U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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EXECUTIVE SUMMARY

Indian Point 3 Nuclear Power Plant
NRC Inspection Report No. 50-286/98010

This inspection included aspects of licensee operations, maintenance, engineering and plant support. The report covered a six-week period of resident and regional inspections.

Operations:

Operator performance during routine activities, including control room alarm responses, communications and shift turnovers, was good. During an observed plant tour and a walk through of an abnormal procedure by plant operators, the operators demonstrated good knowledge of plant equipment and identified several equipment deficiencies. (Section O1.1)

The work planning associated with the 120 volt AC distribution panel No. 32 preventive maintenance package was weak because it did not identify the full impact the associated electrical loads would have on the facility. Following a second NYPA review, the work performed with this package still resulted in unexpected equipment responses as evidenced by an inadvertent containment isolation signal and additional challenges to operations. The quality standards and expectations of several personnel associated with the preparation of this work appeared inconsistent with those of senior plant management. (Section 01.2)

The licensee's immediate response to the identification of the depleted nitrogen bottle to the control room ventilation system was weak. The licensee did not consider potential implications of the rapid reduction in the pressure on the leak tight integrity of the air operation damper actuators and associated piping. This reflected a lack of questioning by personnel screening the deviation event report (DER). Subsequent planned and completed actions by the licensee were appropriate. (Section O7.1)

The previous day prior to the identification of the depleted backup nitrogen bottle for the control room ventilation system, a nuclear plant operator (NPO) failed to notify his supervision of the significant drop in backup nitrogen pressure. As a result, the licensee failed to promptly identify and correct this condition adverse to quality. Although adequate, the licensee's corrective action in response to this failure was weak due to oversights in documenting the NPO performance issue in deviation event report 99-0015. This oversight resulted in the loss of human performance error data point and reduced the reliability of performance trends, which the licensee monitors. The failure to promptly identify and correct a condition adverse to quality is a violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Actions." However, the violation is considered non-cited because the issues were licensee identified and the corrective actions taken were adequate, consistent with Section VII.B.1 of the NRC Enforcement Policy. **(NCV 98010-01)** (Section O7.1)

The licensee's corrective actions were weak in response to a licensee identified deficiency in 1996, in which the minimum nitrogen bottle pressure for the isolation valve seal water system was not maintained during normal standby conditions. The licensee inappropriately revised procedures to change out nitrogen bottles at 1200 psig during both post accident and normal standby conditions rather than during normal standby conditions only, and did not consider whether additional bottles needed to be staged. (Section O7.2)

Maintenance:

Maintenance activities observed were conducted satisfactorily and in accordance with applicable maintenance and administrative procedures. The licensee was appropriately monitoring performance for equipment within the scope of the maintenance rule. Surveillances were conducted appropriately and in accordance with procedural and administrative requirements. As applicable, good coordination and communication with the control room was observed during performance of the surveillance. The test instrumentation was within calibration, and the acceptance criteria were achieved. (Sections M1.1 and M1.4)

Maintenance work packages and procedures used to perform the quarterly preventive maintenance (PM) work on the 32 emergency diesel generator and its associated auxiliary equipment were appropriate to the circumstance. In addition, discrepancies that arose during the implementation of the PM were appropriately dispositioned by maintenance personnel. Lastly, the post maintenance critique identified several areas for improvement and addressed lessons learned from the activity. (Section M1.2)

The licensee effectively implemented a temporary modification on a post-accident sample line that was a part of the containment boundary. The planning and preparation, pre-job briefing, and work coordination and support were excellent, and resulted in minimizing the time in which the plant was in a one-hour limiting condition for operation for containment integrity. The nuclear safety evaluation for installing the temporary modification was appropriate and used risk insights to limit the time that the temporary modification could be installed. (Section M1.3)

Test procedure 3PT-Q100, Turbine First Stage Pressure Analog Channel, was well written and its acceptance criteria appropriately ensured that technical specifications requirements were met. The calculation, which supported the test acceptance criteria, was of high quality and was thorough in considering all potential instrument loop inaccuracies. The technicians' performance of the test was good. (Section M1.5)

The performance of radiography to determine the extent of erosion/corrosion in the service water piping was appropriate. Minor discrepancies with the quality assurance of the weld picture and the numerical designations on the radiographs were identified; however, the information derived from the radiographs was acceptable. (Section M2.1)

The repair of a socket weld in April 1998 was weak in that it did not capture an engineering design change to install a new shim design which may have precluded a second failure of the same socket weld in December 1998. The licensee missed several opportunities to identify this weakness during the review of the completed work package after the first failure. (Section M2.2)

Executive Summary (cont'd)

Engineering:

