U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/89-21

Docket No. 50-423

License No. NPF-49

Licensee: Northeast Nuclear Energy Company
P.O. Box 270
Hartford, Connecticut 06141-0270

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection At: Waterford, Connecticut

Inspection Conducted: October 16 - November 27, 1989

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Approved By: for Donald R. Haverkamp, Chief, Reactor Projects Section 4A, Division of Reactor Projects — 1/3/90 Date

Inspection Summary: Inspection on October 16 - November 27, 1989 (Inspection Report No. 50-423/89-21)

Areas Inspected: Routine safety inspection by resident and regional inspectors and headquarters personnel of plant operations, maintenance and surveillance, engineering and technical support, safety assessment and quality verification, and radiological controls.

Results: Four licensee-identified, non-cited violations were reported during this period. One of these violations concerns the inoperability of fire protection equipment (Section 4.4) and is indicative of a lack of an apparent sensitivity to fire protection. The licensee needs to improve personnel

performance in this area. Two unresolved items were also identified; one concerns the untimely followup of licensee commitments (Section 4.7). The other item concerns continuing NRC review of the licensee's actions regarding the postulated scenario in LER 88-26 (Section 5.2). Final NRC review of the licensee results from the containment integrated leak rate test conducted in July 1989 has been completed with no weaknesses identified (Section 5.1). Additionally, during this report period, good maintenance/surveillance practices were noted by the inspector.

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* The NRC Inspection Manual inspection procedure or temporary instruction that was used as inspection guidance is listed for each applicable report section.

DETAILS

1.0 Persons Contacted (IP 30703)

Inspection findings were discussed periodically with the supervisory and management personnel identified below:

*C. Clement, Unit 3 Superintendent
*M. Gentry, Operations Supervisor
R. Rothgeb, Maintenance Supervisor
*J. Harris, Engineering Supervisor
D. McDaniel, Reactor Engineer
R. Satchatello, Health Physics Supervisor
*B. Enoch, Supervisor, Instrumentation and Controls
*S. Scace, Station Superintendent

*Denotes those attending the exit meeting

2.0 Summary of Facility Activities

The Millstone Nuclear Power Station, Unit 3 (Millstone 3 or the plant) operated at 100% of rated thermal power (full power) during the report period. Power reductions were required on October 19 to perform condenser backwashing operations and on November 16 as a result of storm loading on the intake structure. On November 11, the turbine was taken off the grid and shut down due to a failed electric overspeed trip device. The electrical trip solenoid and valve were replaced and the turbine was synchronized to the grid later that day. Full reactor power was achieved on November 13. During this report period, increased steam generator hand hole leakage was observed on the "C" steam generator. Hand hole leakage is measured as a function of the containment sump pumping rate which has increased from 1.1 gallons per minute (gpm) to 5.1 gpm. Licensee monitoring of the leakage was adequate while preparations were in progress for a potential shutdown to repair the leak. The leak was repaired while the plant was in cold shutdown after the end of this report period.

The inspection activities during this period included 85 hours of inspection during normal utility working hours. In addition, the review of plant operations was routinely conducted during portions of backshifts (evening shifts) and deep backshifts (weekend and midnight shifts). Inspection coverage was provided for 32 hours during backshifts and 13 hours during deep backshifts.

3.0 Plant Operations (IP 71707/71710/93702)

3.1 Control Room Observations

The inspector reviewed plant operations from the control room and reviewed the operational status of plant safety systems to verify safe operation of the plant in accordance with the requirements of technical specifications and plant operation procedures. Actions taken to meet technical specification requirements when equipment was inoperable were reviewed to verify the limiting conditions for operations were met. Plant logs and control room indicators were reviewed to identify changes in plant operational status since the last review and to verify that changes in the status of plant equipment was properly communicated in the logs and records. Control room instruments were observed for correlation between channels, proper functioning and conformance with technical specifications. Alarm conditions in effect were reviewed with control room operators to verify proper response to off-normal conditions and to verify operators were knowledgeable of plant status. Operators were found to be cognizant of control room indications and plant status during normal working hours and backshift observations. Control room manning and shift staffing were reviewed and compared to technical specification requirements. No inadequacies were identified.

