123 Main Street
White Plains, New York 10601
914 681.6240

New York Power Authority

John C. Brons
Executive Vice President
Nuclear Generation

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

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Safety Systems Outage Modifications Inspection
(Installation and Testing) Report 87-015

Reference 1. Letter from Mr. S. A. Varga to Mr. J. C. Brons, dated October 21, 1987, entitled: "Safety Systems Outage Modifications Inspection (Installation and Testing) 50-286/87-015."

Dear Sir:

Reference 1 transmitted the results and conclusions of the installation and test phase of the NRC Staff's Safety System Outage Modification Inspection (SSOMI) at Indian Point 3. We have reviewed the report in detail and have had the opportunity to discuss it with the Staff during a meeting at Indian Point 3 on November 13, 1987. The Authority considers the findings of the inspection, as described in Reference 1, to be significant and has undertaken numerous initiatives aimed at addressing the programmatic issues. The attached information provides an overview of these initiatives in response to Reference 1.

The most significant change to convey regards a major reorganization of the Power Authority's engineering resources. These changes have been instituted in response to several needs including those identified in the design phase of the SSOMI. Attachment 1 includes details of the organizational changes which will affect the Nuclear Generation Department.

Attachment 2 provides additional information and/or clarification for several specific details presented in Section 2 of the report.

Reference 1 identified several equipment operability concerns which the Staff considered startup issues. The Authority responded to these issues and resolved the matters to the satisifaction of the Region I staff prior to startup.

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Therefore, the Authority considers the issues closed. Attachment 3 provides a summary of the actions taken to resolve the identified concerns.

It should be noted that all Environmental Qualification issues were resolved prior to start up. A conscientious effort was made to resolve, prior to start up, any issues regarding material conditions in the plant which were identified during the course of the inspection.

As we explained to the NRC personnel during the meeting of November 13, 1987, the SSOMI inspection demanded a significant expenditure of resources on the part of the Authority. The inspection findings are significant, however they are acknowledged as opportunities for improvement in the programs supporting operation of Indian Point 3. The improvements resulting from changes instituted following the inspection will benefit both our nuclear facilities and the lessons learned will be applied to our entire nuclear support organization.

We remain available to discuss the results of the SSOMI installation and test phase and our actions described herein.

Very truly yours,

John C. Brons
J. C. Brons
Executive Vice President
Nuclear Generation

cc: Resident Inspector's Office
Indian Point Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 377
Buchanan, NY 10511

Joseph D. Neighbors, Sr. Proj. Mgr.
PWR Project Directorate I-1
Division of Reactor Projects - I/II
U.S. Nuclear Regulatory Commission
7920 Norfolk Avenue
Bethesda, MD 20014

U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Attachment 1

Safety Systems Outage Modification

Inspection 50-286/87-015

Installation and Test Phase

Response to NRC Staff

Conclusions

The Authority has reviewed the results of the Safety System Outage Modification Inspection to determine the appropriate response to the programmatic issues which it identified. The actions undertaken, as described herein, also address certain findings identified in the most recent Systematic Assessment of Licensee Performance (SALP) and the Operational Assessment Team (OAT) inspection conducted at the start of the last refueling outage. Specific findings in the SSOMI report are addressed separately and identified by section number from the report. We believe that each separate finding does not represent a concern, but the findings taken collectively are indicative of a weakness worthy of concern which is addressed in this response. On the other hand, we have noted several instances documented in the report which are inaccurate and have provided clarified information for your use.

The Authority has recognized several root causes which we believe contributed to the weaknesses and practices identified in the SSOMI report. First, we routinely have undertaken significant engineering projects at our nuclear facilities. Such projects were the result of company identified needs, commitments made to external organizations, or dictated by hardware failures. In some cases the totality of these commitments have over extended our abilities to control. Secondly, the technical resources of the Authority were not well organized to provide for efficient and timely modifications.

To correct these root causes, the Authority has committed to review all plant modifications scheduled for outages and identify the Company resources necessary to complete the effort. The purpose of this review is to better match available resources to needs thereby ensuring the end products, that is the plant modifications, are timely and complete. To this end, the Authority is proceding with efforts to institute an integrated schedule for the Indian Point 3 facility. This process will facilitate the scheduling of regulatory commitments in concert with licensee identified improvements and maintenance. The process will also necessitate a critical review of all commitments, modifications and maintenance to identify the resources required to complete the work.