The identification and resolution of a containment isolation valve testing deficiency due to a 75 pound spring loaded check valve, which may have invalidated the differential pressure conditions of a containment isolation valve leak test, was good. The corrective actions and extent of condition review for this test deficiency were appropriate. Notwithstanding, the original containment isolation valve leak test was inadequate and is a violation of 10 CFR Appendix B, Criterion V. However, because this issue was identified by the licensee, and adequate corrective actions were promptly taken upon discovery, this violation is being treated as a Non-Cited Violation. **(NCV 50-286/98010-02)** (Section E1.1)

The licensee's immediate actions to a NRC identified single failure vulnerability of the containment vent system were appropriate. However, this vulnerability reflected inadequate design control and is a potential violation of 10 CFR 50, Appendix B, Criterion 3 **(EEI 98010-03)**. The assessment of the licensee's long term corrective actions, causal analysis and safety significance determination remains open, pending the submittal of a licensee event report on this design issue. (Section E2.1)

The reasonable expectation of operability performed by engineering in response to a seismic qualification concern of grating in the diesel generator room was appropriate. The assumptions used in the evaluation were appropriate to the expected seismic conditions specified in the plant design. (Section E2.2)

The scope of the licensee's service water action plan was extensive and appropriately addressed equipment performance issues associated with the service water system. Based on the service water system improvements made by the licensee, the circumstances surrounding need for the notice of enforcement discretion was determined not to be a violation of NRC requirements. Therefore, unresolved item 50-286/98007-04 is closed. (Section E8.1)

Plant Support:

The licensee implemented effective applied radiological controls. Access controls to radiologically controlled areas were effective, and appropriate occupational exposure monitoring devices were provided and used. Personnel occupational exposure was maintained within applicable regulatory limits and as low as is reasonably achievable (ALARA). The radiation work permit program was properly implemented. (Section R1.1)

The licensee implemented overall effective surveys, monitoring, and control of radioactive materials and contamination. Health Physics technicians performed proper surveys and properly documented survey results. Radiological housekeeping conditions were noted to be good. The personnel contamination rate was tracked and trended. The radiological surveys, monitoring, and controls were implemented with calibrated and properly used devices. (Section R1.2)

The licensee implemented a very effective program to maintain occupational radiation exposure as low as is reasonably achievable (ALARA), and the ALARA efforts and results for 1998 were exceptionally good, including the management of radiologically significant work and a station record for lowest annual person-rem. (Section R1.3)

The licensee's self-identification and corrective action processes in the area of radiation protection were effective. A Quality Assurance audit, self-assessments, and the corrective action program continued to be effective in identifying, at a low threshold, deficiencies and improvement opportunities. Effective corrective actions were implemented for findings. (Section R7)

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- Items Opened, Closed, and Discussed
- List of Acronyms Used

Report Details

SUMMARY OF PLANT STATUS

Indian Point Unit 3 was operated at full power during this inspection report period.

I. OPERATIONS

O1 Conduct of Operations

O1.1 General Observation

a. Inspection Scope (71707)

The inspection consisted of observations of shift turnover meetings and control room activities, selected verifications of equipment configuration and status, and tours within the facility.

b. Observations and Findings

Overall, the inspectors observed good performance in the area of shift turnovers and control room communications. Control room operators responded to alarms and abnormal plant conditions appropriately. These conditions were properly logged and deviation event reports were initiated when appropriate. Good response was noted when a fire under the spare circulating water pump motor was suspected. Although there was not fire (only smoke from portable heaters), the fire brigade response was timely, and control room operators kept the plant staff well informed of the status of the event.

Operator knowledge of the plant equipment and abnormal procedures was noteworthy. The inspector observed a nuclear plant operator (NPO) tour and noted that the NPO was knowledgeable of watch objectives and plant log readings. The NPO was thorough in performing his tours, as evidence by the identification of several minor packing leaks and deficiencies. The inspector also observed a licensed operator walk through portions of ONOP-FP-1A, "Safe Shutdown from Outside the Control Room." The licensed operator was knowledgeable of the procedure and associated plant equipment. The quality of the procedure was good.

c. Conclusions

Operator performance during routine activities, including control room alarm responses, communications and shift turnovers, was good. During an observed plant tour and a walk through of an abnormal procedure by plant operators, the operators demonstrated good knowledge of plant equipment and identified several equipment deficiencies.