3.2 Turbine Shutdown

On November 3, 1989, while testing the turbine electric overspeed device per surveillance procedure SP 3623.1 Turbine Generator Testing, an unsatisfactory trip time was obtained. The overspeed trip was then successfully retested three times with satisfactory results, i.e. trip time was less than five seconds. The cause of the first failure was attributed to "sticking" components. Accordingly, the licensee increased the testing frequency of the trip circuitry from once a week to once per shift. Performance of the component was at first satisfactory however, on November 9, test failures began to reappear. The overspeed device failed to actuate within the required time period and was then followed by two successful tests. Testing was then increased to once every four hours. On November 10, after two successive test failures, the electric overspeed trip was declared inoperable. Technical Specification (TS) 3.3.4 requires the turbine overspeed trips to be tested per the Millstone 3 Turbine Overspeed Protection Maintenance and Testing Program. If the testing determines that an overspeed trip device is inoperable, TS 3.3.4 requires the turbine to be isolated from the steam supply in six hours.

The inspector was informed of the failed surveillance at 7:30 p.m. on November 10. In responding to the notification, the inspector arrived on site to observe plant operations. Good coordination was observed between the balance of plant and control room operator during the downpower maneuver. Plant shutdown procedures were being used and technical specifications were followed.

When reactor power was reduced to 20%, the plant turbine was tripped and reactor power was controlled by use of the steam dumps. The suspected defective components which consist of an electrical solenoid and actuating valve were replaced and subsequently retested with satisfactory results obtained. Approximately seven hours after the turbine was tripped, it was placed back on line and synchronized to the grid. Full reactor power was reached on November 13.

Plant engineering could not determine the root cause of the failure, although dirt buildup on the actuating valve, or distortion of solenoid components as a result of excessive heat, has been considered as possible causes. The assembly has been subsequently shipped to the vendor for testing and refurbishment. Overall licensee response to the failure was good and the inspector had no further questions on this matter.

3.3 Review of Plant Incident Reports

The plant incident reports (PIRs) listed below were reviewed during the inspection period to (i) determine the significance of the events, (ii) review the licensee's correlation of the events; (iii) verify the licensee's response and corrective actions were proper; and (iv) verify the licensee reported the events in accordance with applicable requirements, if required. The PIRs reviewed were: numbers 3-89-164, 3-89-172, 3-89-173, 3-89-175, 3-89-179, 3-89-180, 3-89-181, 3-89-182, 3-89-183, 3-89-184, 3-89-185, 3-89-186, 3-89-187, 3-89-188, 3-89-189, 3-89-190, 3-89-191, 3-89-192, 3-89-193, 3-89,194, 3-89-195, 3-89-196, 3-89-197, 3-89-198, 3-89-199, 3-89-200.

No inadequacies were identified. The following PIRs were selected for additional followup: 3-89-172, 3-89-173, 3-89-179, and 3-89-181 and are discussed further in detail sections 4.6, 4.5, 4.4 and 4.3 of this report, respectively.

4.0 Maintenance/Surveillance (IP 62703/61726/92702)

4.1 Observation of Maintenance Activities

The inspector observed various maintenance and problem investigation activities for compliance with procedures, plant technical specifications, and applicable codes and standards. The inspector also verified the appropriate quality services department (QSD) involvement, safety tags, equipment alignment and use of jumpers, radiological and fire prevention controls, personnel qualifications, post maintenance testing, and reportability. Portions of the following maintenance activities were reviewed on November 14, 1989.

-- "A" emergency diesel generator preventive maintenance
-- "B" reactor plant component cooling water pump removal
-- troubleshooting activities on service water valve AOV-24A
-- resetting of limit switch on service water valve SWP-AOV39B

No inadequacies were identified.

4.2 Observation of Surveillance Activities

The inspector witnessed selected surveillance tests to determine whether properly approved procedures were in use; plant technical specification frequency and action statement requirements were satisfied; necessary equipment tagging was performed; test instrumentation was in calibration and properly used; testing was performed by qualified personnel; test results satisfied acceptance criteria; and, unacceptable results were properly dispositioned. Portions of the following activities were reviewed.

-- "B" motor control center/rod control pump test, 3626.10
-- high range rad monitor operations test, 3449H31-1
-- "B" diesel generator operational test, 3646A2-1
-- service water 'B' train quarterly valve test, 3626.3

No inadequacies were noted.