As discussed in the Authority's response to the "Design" portion of the SSOMI (J. C. Brons letter to NRC, dated, November 13, 1987) the Authority has also initiated action to unify design and modification control procedures between the corporate office and the nuclear plants. This effort has addressed root causes of problems and identified areas for improvements, particularly in the area of design change control. This change control program will be a part of the corporate Design and Modification configuration management program and will define the activities of all Power Authroity organizations performing design and modification work for the nuclear facilities. The program manual will define the responsibilities and interface of all departments involved in design and modification work. The main elements of this effort are scheduled to be completed prior to the next refueling outage.

The design control and configuration management program consists of four major areas of control: design bases, design standards, design control and modification control. For each of these areas, a separate manual with implementing procedures is being developed. The policy defines the responsibility of various Authority organizations. The Authority has also recently been reorganized to improve the efficiency and accountability of the technical groups. The reorganization has resulted in the shifting of resources such that all engineering, construction and operations associated with the nuclear facilities are focused within one department under the control of the Executive Vice President of Nuclear Generation. This single change will result in several benefits including improved modification design packages for outage work, improved matching of resources to commitments, a reduction in field changes which potentially compromise the original design, and better adherence to work schedules. By concentrating technical and construction resources within the Nuclear Generation Department, improvements in the overall quality of support for the two nuclear facilities will result.

A management directive has been prepared and distributed to all site personnel reaffirming the Authority's commitment to adherence to procedures. All Department Heads and Managers will review this directive with their respective staffs and assume responsibility for implementation. We will design a station goal to appraise management of instances of procedural non-conformance with the aim of aggressively reducing the number of instances of procedural non-compliance.

The Authority has undertaken a significant effort to revise and revamp procedures by which maintenance and modifications are performed at Indian Point 3. Administrative Procedure 22.1, Maintenance Procedure Controls, has been reviewed and is being revised. The procedure will now cover in much greater detail the applicability and use of work step lists, check lists, and specific maintenance procedures. With respect to work step lists, AP-22.1 will detail the review requirements for step lists and include a formal mechanism for changing approved step lists. Changes to a work step list will be reviewed and approved by at least maintenance and quality assurance personnel. Changes to AP-22.1 will be completed by July 1, 1988.

All maintenance and I&C procedures are undergoing a review to identify areas for improvement in the level of detail provided to the user. Human factor issues will be included in the upgrades. The intent of the procedure review is to identify gaps in procedures which, due to plant personnel familiarity with the particular task, have not been recognized to date. Furthermore, additional details derived from equipment or system technical manuals, when available, will be incorporated into procedures when appropriate. Quality attributes will also be identified and included in procedures to provide additional guidance to the procedure user and to facilitate quality control functions. Procedure formats will be defined by April 1, 1988. Procedure revisions will be completed within two years in accordance with the biennial review schedule for the procedures.

The Authority recognizes the benefits that can be realized through standardization. This concept is being applied to the installation engineering of plant modifications and maintenance. Specifications are being developed to standardize various aspects of the design and installation process. Electrical specifications which address cable pulling, termination and splicing are under development. Mechanical specifications which address anchor bolt and pipe hanger installation are also being developed. With these specifications available, fewer changes to modifications due to field conditions will be necessary. The availability of engineering specifications will also facilitate quality control by providing a more precise set of criteria to inspect against. These additional specifications are being developed on a schedule to support the 1989 refueling outage.

Following the SSOMI, the operations of the Quality Assurance Department were reviewed. Areas needing improvement have been identified and actions initiated to strengthen the overall corrective action program at Indian Point 3.

We are continuing to enhance the knowledge level of the QA Department in the area of plant operations by enrolling quality assurance personnel in the licensed operator certification program. The Department currently includes an individual with a senior reactor operator's certification. The functions of the QA Department are being formalized to an even greater degree through the development of various Quality Assurance Instructions (QAIs) which address the inspection of maintenance work such as electrical terminations and environmentally qualified splicing. As installation specifications are developed, additional QAIs will be developed.

The QA Department is developing a procedure for trending deficiencies identified during their inspections. It is anticipated that this procedure will be in use by April 1, 1988. A station goal will be developed to address procedural adherence. This will further highlight to plant personnel the requirement to follow procedures. The goal will be designed to reduce the frequency of identified failures in procedural adherence during station maintenance and operations. A Material Review Board has been established to facilitate the resolution of material deficiencies and enhance the technical bases for the same. The Board includes representation from the station QA, Maintenance, and Engineering Departments.