O1.2 120-Volt AC Distribution Panel No. 32

a. Inspection Scope (71707)

The inspector reviewed work request package 97-05011-00, "Preventive maintenance on the 120 volt-AC distribution panel 32." This preventive maintenance package was part of a licensee initiative to inspect molded circuit breakers for cracking in response to recent industry information regarding the potential for cracking in a specific type of circuit breaker. The review included the adequacy of the circuit loading diagrams used to prepare the package and the impact on plant operations as a result of the implementation of the work package.

b. Observations and Findings

On January 27, 1999, the inspector reviewed a work package in ready status that was on the schedule for that morning. The inspector observed hand written notes on the 32 ac panel distribution circuit diagram (93210-LL-30412) that indicated some concerns still existed as to exactly what equipment would be removed from service when this panel was de-energized. The inspector raised the question of the adequacy of the drawing to the field support supervisor at that time. Further discussions with operations personnel revealed that the circuit drawing used to develop the package was not verified and may not have listed all the electrical loads which would be affected when the panel was de-energized. Subsequently, the licensee put the work package in a hold status until an additional review could be performed to resolve the outstanding operational concerns. Although the reason why the licensee placed the work package on hold was indeterminate, the inspector concluded that the plant staff that planned, reviewed and approved the package exhibited poor quality standards in accepting this deficient work package.

The licensee's additional review resulted in the identification of nine additional limiting conditions for operation (LCOs) added to the work package. These LCOs were for equipment in the licensee's Technical Specifications and Operation Specifications. The second review also provided a more detailed circuit by circuit description of the operational concerns associated with de-energizing this panel. This was necessary because the original review did not clearly address the potential concerns for each circuit. This was a weakness in the work control process as it created additional challenges to plant operations personnel.

The package was released for work on January 29, 1999. When the panel was de-energized, unexpected electrical loads were shed causing spurious responses in the plant. These unexpected responses included an auto start of the fuel storage building vent, waste processing secured due to an erroneous chemical and volume control system low pressure alarm, a containment isolation signal was initiated which isolated the pressurizer relief tank gas analyzer containment isolation valve (AOV-549) and several personal safety lights were extinguished. A deviation event report (DER) was initiated regarding the unexpected loads on the circuit diagram.

The inspector questioned the practices of using unverified drawings to prepare a work package which has the potential to unexpectedly change the state of operationally important equipment in the plant. Discussions with engineering revealed that the problems with the panel distribution diagrams are known; however, it was decided to address the problem on a case by case basis as the PMs for the panels are planned and implemented. Although the panels in question do not directly impact equipment which has been designated as safety related, it does create additional challenges for the operators in preparing the package, scheduling concurrent work and identifying and responding to equipment that unexpectedly changes state as a result of the panel de-energization

c. Conclusions

The work planning associated with the 120 volt AC distribution panel No. 32 preventive maintenance package was weak because it did not identify the full impact the associated electrical loads would have on the facility. Following a second NYPA review, the work performed with this package still resulted in unexpected equipment responses as evidenced by an inadvertent containment isolation signal and additional challenges to operations. The quality standards and expectations of several personnel associated with the preparation of this work appeared inconsistent with those of senior plant management.

O7 Quality Assurance in Operations Activities

O7.1 Backup Nitrogen to the Control Room Ventilation System

a. Inspection Scope (37551, 71707)

During periodic review of deviation event reports (DERs), the inspector noted DER 99-0015, which documented the depletion of the backup nitrogen bottle to the control room ventilation system. The inspector reviewed and assessed the licensee's immediate corrective actions to the identified deficiency.

b. Observations and Findings

On January 4, 1999, a nuclear plant operator (NPO) identified that the backup nitrogen bottle for the control room ventilation system was depleted. Backup nitrogen is used to actuate dampers within the control room ventilation system, in the event that instrument air, which is not safety-related, is unavailable. In response to the identified deficiency, the control room declared the control room ventilation system inoperable and entered a 72 hour limiting condition for operations (LCO) at 5:00 p.m. The nitrogen bottle was changed out and snooped for leaks. The LCO was exited at 6:30 p.m., and a DER was initiated.

On January 6, 1999, the inspector reviewed DER 99-0015 on the depleted nitrogen bottle. The inspector interviewed personnel from operations, system engineering and the operations review group to ascertain the extent of the licensee's immediate corrective actions and determined that a work request for replacing the pressure regulator for this backup nitrogen bottle was being expedited. The licensee believed that the regulator may have been set slightly higher than the intended 20 psig as a result of difficulty in setting 20 psig on a 400 psig range regulator. As a result, the pressure may have been higher than instrument air pressure and the control room ventilation system may have been using and exhausting the nitrogen bottle. Another corrective action initiated was for operations to determine the cause of the depletion. This action was identified as a long term action to close out the DER. The response time for DER responses is 21 days.

The inspector, however, was concerned that the potential implication of the rapid depletion of the nitrogen was not thoroughly assessed by the licensee. Specifically, the licensee did not consider the potential operability implication if, in fact, the control room ventilation system was using nitrogen rather than instrument air. The nitrogen bottle pressure is recorded by the NPOs daily, and during the two days previous to identifying the bottle depleted, nitrogen pressure was 1850 psig and 900 psig, respectively. If the nitrogen depletion was caused through usage by the control room ventilation system, this rate of depletion represented usage of about 950 psig per day. However, IP3 calculation IP3-CALC-IA-00969 determined that a pressure of 165.4 psig was sufficient to support the control room ventilation system for 24 hours.