4.3 MSIV Degradation Reported - LER 89-24

Reference: Plant incident report 3-89-181

While conducting partial stroke testing of the main steam isolation valves per plant test procedure SP 3616A.1, Main Steam Valve Operability Test, the 2A closing solenoid on the "C" MSIV did not operate within the required five-second time period. The solenoid was in this degraded condition for approximately 20 hours until it operated satisfactorily when it was retested the following day on October 11. The MSIVs are designed so that either set of solenoids (1A, 2A or 1B, 2B) must actuate within five seconds of receiving a close signal, in order for the MSIV to shut within five seconds as required by TS 3.7.1.5.

The licensee did not enter into a TS action statement based on its initial assumption that the valve was still operable, since the 1B and 2B solenoids acted satisfactory when tested and the plant would be in an analyzed condition in the event that the 1B and 2B solenoids failed to work and the MSIV did not close. Subsequent licensee review of the accident analyses revealed that the plant was not in an analyzed condition when the 2A solenoid was degraded. The licensee also concluded that the action statement of TS 3.7.1.5, which requires MSIVs to be operable, should have been entered. The sequence of solenoid operations that enables an MSIV to close and the issues that developed during the MSIV testing are described in further detail in inspection report 50-423/89-16.

Inspector review of the licensee's accident analysis for a faulted steam generator, as stated in chapter 15 of the Final Safety Analysis Report (FSAR), noted that the analysis description could be misleading. The FSAR stated that the plant is analyzed for a faulted generator condition coincident with a stuck open MSIV. However, through discussions with the nuclear steam supply system vendor (Westinghouse) the licensee learned that the worst case steam line break analysis assumes only a single failure of one safety injection train. In the analysis, all four MSIVs are assumed to close. This scenario limits the boron injection flow rate from the charging pumps which is used to prevent return to criticality in the core. Consequently, when the 2A solenoid was in a degraded condition, if a faulted generator event occurred and the single active failure that is assumed was the loss of the "B" emergency safety features train, the "C" MSIV would not close within the Technical Specification 3.7.1.5 time limit of five seconds. This event would then place the plant in an unanalyzed condition. It is important to note that the end point for the analyzed and unanalyzed condition would essentially be the same - one faulted generator would blow down. The only difference is that in the second event, one safety train would be inoperable.

Through discussions with personnel in the reactor engineering department, the inspector was informed that the licensee intends to have the unanalyzed condition examined by Westinghouse. If the analyses results show that no decrease in the margin of safety results from the condition, a change to TS 3.7.1.5 will be submitted to Nuclear Reactor Regulation (NRR) to allow reactor operation for a greater length of time than is currently allowed by the existing technical specifications when a closing solenoid is in a degraded condition. This change request is expected prior to the commencement of the fourth refueling cycle in December 1990.

The inspector agrees that since the plant was in an unanalyzed condition, a TS action statement should have been entered. However, the fact that surveillance testing revealed a degraded solenoid should have been cause to enter a TS action statement. The inspector discussed this position with the unit superintendent who stated that if future testing revealed the closing solenoids were in a degraded condition, the Technical Specification 3.7.1.5 action statement would be entered. Also, the proposed technical specification change discussed above (and in inspection report 89-16) may not be pursued, due to plant management's perceived lack of additional gain in limiting condition for operation action statement flexibility. The superintendent also stated that as part of the licensee review of this matter, a change to the FSAR will be considered to further clarify what assumptions are made in the steam line break accident analyses. The discovery that the plant was in an unanalyzed condition was reported in LER 89-24. No violation will be issued for this discovery, per the policy in 10 CFR 2, Appendix C, since the licensee-identified item had minor safety significance, the final plant end state is essentially identical to the analyzed condition, the item was reported, and corrective actions were appropriate to prevent recurrence (423/89-21-01).

4.4 Inoperable Fire Equipment

Reference: Plant incident report 3-89-179

This event was discovered by the licensee on October 6, 1989 and reported in LER 89-23 while the plant was operating at 100% of rated power. Plant TS 3.7.12.6.a requires eleven hose houses to be operable by having sufficient lengths of hydrostatically tested hoses available for use in each designated storage location. If a house contains insufficient lengths of hose, as compensatory measures adjacent houses are to be supplied with additional lengths of 2 1/2" hose to cover the unprotected area within one hour.