The corrective action program is being enhanced through the upgrade of procedures directing responses to findings and corrective action requests. The procedures are being clarified and additional detail is being provided which will result in improved responses to deficiency reports. Once the procedures are complete, site personnel will be trained on the purpose of the program and the proper use of the procedures.

Improvements in the working relationship between QA and Maintenance Departments have occurred since the QC Supervisor now typically attends the daily Maintenance Department meeting. The QA Department has increased coverage of operations and surveillance testing activities. In the past, audits of the activities of these departments were performed to ensure conformance with the requirements of the plant's operating license. This will continue with the addition of field activity monitoring.

The Power Authority fully recognizes the significant concern about adherence to procedures. Careful review of all findings in the inspection report related to procedural adherence has resulted in the general conclusion that the instances of procedure violation were correctly taken in a "practical" sense and were not violations of quality. Nevertheless, they were inconsistent with overall governing directives and consequently were wrong. The procedural overhauls, development of standards and other corrective measures described above are intended to provide procedures and overall guidance that permits practical and reasonable latitude to field engineers and supervisors involved in installation and test activities, yet retain the control, documentation and assurance of quality which is essential.

Attachment 2

Safety Systems Outage Modification

Inspection 50-286/87-015

Installation and Test Phase

Response to Section 2

The Authority has reviewed the details provided in Section 2 of the Safety System Outage Modification Inspection Report. Specific sections were identified that require clarification or additional details to ensure the proper perspectives are conveyed by the report.

The format of the attached comments provides a restatement of the finding as documented in the inspection report followed by additional information which is meant to clarify the issue or concern.

Section 2.1.2.1: Station Batteries

NRC Concern

The four surveillance test procedures, 3PT-R29 A through D, provide detailed requirements for load testing of Station Batteries 31, 32, 33, and 34. These procedures also incorporate the battery maintenance requirements specified in IEEE Standard 450-1975, "Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." In reviewing Rev. 0 of these procedures, the NRC inspectors noted that the procedures did not contain acceptance criteria for the terminal and intercell connection detail resistance readings taken during test performance. The lack of acceptance criteria for intercell resistance readings that are established through trend evaluations would prohibit taking the required corrective action specified by the IEEE Standard.

In reviewing the associated historical data sheets for this activity, the NRC inspectors observed a number of intercell resistance readings which exceeded the corrective action values specified by the IEEE Standard. Increased resistance values were observed in each of the four station battery banks but were significantly high between cells 26 and 27 of Battery 34 where the recorded intercell resistance reading was 335 micro-ohms. This value is substantially above industry accepted operational levels and represents a potential source of battery fire.

The inspector also reviewed the then current revision (Rev. 1) of the station battery load test procedure and noted that a 90 micro-ohm value has been established as acceptance criteria for intercell resistance readings. However, this value does not meet the requirements of the IEEE Standard which prescribes a data trend evaluation, and corrective action for cells whose intercell resistance has increased 20 percent over previous readings.

These observations were discussed with licensee personnel who provided a handwritten draft which would change Surveillance Test Procedure 3PT-R29 to read as follows: "If terminal connection resistance increases by 20 percent from the previous test, a work request shall be initiated to clean and recheck connection resistance values prior to the next general inspection."

Response

The 125 volt DC power systems at Indian Point 3 are tested in accordance with the requirements of Technical Specifications. IEEE-450 is not referenced in the Technical Specifications and no formal commitment to the standard has been established.

The Authority recognizes the benefits such industry standards provide in developing test programs and procedures, however, the applicability of various standards depends on specific plant designs and features. When appropriate, salient provisions of industry standards are incorporated into plant procedures. For instance, battery intercell resistance measurements have been taken at Indian Point 3 in the course of battery surveillance testing. In the past, an upper limit on intercell resistance was applied and any changes were evaluated by technical staff and appropriate recommendations for corrective actions made.

The SSOMI report notes a concern with a high intercell resistance measurement in Battery No. 34 and expresses a concern for a potential battery fire. A visual observation of the Battery revealed a length of 350 MCM cable utilized as the intercell connector between cells 26 and 27. The nominal resistance rating of this type of cable is 367 micro-ohms and was determined to be the source of the anomalous intercell resistance measurement. The Authority concluded that no potential existed for a battery fire as stated in the report since the cable did not represent an ignition source.