The inspector determined that this concern was not considered during the DER review committee or by operations personnel. Subsequent to the inspector raising this concern, the licensee indicated that a hand over hand walkdown of the air operated equipment in the control room ventilation system was performed, and no gross leakage was noted. As a result, the licensee concluded that the likely source of the leak was the connection between the regulator and nitrogen bottles. The new connection was snooped for leakage. In addition, system engineering indicated that they were planning to performed a leakage test of the system and were considering developing a periodic test to determine system leakage.

The inspector noted another issue associated with the timeliness of identification of the depleted nitrogen bottle. As previously indicated, the nitrogen backup pressure is recorded daily by the nuclear plant operator (NPO). The control room logs indicated that prior to the NPO identifying that the nitrogen bottle was depleted on January 4, 1999, the nitrogen pressures were about 1850 psig and 900 psig on January 2 and 3, respectively. Discussions with the control supervisor indicated that the NPO had identified that the pressure had dropped to about 900 psig during the previously day, but had forgotten to notify his supervision after his rounds.

In reviewing the completed DER to determine how this issue was addressed by the licensee, the inspector noted that the DER did not addressed the performance issue associated with the untimely identification of the deficiency. Discussions with operations management indicated that actions were taken to emphasis lessons learned from this incident; however, the licensee did not document this issue in the DER. The operations manager stated that the DER would be reopened to address and document this issue. The inspector considered the documentation of corrective actions for this DER to be weak. This lack of documentation resulted in the loss of a human performance error data point and reduced the reliability of performance trends, which the licensee monitors. The licensee acknowledged this observation.

The inspector determined that the nuclear plant operator failed to promptly identify a condition adverse to quality to his supervision, which resulted in the failure to promptly correct the deficiency. This failure is a violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Actions." However, the violation is considered non-cited because it was licensee identified. Although the licensee's corrective actions were weak due to documentation oversights, the licensee's corrective actions to the untimely identification of the deficiency were adequate. **(NCV 98010-01)**

c. Conclusions

The licensee's immediate response to the identification of the depleted nitrogen bottle to the control room ventilation system was weak. The licensee did not consider potential implications of the rapid reduction in the pressure on the leak tight integrity of the air operation damper actuators and associated piping. This reflected a lack of questioning by personnel screening the deviation event report (DER). Subsequent planned and completed actions by the licensee appeared appropriate.

The previous day prior to the identification of the depleted backup nitrogen bottle for the control room ventilation system, a nuclear plant operator (NPO) failed to notify his supervision of the significant drop in backup nitrogen pressure. As a result, the licensee failed to promptly identify and correct this condition adverse to quality. Although adequate, the licensee's corrective action in response to this failure was weak due to oversights in documenting the NPO performance issue in deviation event report 99-0015. This oversight resulted in the loss of human performance error data point and reduced the reliability of performance trends, which the licensee monitors. The failure to promptly identify and correct a condition adverse to quality is a violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Actions." However, the violation is considered non-cited because the issues were licensee identified and the corrective actions taken were adequate, consistent with Section VII.B.1 of the NRC Enforcement Policy. **(NCV 98010-01)**

O7.2 Isolation Valve Seal Water System Walkdown

a. Inspection Scope (37551, 71707)

The inspector conducted a walkdown of the isolation valve seal water (IVSW) system, verified administrative controls and procedures to support system operations, and reviewed the design basis document to assess the ability of the system to perform its design function.

b. Observations and Findings

The IVSW system provides either pressurized water or nitrogen gas between containment isolation valves or between the discs of containment isolation gate valves. Pressurized water to the containment isolation valves is provided both automatically and manually from the IVSW seal tank at about 60 psig. Pressurized nitrogen to the containment isolation valves is provided manually from standard compressed nitrogen bottles at about 250 psig. The same nitrogen bottles provide the pressure source to pressurize the IVSW seal tank to about 60 psig. The inspector noted that the overall material condition of the system was good. No significant equipment deficiencies were identified during the walkdown. System indications were consistent with indications expected when the system is in standby. Major system valves were properly positioned and controlled.

The inspector reviewed the system design basis document and calculations, and identified that the design basis requirement for nitrogen did not include potential leakage through the containment isolation valves that are sealed by 250 psig nitrogen. The inspector reviewed the latest containment local leak rate test results on these containment isolation valves and determined that the leakages from the nitrogen-supplied valves were much more significant than the leakages from the seal water supplied valves. Upon raising this concern, the system engineer indicated that this concern was previously identified by NYPA, and as a result a draft calculation was developed and in the process of being reviewed and finalized.