On October 2, 1989, operations department personnel removed several lengths of hose from three houses to support work in the intake structure area. The operations department personnel who performed this action did not consider that removing the hose from the station rendered the system inoperable; therefore, the shift supervisor on duty was not notified of the hose removal.

The house inoperability was not discovered until October 6 when operations personnel who were flushing the fire main system discovered the missing hose. Further examination of the eight other houses revealed that each house was missing one length of 1 1/2" hose, which renders the houses inoperable.

Per plant TS a monthly audit of the house hoses is conducted to verify that sufficient equipment is maintained in each house. This surveillance is conducted by the building services department per procedure 1502/21502/31502, Hose Station Surveillance. This surveillance was performed on October 5, 1989 and correctly documented the fact that three houses contained insufficient lengths of hose. However, the procedure did not require the operations department to be notified of any inadequacy, therefore, no notification was performed until a day later.

The missing hose in the eight houses occurred because the individual performing the surveillance referred to an out-of-date equipment listing marked on a placard in each hose house rather than the current surveillance procedure. Additionally, the individual who performed the surveillance did only a visual check of the components in the house. Each individual component was not counted. The individual acknowledges that a visual check was an inadequate method of determining the quantity of equipment in the house. The individual also stated that the visual method of taking inventory was the manner in which he was instructed to perform the surveillance.

Immediate corrective action was to replace the missing hoses on October 6. To prevent recurrence of the event, the licensee removed the out-of-date placards from the hose houses, revised the surveillance procedure to require the immediate notification of the operations department if an inadequate equipment inventory is discovered and placed signs in the hose house stating the following: "Fire Brigade Use Only, Notify Shift Supervisor When Removing Equipment from this TS Hose House." Finally, changes to the procedure must be reviewed by a Unit 3 senior reactor operator prior to implementation.

Review of this event by the inspector revealed that it would not have happened if personnel were more sensitive to fire protection equipment. Fire protection equipment should be dedicated for one use and personnel should be informed of this fact. Additionally, the individual who performed the surveillance did not adhere to the procedure and apparently was not properly trained in the proper method of surveillance performance. The inspector discussed the training concern with a station services engineer who is responsible for fire protection equipment. The engineer acknowledged that the individual was improperly trained. To correct the weakness, the inspector was informed that the individuals who will perform the surveillances would be given training in the beginning of 1990 in fire protection equipment and their use and storage. As an interim measure, the station services engineer will accompany the individuals during the performance of the monthly surveillances to ensure quality. The inspectors consider the corrective actions are appropriate. No violation will be issued per the policy in 10 CFR 2, Appendix C, because the licensee-identified item had minor safety significance, the item was reported as required, and corrective actions were appropriate to prevent recurrence (423/89-21-02).

4.5 Improper Valve Stroke Surveillance

Reference: Plant incident report 3-89-173

This event was discovered by the licensee on September 25, 1989 while the plant was at 100% of rated power and is documented by LER 89-22. Safety injection pump suction isolation valves SIH-MV8923A & B are required by the inservice test program to be tested and timed in the open-to-close direction. However the valve stroking surveillance procedure SP3608.6 required the valves to be tested in the close-to-open position. The failure to test the valves in accordance with the Inservice Test program is a violation of TS 4.05 which requires ASME Section XI to be implemented as identified by 10 CFR 50.55 (A)(9).

The cause of this event was procedural error. The procedure did not specify the correct valve stroking direction as required by the inservice test (IST) manual. The licensee had adequate assurance that the valve would operate in the desired direction, since it was routinely repositioned to perform the surveillances, but it was never timed in the required direction. Upon discovery of the event, the valve was tested satisfactorily in the close direction.

The inspector reviewed the IST manual and noted that an incorrect position could have been transcribed. Through conversations with plant engineering, the inspector was informed that other IST procedures were reviewed and no discrepancies were found. Therefore, this appears to be an isolated occurrence.

No violation will be issued for this event per the policy in 10 CFR 2 Appendix C, since the licensee-identified item had minor safety significance, the item was reported as required and corrective actions were appropriate to prevent recurrence (423/89-21-03).