Section 2.2.2.9: Accumulator Pressure Transmitter Replacement

NRC Concern

The completed engineering test procedure ENG-213, Rev. 0, "Accumulator Pressure Calibration," was reviewed as were the retest Work Requests (WRs) 9790B, Leak Testing of Transmitters, and 9790C, In-service Leak Testing of Transmitters. These documents recorded the functional testing and calibration of the accumulator tank pressure transmitters replaced under MOD 86-03-006 SIS.

Two deficiencies were identified with the Post-Maintenance/Modification Retest Forms associated with the WRs. Contrary to the requirements of PFM-5, "Retest Program," the Retest Form for WR 9790B had "results of the test" and "test performed by" blocks signed by a supervisor who did not perform or witness the test. The Retest Form for WR 9790C (tubing leak test) did not specifically identify the joints affected by the modification package and therefore did not identify the possible points of leakage to observe.

Response

The modification/maintenance program at Indian Point 3 provides for retesting systems and equipment utilizing a customized test for the particular situation or an existing surveillance procedure. In the latter case, the retest program instituted through Administrative Procedure 9 (AP-9) provides the programmatic controls which serve to document the performance of an existing procedure as fulfilling the retest requirements of a work request. Attachment VII of AP-9 is maintained on file to document the completion of the surveillance procedure. The example cited in the report where a signature line on attachment VII of AP-9 was signed by a supervisor not directly involved in the test is not relevant in the context of the programs at Indian Point 3. In this example, as described above, the retest involved the performance of a surveillance procedure and the retest form (attachment VII of AP-9) merely served to document the fact that the surveillance test fulfilled the retest requirements of the modification. The surveillance procedure was signed by the personnel involved in the testing.

The report also identified an apparent deficiency in the retest for Work Request 9790C in that specific tubing joints affected by the maintenance were not identified as potential leakage points to observe.

The Authority does not agree with this finding since, as discussed with the SSOMI team, the retest specified an inservice leak test for the tubing associated with the entire level sensing system from tanks to transmitters. As such, all joints in the system were inspected for leakage, not merely the joints opened for the modification. The retest encompassed a larger scope since the Authority considered it prudent to verify the integrity of the entire level sensing system to verify no inadvertent damage occurred to those portions of the system not specifically covered under the modification.

Section 2.4.2.1.1: Inservice Testing

NRC Concern

The stroke time test summary computer data base from November 1985 was reviewed for the selected containment isolation valves. These valves required quarterly stroke time testing to verify operational readiness. The team noted that valves WDS-1788 and 1789 apparently had not been tested between November 6, 1986, and April 30, 1987. The January 31, 1987, test record for these valves did not list any test data due to a broken selector switch. The work request that replaced the defective selector switch apparently did not require a retest, contrary to the requirements of the ASME Code and IP-3 procedure PFM-22.

Response

In accordance with the quarterly stroke testing requirements of the ASME Boiler and Pressure Vessel Code, Valves WDS-1788 and WDS-1789 were scheduled to be stroke tested on January 31, 1987. These valves are normally closed, containment isolation valves which are opened to draw gas samples from the Reactor Coolant Drains Tank.

At the time of the January, 1987 stroke test, the valves failed to operate. A work request was issued to investigate and correct the inoperable valves. This work request was written to the Instrument and Controls Department to investigate what was thought to be a deficiency in the valve control switch. The investigation concluded the control switch operated satisfactorily and, since no actual work was performed on the switch, no retest was required or performed. Since the initial problem with the valves had not been identified or corrected, a work request was written to Maintenance to repair the solenoids associated with the valves. Prior to the action being taken on the maintenance work request, the next quarterly test came due. The valves operated and the test was completed satisfactorily. The maintenance work request was completed during the refueling outage (May, 1987 - September, 1987) and the valves were retested accordingly. Contrary to the statement in the Inspection Report, no work was performed on the valves during the period January 31, 1987 and April 30, 1987 which necessitated a retest.

Section 2.4.2.2: Post Modification Valve Testing

NRC Concern

The stroke timing and functional test of containment isolation valve PCV-1191, after replacement of its actuating solenoid operated valve, were observed. Although the test results were acceptable, several discrepancies in test performance were noted. The procedure required local observations and timing of the valve stroke. However, although an operator was at the valve, stroke was not timed at the valve, nor was a close observation of valve operation, such as solenoid noise monitoring, performed. In addition, procedure 3PT-Q28 did not provide instructions on how to time the valve locally.

Response

The inspection report documents an apparent failure to locally time the stroke of valve PCV-1191 and states that the test procedure required local valve stroke timing.