Although draft, the inspector reviewed the calculation and identified two questionable assumptions. The first assumption was that the leakages through the nitrogen-supplied valves would be limited by the permissible leakages established by NYPA to meet containment integrity (i.e., containment local leak rate testing). However, the permissible containment isolation valve leakage testing, which is based upon maximum containment pressure, is performed at about 50 psig. The nitrogen pressure to these valves is about 250 psig.

The second assumption was that the unusable volume of nitrogen in the bottle occurs when the pressure in the bottle drops to about 250 psig. However, the inspector noted that, during a post accident scenario, Indian Point 3 procedure SOP-CB-11, revision 6, Non-Automatic Containment Isolation, Attachment 1, Valves Local Lineup, directs operators to replace the bottles if the pressure reaches 1200 psig. The inspector subsequently determined that this inconsistency was due to an inappropriate procedure change in 1996, in response to a licensee identified deficiency concerning the minimum nitrogen bottle pressure for the IVSW system during normal standby conditions. The licensee had inappropriately imposed this pressure limit for the system during both post accident and normal standby conditions, rather than during normal standby conditions only.

Based on the draft calculation results and the number of nitrogen bottles that NYPA dedicated to emergency operating procedures, the inspector determined that the inappropriate procedure change reduced the assurance that an adequate dedicated supply of nitrogen would be available. However, the inspector did not consider this condition to be a safety concern because actual leakage during the latest containment leakage test showed valve leakages were only 1 - 2% of the permissible limit. Further, other sources of nitrogen existed that can supply the IVSW system.

c. <u>Conclusions</u>

The licensee's corrective actions were weak in response to a licensee identified deficiency in 1996, in which the minimum nitrogen bottle pressure for the isolation valve seal water system was not maintained during normal standby conditions. The licensee inappropriately revised procedures to change out nitrogen bottles at 1200 psig during both post accident and normal standby conditions rather than during normal standby conditions only, and did not consider whether additional bottles needed to be staged.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Maintenance General Comments (62707)

a. Inspection Scope (62707)

The inspectors reviewed selected maintenance work activities and supporting work documentation. Activities were selected based on the systems, structures, or components being contained within the scope of the maintenance rule.

b. Observations and Findings

The inspectors observed all or portions of the following work activities:

- WR 96-07704-04, Replacement of Field Flash Resistors in the 32 EDG Control Cabinet
- WR 98-00164-00, Emergency Diesel Generator Quarterly Preventive Maintenance
- WR 98-03027-00, Inspection of Emergency Diesel Generator Exhaust Fan 316
- WR 98-03028-00, Inspection of Emergency Diesel Generator Exhaust Fan Motor
- WR 98-05042-00, 36 Service Water Pump Replacement
- WR 98-05050-00, Exhaust Damper Air Start Motor
- WR 98-05396-02, Inspection of 31 Service Water Pump Breaker Cubicle
- WR 99-00057-00, 32 Instrument Air Dryer Corrective Maintenance
- WR 99-00242-00, Install Temporary Modification to Line Going to SP-MOV-990A

c. Conclusions

Maintenance activities observed were conducted satisfactorily and in accordance with applicable maintenance and administrative procedures. The licensee was appropriately monitoring performance for equipment within the scope of the maintenance rule.

M1.2 Emergency Diesel Maintenance

a. Inspection Scope (62707)

The inspector observed the implementation of the preventive maintenance (PM) work on the 32 emergency diesel generator (EDG). The scope of work performed during the 72 hour limiting condition for operation (LCO) work included the quarterly PM for the diesel engine, two corrective maintenance packages and 3 additional preventive maintenance packages on the diesel generator auxiliary support systems.

b. Observations and Findings

Overall, the work packages and procedures were thorough and appropriate to the work being performed.

A temporary procedure change (TPC) was required for the PM procedure for the EDG exhaust fan," FAN -010-VSS". The TPC was due to the runout measurement of the fan sheave being out of specification. The mechanic appropriately contacted the maintenance engineer to review the data. The EDG exhaust fan sheave is not adjustable; however, the procedure required adjustments to the sheave if the runout data was out of specification. A review of the data indicated that sheave was in acceptable condition and the runout data did not indicate a degraded component. Additionally, vibration data from December 1997 and after the PM was completed indicated that the equipment was functioning properly.

During the maintenance work, it was noted that the gratings on the 32 foot elevation of the diesel room were not fixed in position. This is of concern due to the seismic design requirements of the area. System engineering provided a reasonable expectation of operability (REO) to the shift manager upon identification of this issue. Section E.2.2 details the inspectors' review of the REO.

