of Coefficient of Friction For Generic Letter 89-10 Motor Operated Valves," Revision 3, dated February 17, 1998. NYPA statistically analyzed the opening strokes of 31 dynamically tested valves and calculated the mean plus two standard deviations of the data. The resulting coefficient of friction was 0.178, validating the bounding MOV program assumption.

V. Management Meetings

X1 Exit Meeting Summary

The team presented the preliminary inspection results to members of NYPA staff and management at the conclusion of the inspection on April 24, 1997. The findings were evaluated the following week by the inspection team and a final exit meeting was held at the IP3 site on May 1, 1998. NYPA acknowledged the findings presented.

The team asked NYPA whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

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* Indicates present at the exit meeting of 5/1/98