Procedure 3PI-Q28 requires local observation of valve stroking only. Pursuant to ASME Boiler and Pressure Vessel Code Section XI IWV-3300, "Valves with remote position indicators shall be observed at least once every 2 years to verify that valve operation is accurately indicated." All valve timing at Indian Point 3 is done 'switch to light'. This testing philosophy ensures consistency from test to test. Valve stroke timing in this manner is also consistent with the principles of ALARA. Since many of the valves required to be timed are located in radiation areas or are inaccessible during normal operations, 'switch to light' timing considerably reduces the exposure to plant personnel.

Section 2.4.2.3.1: Post Modification Leak Checking

NRC Concern

Modification 85-03-140 AFW replaced the solenoid operated valves that actuate AFW Pump Pressure Control Valve (PCV)-1139. The modification replaced two valves (EQ requirements), terminal boxes, conduits, cable, terminal blocks and approximately eight feet of copper tubing. Several test-related deficiencies were noted:

- o The modification procedure did not specify that new or remade tubing mechanical joints be leak checked and no test procedures or records existed to show that they were checked.

- o Section"H" of the modification procedure required that cable insulation tests and continuity checks be performed on new cables. These tests were not performed during installation or retesting.

Failure to specify and perform piping system leak tests is inconsistent with the requirements of ANSI B31.1, AP-3, AP-12 and PFM-12. Failure to perform the insulation resistance tests and continuity checks on replacement cables was inconsistent with the requirements of PFM-5 and AP-3.

Response

Valve PCV-1139 had been retested using three different test procedures. These tests included a simple stroke test and two functional tests at rated conditions prior to plant start-up. The valve functioned as required without any noted deficiencies on these tests. This confirmed the continuity of the electrical circuits.

Insulation resistance measurements were specified in the modification procedure. Such tests are normally performed during installation before cable termination is completed and is not typically performed during modification acceptance testing or retesting. The Authority acknowledges the failure to perform this test, however it is not within the cognizance of the retest group to perform insulation resistance checks. To perform the test at that time would have involved taking the modification apart.

As discussed in Attachment 1, the Authority is taking steps to ensure procedural adherence at Indian Point 3. These steps include issuance of management directives, station goals which elevate personnel cognizance to the issue, and improvements in the design change control process.

Section 2.4.2.3.3: Post Modification Leak Checks

NRC Concern

Modification 85-03-094 MULT replaced 16 steam generator blowdown and sampling system and two demineralized water (DM) supply isolation solenoid operated valves. The modification procedure did not specify air line joint leakage tests. Retest procedures 3PT-Q35 (for the DW valves) and ENG-190 (for the blowdown valves) did not test for leakage. Failure to specify and test for piping system leaks was inconsistent with the requirements of ANSI B31.1, AP-3, AP-12, and PFM-12.

Response

The subject modification replaced a number of solenoid operated valves. The joints of the air supply lines to the valves were effectively leak tested through functional tests.

Stroke testing of all the valves was performed including local observation. No discrepancies in these tests were noted. Two valves were checked specifically for proper air pressure to verify proper operation of an additional pressure switch incorporated in their control circuitry.

It is the Authority's position that the functional capabilities of the valves were tested after modification and that all applicable code and procedural requirements were adhered to.

Section 2.4.2.1.1: Inservice Testing

Section 2.6.2: Quality Assurance

NRC Concern

DCARs were not trended by the QA/QC Department. In addition, as many nonconforming conditions at IP-3 were identified and resolved via Work Requests (by the Maintenance Department) as by engineering group (Maintenance vs. Technical Services) and did not receive any QA/QC review or concurrence.

As a further example, DCARs 87-021 and 021A identified debris clogging instrumentation piping during modification to piping penetration assemblies. The response was that prior to welding and prior to closing of the penetration, the lines were blown clear. On 87-021A, the repair block was checked. The "repair/rework inspected by" blocks were signed off by QC three days (87-021) and six days (87-021A) after disposition, apparently long after access for inspection had been.

Response

The staff notes in the report that non-conforming conditions at Indian Point 3 are resolved, in many instances, by work requests which do not receive any QA/QC review or concurrence.

Criterion XV and XVI of 10CFR50, Appendix B, are applicable to anyone performing safety related functions at a nuclear power facility. The Authority encourages its employees to identify and promptly resolve observed deficiencies in plant material conditions. Resolution of such deficiencies has never been the exclusive responsibility of the Quality Assurance organization and Quality Assurance review of resolutions is not required.