Prior to running the diesel for the retest after the PM, the jacket water that had been removed had to be heated back up to operating temperature of 120° F. This took approximately 13 hours. While all of the work was completed in a timely manner, the restoration for the retest used up a considerable amount of LCO time. The post maintenance critique for the diesel identified several ways to solve this problem and thereby reduce the unavailability of the diesel. In addition, the critique identified several other issues associated with the work performed on the diesel including actions to capture the lessons learned from this PM and to correct any potential repeat discrepancies.

c. Conclusions

Maintenance work packages and procedures used to perform the quarterly preventive maintenance (PM) work on the 32 emergency diesel generator and its associated auxiliary equipment were appropriate to the circumstance. In addition, discrepancies that arose during the implementation of the PM were appropriately dispositioned by maintenance personnel. Lastly, the post maintenance critique identified several areas for improvement and addressed lessons learned from the activity.

M1.3 Sample System Temporary Modification

a. Inspection Scope (37551, 61726, 62707)

Section E1.1 of this report describes the licensee's identification of inadequate leak testing associated with containment isolation valve SP-MOV-990A, which is in the recirculation discharge post accident sample line. In response to this identified deficiency, the licensee implemented a temporary modification to the sample system to allow the testing of valve SP-MOV-990A. The inspector reviewed the associated nuclear safety evaluation, and observed the maintenance activities to install and test the temporary modification.

b. Observations and Findings

The licensee developed nuclear safety evaluation (NSE) 99-3-006PS to address the adequacy of disconnecting and installing a 3/8-inch shut-off valve and a tubing cap upstream the inboard isolation valve of the recirculation discharge sample line through temporary modification (TM) 99-00242-00. The modification would allow the leak testing of valve SP-MOV-990A by ensuring the upstream side of the valve would be depressurized.

The inspector reviewed the nuclear safety evaluation that supported the temporary modification and determined the licensee's evaluation to be appropriate. The licensee considered the impact of the unavailability of normal post accident sampling for the containment sump. The licensee identified alternate methods of obtaining a post accident containment sump sample and limited the time that the temporary modification would be allowed to be installed to seven days. Although the alternate methods would result in bringing post accident water outside of containment either through the residual heat removal system or high head safety injection system, the licensee's nuclear systems analysis group determined that the probability of core damage within any seven day period would be non-risk significant.

The work planning and preparation, pre-job briefing, and work coordination and support for installing the temporary modification were excellent. As a result, the time in which the plant was in a one-hour limiting condition for operation for containment integrity was minimized. A total of 26 minutes was required to install and test the temporary modification. Prior to the installation of the modification, the licensee tested the shut-off valve to be installed to ensure it met containment leakage requirements, entered containment to isolate the sample line, and vented the line to ensure no leakage during the installation of the modification. During the pre-job briefing, contingency actions, communications with the control room, the scope and location of the work were emphasized. During the installation of the temporary modification, good support was noted by all work groups. In particular, a design engineer was present and provided the workers information on re-testing when a question regarding the time criteria for the post maintenance testing was raised.

The post maintenance testing adequately ensured that the newly installed mechanical joints were leak tight. However, the procedure did not specify a period of time after the new connections were pressurized for snooping for leaks. This was clarified verbally by the design engineer who was present at the work site. Also, the inspector noted that the procedure specified performing contingency testing in series with the snoop test. Specifically, if the snooping of the new connections identified some leakage, a type C leak test would be required to be performed to verify that amount of leakage was acceptable. However, the contingency type C leak test could have been performed concurrently with the ten-minute snoop test. This would have minimized the time the plant was in a one-hour LCO if the snoop test were to have failed.

The temporary modification was installed on January 13, 1999. After successful type C testing of valve SP-MOV-990A, the temporary modification was removed on January 14, 1999. The period of time, in which the temporary modification was installed, was within the seven day period specified by the nuclear safety evaluation.

c. Conclusions

The licensee effectively implemented a temporary modification on a post-accident sample line that was a part of the containment boundary. The planning and preparation, pre-job briefing, and work coordination and support were excellent, and resulted in minimizing the time in which the plant was in a one-hour limiting condition for operation for containment integrity. The nuclear safety evaluation for installing the temporary modification was appropriate and used risk insights to specify the time that the temporary modification could be installed.

M1.4 Surveillance General Comments (61726)

a. Inspection Scope (61726)

The inspectors reviewed selected surveillance activities and supporting documentation. Activities were selected based on the systems, structures, or components being contained within the scope of the maintenance rule.

b. Observations and Findings

The inspectors observed all or portions of the following surveillances:

- 3PT-M79B, 32 Emergency Diesel Generator Quarterly Functional
- 3PT-M79C, 33 Emergency Diesel Generator Quarterly Functional
- 3PT-Q100, Turbine First Stage Pressure Analog Channel
- WR 99-00242-05, Post Work Testing for Temporary Modification 99-00242-0

c. Conclusions

Surveillances were conducted appropriately and in accordance with procedural and administrative requirements. As applicable, good coordination and communication with the control room was observed during performance of the surveillance. The test instrumentation was within calibration, and the acceptance criteria were achieved.