4.6 Miscalculations of Component Response Times

Reference: Plant incident report 3-89-172

This deficiency was discovered by the licensee on September 25 and reported in LER 89-21 with the plant operating at 100% of rated power. Plant TS requires engineered safety feature (ESF) response times to be calculated as the time interval when the monitored parameter exceeds its ESF actuation setpoint to the time the actuated equipment performs its safety function. The Millstone Station procedure for calculating response time was inadequate in that it failed to account for the time interval between master relay and slave relay actuation. When calculating ESF response times, the time period from sensor input to master relay is measured. The time is then added to an operation department surveillance result which measures the time from initiation of a change in state signal from the main board switch, until the component is in the closed state. Therefore a true response time was not obtained.

Operability of the master and slave relays is verified through quarterly operations department surveillances. To develop a valid test, the times obtained from the most recent slave relay testing were added to the original ESF operational data. No TS response time requirements were exceeded. For equipment whose logic sequence does not cause component actuation, a specific test was written to measure the response times of the affected relays. The times obtained were verified to be within the TS range. Therefore, no impact on plant safety resulted from this procedural inadequacy.

This procedural weakness was discovered earlier by the licensee during a procedural review program as corrective action for LER 87-42. However, the review did not evaluate the item as reportable at that time. The item was subsequently classified as an issue that should be corrected during routine procedural upgrade. This was not accomplished due to miscommunication between the instrument and control (I&C) and operations departments. Specifically, the I&C shop personnel thought slave relay timing was already included as part of the operations department surveillance. Consequently, the weakness was never corrected during the procedure review process. To prevent recurrence of the aforementioned problem, when procedural comments are not incorporated the responsible department must document the reason for not incorporating the comment and notify the individual who identified the potential problem. Through conversations with the reactor engineer, the inspector was informed that all outstanding comments to this procedure have been examined and none are reportable.

The failure to adequately measure ESF response times is a violation. No violation will be issued per the policy of 10 CFR 2 Appendix C, as the licensee-identified item had minor safety significance, the item was properly reported as required and corrective actions were appropriate to prevent recurrence. (423/89-21-04)

4.7 Followup to May 11 Reactor Scram

On May 11, 1989 a reactor scram occurred that was caused during connection of the rod drop monitoring computer in preparation for scheduled testing. This event was described in LER 89-09 and was reviewed further in inspection report 89-04. Inspector review of the LER issued June 12, 1989 noted the report accurately described the event, provided an accurate description of the equipment problems and repairs, and contained a good root cause determination for the scram.

The event was found to be procedure inadequacy in that surveillance procedure SP 3451N21 did not specify that the control rods must be unlatched prior to connecting the computer to the control rod drive system. Even though this procedure had been performed under similar conditions in the past without causing a scram, the licensee learned from discussions with the vendor that Millstone 3 operating and technical manuals assumed the rods would be unlatched prior to connecting the computer. The vendor stated spurious signals could be generated within the rod logic system, which in turn could generate a drop signal, when the rod drop test cart is turned off. Licensee planned corrective actions were to change the surveillance procedure to require that the control rod drive mechanisms be unlatched prior to connecting the computer to the CRD system. Except as noted below, no inadequacies were identified with the licensee's followup to the scram.

The LER issued June 12, 1989 reported the surveillance procedure had been revised as indicated above. Inspector review on November 8, 1989 noted that this had not been accomplished. Procedure SP 3451N21, Rod Drop Time Test, Revision 2 dated May 9, 1989, inclusive of Change 2 dated May 12, 1989, contained no provisions to ensure the CRDMs were unlatched prior to connecting the computer. The procedure had not been updated since the test was last performed at the start of the refueling outage.

The inspector discussed this matter on November 13 with the I&C supervisor responsible for procedure development. The individual stated the changes were scheduled to be performed to support further rod drop timing testing prior to startup from the outage. However, the change was deferred and re-scheduled to occur prior to the next performance of the test, which would occur after startup from the refueling outage. The licensee stated the procedure changes were scheduled to be reviewed by the PORC on November 17. The inspector reviewed SP 3421N21 Revision 2 Change 4 dated November 17 and found the aforementioned precaution was incorporated in the procedure. The procedure revision was completed prior to December 1, 1989, which was in conformance with the target completion date established by the PORC review of plant incident report 3-89-66.