The work control system instituted through AP-9 is audited by Quality Assurance to ensure conformance to all requirements of the program.

The report identifies two DCARs which were signed by the Quality Control inspector several days after access for inspection was available. The Authority reviewed this matter and has concluded that appropriate actions were completed in the field to resolve DCARs 87-021 and 87-021A in a timely manner. The implication noted in the inspection report resulted from an administrative delay in documenting the paperwork associated with DCARs 87-021 and 021A.

Attachment 3

Actions Taken in Response

to

Startup Concerns Identified

in the

Safety System Outage Modification Inspection

Installation/Test Phase

In the course of the Safety System Outage Modification Inspection, several issues were identified which the Authority agreed to resolve prior to startup from the Cycle 5/6 refueling outage. The resolution of the issues was completed prior to startup as documented in W. A. Josiger's letter to W. F. Kane of Region I, dated August 21, 1987.

A summary of the actions taken to resolve each of the issues prior to startup follows.

1. The Authority performed an alignment and flow balance test on the Service Water System essential header. This test, ENG-281, was conducted on August 11 and 12, 1987. During the test, various valves were throttled to provide the proper flow distribution throughout the system including the safety related loads on the essential header.

2. Data taken during the performance of ENG-281 has been provided to our consultant and utilized to benchmark and revise the analytical model of the Service Water System.

3. The Authority has completed modifications to the Diesel Generator Service Water outlet flow control valves. These modification eliminated all automatic actuation features for these valves and established a fixed Service Water flow rate through the Diesel Generator jacket water and lube oil coolers which conforms with the cooling demands of the diesels during emergency operations. The positions of these valves were established and set during the conduct of the Service Water Flow Test, ENG-281.

4. The Service Water System flow test, ENG-281, also demonstrated the response of the Service Water System essential header to a simulated Loss-of-Coolant Accident and its attendant Service Water flow demands. This portion of the test included the effects of a loss of instrument air on the flow control systems of all components supplied by the essential header. This portion of the test confirmed that the Containment Fan Cooler Units Service Water flow requirements are achieved.

5. The Authority has revised the appropriate Emergency Operating Procedures to ensure proper operation of the Service Water System non-essential header during the transition to the recirculation phase following a loss-of-coolant accident. These procedure changes are designed to minimize the potential for pump run out on the non-essential header during the process of realigning the Service Water System for long term post LOCA cooling.

6. Alarm Response Procedure ARP-5 has been revised to reflect the appropriate Service Water flow requirements for the Fan Cooler Units. Operating procedure SOP-RW-6, Fan Cooler Unit Flow, has been eliminated. The provisions of this procedure, including future manipulations of the Fan Cooler Unit Service Water outlet valves, will be performed in accordance with a performance test which will ensure the proper Service Water flow balance between the Fan Cooler Units.

7. The 480v power cable terminations for seven safety related motors have been rebuilt. The splices for the five (5) Fan Cooler Unit motors and

two (2) Residual Heat Removal Pump Motors have been made up using procedures and material which provide for a qualified splice. The splices for the two (2) Recirculation Pump motors were also rebuilt prior to exceeding cold shutdown. The appropriate qualification documentation for these splice designs is on file.

8. All standard Vulkene SIS cable subject to a harsh environment had been previously replaced with appropriately qualified cable or qualified for the environment in which it must operate. With respect to this concern, a field walkdown identified only one installation utilizing standard Vulkene cable involving four short pieces of wire. The harsh environment which this cable could be subject to is not LOCA induced. The cable could be exposed to elevated temperature and humidity for a short duration. The qualification of the cable for this particular installation has been established.

9. The environmental qualification for low voltage splices and RTV-7403 field installations has been established and is on file at the plant. Records which document the construction of low voltage splices and link the filed splice to appropriate qualification documents have been reorganized and enhanced. The Authority has confirmed the acceptability of the RTV-7403 installations at Indian Point 3 through tests and has identified independent qualification documentation supporting installations at the plant. The environmental qualification of the No. 32 Residual Heat Removal pump motor, which was installed during the current outage, has been established. The documentation which includes the results of a material history search for the motor

is on file at the plant.

10. A field walkdown of the equipment on the Environmental Qualification Master List has been completed. Discrepancies in equipment installation and condition were identified, documented, and classified. Those discrepancies which could potentially impact the qualification for the environmentally qualified equipment have been corrected. Details of the discrepancies identified during the field walkdown and the associated corrective actions are documented and available for review by the staff.