M1.5 Turbine First Stage Pressure Analog Channel Quarterly Test

a. Inspection Scope (61726)

The inspector conducted a detailed review of procedure 3PT-Q100, revision 9, Turbine First Stage Pressure Analog Channel. This detailed review included observing the test being performed on January 7, 1999. Procedure 3PT-Q100 is performed quarterly to demonstrate the operability of the first stage turbine bistables, high steam flow injection bistables and programmed setpoints.

b. Observations and Findings

The instrument and control (I&C) technicians demonstrated good performance in implementing procedure 3PT-Q100. Throughout the performance of the procedure, the technicians demonstrated the use of the STAR (stop, think, act, review) technique, good knowledge of the procedure and system operation, clear communications and good procedural adherence. During the test, the technicians identified a potential enhancement to the procedure and provided feedback to the procedure writer.

The procedure was well written, and provided important information and guidance to the technicians and control room operators on the potential impact of performing the test on the plant. The surveillance test accurately reflected information in calculation IP3-CALC-ESS-00277, revision 2, Engineered Safety Feature Actuation System Instrument Loop Accuracy/Setpoint Calculation/High Steam Flow. This calculation determined the acceptable test criteria to ensure that technical specification requirements were met. The calculation was thorough in considering all potential instrument loop inaccuracies.

The inspector also verified the adequacy of following: administrative approvals and reviews, qualification status of the technicians, calibration of test equipment, test data results, and system restoration. No discrepancies were identified.

c. Conclusions

Test procedure 3PT-Q100, Turbine First Stage Pressure Analog Channel, was well written and its acceptance criteria appropriately ensured that technical specifications requirements were met. The calculation, which supported the test acceptance criteria, was of high quality and was thorough in considering all potential instrument loop inaccuracies. The technicians' performance of the test was good.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Radiography of Cement Lined Service Water Piping

a. Inspection Scope (49001)

The inspector reviewed radiography procedure NDEP9.3-1, two associated addenda to the procedure and a sample of ten radiographs (RTs) to assess the quality of the non destructive examinations being performed on cement lined piping.

b. Observations and Findings

The inspector reviewed procedure NDEP 9.3-1, which describes methods for radiography of welds and pipe to determine the extent of erosion/corrosion in cement lined piping. A sample of ten radiographic film sets representative of various size pipe welds and piping to elbows and tees were reviewed. The RTs examined clearly showed the presence or absence of pipe metal and/or cement lining corrosion. The addenda 5.6 and 5.7 of procedure NDEP 9.3-1 stated that penetrameters (or image quality indicators) *may* be used as an evaluation aid. The inspector observed that penetrameters were not present on any of the RTs performed to determine the extent of corrosion in the piping. Additionally, on the RT set for weld 10"-11B-VC 5-96-015118-26, the suffix 26 was incorrectly present on the radiograph as "25". While this suffix did not result in loss of traceability of the RT to the corresponding weld, it was an error in RT identification by the RT Contractor and an oversight in later RT review. The NYPA staff indicated that additional RTs and the corresponding documentation will be reviewed to determine if more or similar problems are present.

c. Conclusions

The performance of radiography to determine the extent of erosion/corrosion in the service water piping was appropriate. Minor discrepancies with the quality assurance of the weld picture and the numerical designations on the radiographs were identified; however, the information derived from the radiographs was acceptable.

M2.2 Chemical and Volume Control System Drain Valve Leaks.

a. Inspection Scope (62707)

The inspector reviewed the work packages and engineering drawings associated with the repeat failure of a socket weld on the chemical and volume control system (CVCS) drain valve, CH-148.

b. Observations & Findings

NRC inspection reports 50-286/98002 and 50-286/98009 document the failure of a socket weld on CVCS drain valve CH-148. The licensee has developed and is implementing an action plan to address the excessive vibrations believed to cause this repetitive failure.