The incorrect information in the LER 89-09 apparently was the result of insufficient verification of completed action by the persons assigned to prepare the LER and by the applicable review organizations. The inspector noted that previous resident reviews and inspections by region-based personnel (reference inspection 50-423/88-22) have found LER corrective actions to be properly implemented as stated in the reports. However, another recent inspection (reference inspection 50-423/89-14) noted a finding in which procedure upgrades targeted as part of the corrective actions addressed in a plant incident report would not have been completed as required, if NRC inspection had not identified the discrepancy to licensee personnel.

During a subsequent meeting with the unit superintendent, the inspector expressed his concerns for the need for LER information to be accurate as well as timely. The unit superintendent stated that the commitment tracking system is currently being updated to produce a list of all outstanding commitments that are within seven days of the stated completion dates. This list would then be reviewed weekly at the department meetings to verify commitments are being met. Section XVI of the licensee's QA Topical Report, Revision 12, dated July 12, 1989 requires measures be established to correct conditions adverse to quality. The effectiveness of the licensee actions will be reviewed further during routine resident inspections to verify the licensee's program to assure conditions adverse to quality are corrected per the license commitments. This is an unresolved item pending licensee submittal of a corrected LER 89-09 and an explanation of the reason for the incorrect information. (423/89-21-05).

4.8 Plant Housekeeping

The inspector toured the following areas at the Millstone Unit 3 to assess housekeeping and general plant conditions:

-- emergency diesel generator rooms
-- auxiliary building
-- engineered safety feature building
-- main steam valve building
-- intake structure

Good housekeeping practices were observed in all locations and equipment appeared to be well maintained. Minor discrepancies noted included leaking lube oil fittings on the charging and auxiliary feed water pumps and a leaking fuel oil return line on the "A" emergency diesel generator. The licensee is aware of the leaks on the charging and auxiliary feedwater pumps and is evaluating a design change to eliminate the excessive amount of pipe fittings. The leak on the diesel generator fuel oil line was subsequently repaired. No other deficiencies were noted.

5.0 Engineering/Technical Support (IP 37700/37828/92702)

5.1 Containment Leak Rate Test Results Accepted

The inspector reviewed the licensee's July 1989 CILRT results documented in accordance with 10CFR 50, Appendix J, Paragraph V.B. These results were summarized in a technical document entitled "Reactor Containment Building Integrated Leak Rate Test" and were attached to the licensee's letter dated October 25, 1989 to the NRC. The report contains a test summary and general test description, presentation of test results, and other data such as descriptions of plant and computer software, and data analysis techniques.

The total time calculation method of Bechtel Nuclear Topical Report BN-TOP-1 for reduced duration tests was utilized. This method is acceptable per 10CFR 50, Appendix J requirements which stipulate that all Type A tests be conducted in accordance with the provisions of the American National Standard N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors.

The purpose of the test was to demonstrate that leakages through the primary containment building and systems penetrating containment do not exceed that allowed by plant technical specifications. The test was conducted with containment isolation valves and containment pressure boundaries in an "as-left" condition. The containment also met the leakage criterion in the "as-found" condition. The test was witnessed by two NRC region-based inspectors and was followed by a successful verification test, and these inspection findings were documented in inspection report 50-423/89-11. The CILRT results are presented below:

A. Type A Test Parameters

1.	Test Method	Absolute
2.	Calculational Method	Total Time (per BN-TOP-1)
3.	Test Duration:	
	Stabilization Period	4 hours
	Data Gathering for Leakage Calculation	8 hours
	Verification Leak Rate Test	4 hours
4.	Test Pressure	39.4 psig (full pressure test)

B. Test Results

		wt %/day
1.	Maximum Allowable Leakage Rate	0.9
2.	Acceptance, 75 Percent La.	0.675
3.	Measured Leak Rate, Lam, In "As-Found" Condition	0.2937
4.	Measured Leak Rate, Lam, In "As-Left" Condition	0.2919
5.	Conclusion	Acceptable

The inspector performed independent calculations of containment leakage rates and concluded that the containment has met its acceptance criteria for leakages in both the "as-left" and the "as-found" conditions.

5.2 NRC Followup of Bus Transfer Scenario LER 88-26, Rev. 3

On November 18, 1988 at 4:30 p.m., with the plant in Mode 1 at 100% of rated power, the licensee reported a postulated scenario which could in the extreme case result in a loss of redundant trains of safety related (vital) equipment. It was discovered that certain circumstances could lead to Millstone Unit 3 becoming isolated from the Millstone Station switchyard while on line. This could lead to an out-of-phase fast transfer to the reserve station service transformer (RSST) resulting in a potentially damaging transient on both trains of vital 4160 V busses.

The above scenario together with temporary remedial actions were described in LER 88-26. Revision 3 to LER 88-26 dated October 10, 1989, describes the final design modification which eliminated the fast transfer to the RSST on undervoltage signal.

The design modification was inspected during the second refueling outage for technical adequacy and 10 CFR 50.59 safety assessment and was determined to be satisfactory. During this report period, additional technical review of the modification by headquarters personnel was conducted with no discrepancies noted. While conducting the technical review, the licensee indicated that they would investigate the use of syncheck relays in their fast bus transfer scheme. These relays would prevent an out-of-phase fast bus transfer from occurring which could potentially damage safety-related equipment. In this regard, the inspector agreed to supply the licensee with the names of several other licensees who employ syncheck relays in their fast bus transfer design.

On November 27, the licensee informed the inspector that an additional problem has been identified with the fast transfer scheme. Specifically, the licensee concluded that when a fast transfer occurs, safety related motors could be exposed to a peak voltage of 1.85 V/Hz, which is greater than the design of 1.3 V/Hz. The licensee concluded that if a fast transfer occurs, no damage to safety related equipment would result on the first transfer; however, subsequent transfers should be prevented through use of a jumper assembly. The licensee has fabricated the jumper assemblies, and an engineering analysis which supports the licensee's position that one fast transfer can occur with no significant change to safety related equipment has been prepared for NRC review. This issue is considered an unresolved item (423/89-21-06), pending review of the licensee actions during subsequent NRC inspection.

6.0 Safety Assessment/Quality Verification (IP30703/40500/90712/92702)

6.1 Committee Activities

The inspector attended several Plant Operations Review Committee (PORC) meetings. Technical Specification 6.5 requirements for required member attendance were verified. The meeting agendas included procedural changes, proposed changes to the technical specifications, plant design change records, and minutes from previous meetings. The PORC meetings were characterized by frank discussions and questioning of the proposed changes. In particular, consideration was given to assure clarity and consistency among procedures. Items for which adequate review time was not available were postponed to allow committee members time for further review and comment. Dissenting opinions were encouraged and resolved to the satisfaction of the committee prior to approval. The inspectors observed that PORC adequately monitors and evaluates plant performance and conducts a thorough self-assessment of plant activities and programs.

6.2 Review of Licensee Event Reports (LERs)

Licensee event reports (LERs) submitted during the report period were reviewed to assess LER accuracy, the adequacy of corrective actions, compliance with 10 CFR 50.73 reporting requirements and to determine if there were generic implications or if further information was required. Selected corrective actions were reviewed for implementation and thoroughness. The LERs reviewed were: 86-11-01; 89-21-00; 89-22-00; 89-23-00; 89-24-00; 89-25-00. The following LERs were selected for additional inspector follow up: 89-21-00; 89-22-00; 89-23-00; 89-24-00 as previously described in this report. Review of LER 89-23-00 described in section 4.4 of this report noted a degradation in fire protection. This condition has also been reported in five other LERs during the previous Systematic Assessment of Licensee Performance (SALP) period. Continued problems in the Fire Protection area warrant licensee attention to assure quality. This issue was discussed with the unit superintendent who noted the inspector's comments.

7.0 Radiological Control (IP 71707/92701)

7.1 Posting and Control of Radiological Areas

During plant tours, posting of contaminated, high airborne, radiation, and high radiation areas were reviewed with respect to boundary identification, locking requirements, and appropriate control points. No inadequacies were noted.

8.0 Management Meetings (30703)

Periodic meetings were held with station management to discuss inspection findings during the inspection period. A summary of findings was also discussed at the completion of the inspection. No proprietary information was covered within the scope of the inspection. No written material was given to the licensee during the inspection period.