The inspector reviewed an engineering modification package implemented in 1995 to reduce the effects of the vibration on the system and how errors in that modification may have impacted the current performance of the socket weld. In 1995, a modification to minimize the probability of fatigue failure due to the vibratory interaction in the CVCS piping was installed, but did not include an engineering change notice (ECN) change on a shim detail between a U-bolt and the CH-148 valve body. Specifically, the new shim was designed to be captivated by the U-bolt assembly. The drawing was changed in 1996, but the difference between the shim detail as installed versus that on the drawing (9321-LL-60112, Sheet 60) was not reconciled. The work package issued for the repair of the socket weld leak in April 1998 did not have a work step for either reuse or replacement of the shim. The work process, including drawing interpretation, resulted in the original but incorrect shim being used in April. The incorrect shim configuration did not include a provision to ensure the shim was properly secured in the U-bolt assembly. The result of this inappropriate use of the incorrect shim was that the shim fell out, providing increased vibration of the piping that may have contributed to the rapid fatigue failure in December 1998. When the repairs were made for the second weld leak in December, the correct shim was used, however the use of the shim is not shown in the work steps and not recorded in the work package. This is contrary to the step text in the package which requires the mechanics to list all materials used during the repair. The shim material was listed on the contingency materials list.

The inspector concluded that the licensee missed opportunities to identify and correct an error in the configuration of the piping supports. These include recognizing a design change in an engineering drawing, addressing configuration control after an ECN incorporation, verifying work package completeness consistent with skill of the craft and overall review of a completed work package.

c. Conclusions

The repair of a socket weld in April 1998 was weak in that it did not capture an engineering design change to install a new shim design which may have precluded a second failure of the same socket weld in December 1998. The licensee missed several opportunities to identify this weakness during the review of the completed work package after the first failure.

III. ENGINEERING

E1 Conduct of Engineering

E1.1 Containment Isolation Valve Leak Testing

a. Inspection Scope (37551)

The inspector reviewed the licensee's actions regarding an additional identification of a potentially inadequate containment isolation valve test.

b. Observations and Findings

On January 8, 1999, a system engineer identified an potentially inadequate containment isolation valve test for the containment recirculation sampling valve, SP-MOV-990A. The system engineer was reviewing the completed re-tests which resulted after a forced outage in November due to the original identification of inadequate containment isolation valve testing for several of the stations surveillances . NRC Inspection report 50-286/98009 documented the identification and resolution of the containment isolation valve issue. Special re-tests were developed and performed to verify the operability of the licensee's containment isolation valves. In this case, the system engineer identified a unique condition regarding a spring loaded check valve in the piping system associated with the SP-MOV-990A. This spring check valve had a loading of approximately 75 pounds which needed to be overcome prior to the valve lifting off its seat. This additional loading was not accounted for in the re-test, and therefore the pressure differential of at least 43 psig across the containment isolation valve could not be verified. This is because the spring load was not accounted for when the operators depressurized the upstream portion of the system prior to testing SP-MOV-990A.

Immediate corrective actions included de-energizing both the inboard (990A) and outboard containment isolation valves (990B). An extent of condition looking for other check valves with spring loads on them revealed several other spring loaded check valves; however, these valves had low opening pressures and would not have impacted the any other test results. The licensee developed a second test for SP-MOV-990A and satisfactorily tested the valve on January 14, 1999.

The inspector reviewed the licensee's actions in response to this event and observed that this spring check valve appeared to be unique and the identification of this issue by the system engineer was good. The extent of condition review for this event was appropriate to the circumstances. Notwithstanding, the original containment isolation valve leak test procedure was inadequate and is a violation of 10 CFR Appendix B, Criterion V. However, because this issue was identified by the licensee, and adequate corrective actions were promptly taken upon discovery, this violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1 of the NRC Enforcement Policy. **(NCV 50-286/98010-02)**

c. Conclusions

The identification and resolution of a containment isolation valve testing deficiency due to a 75 pound spring loaded check valve, which may have invalidated the differential pressure conditions of a containment isolation valve leak test, was good. The corrective actions and extent of condition review for this test deficiency were appropriate. Notwithstanding, the original containment isolation valve leak test was inadequate and is a violation of 10 CFR Appendix B, Criterion V. However, because this issue was identified by the licensee, and adequate corrective actions were promptly taken upon discovery, this violation is being treated as a Non-Cited Violation. **(NCV 50-286/98010-02)**

E2 Engineering Support of Facilities and Equipment

E2.1 Containment Vent Design Deficiency

a. Inspection Scope (37551, 71707)

On January 21, 1999, the inspector identified a potential safety single failure design issue associated with the containment vent system. The inspector assessed the licensee's immediate corrective actions in response to this identified deficiency.

b. Observation and Findings

The containment vent system consists of three 10-inch valves in series (one valve located inside containment; the other two located outside), which are normally closed during plant operations. The spaces between the valves are normally pressurized with 55 psig air from the weld channel and containment penetration pressurization system (WCCPPS). These valves are open periodically during power operations to relieve pressure within the containment caused by nitrogen and air leakage from components within the containment. Because a direct pathway through containment is created these three valves automatically close when a phase A containment isolation signal is received.

When the containment vent valves are to be opened, 55 psig air is vented from the between the containment vent valves through three-way solenoid valves to a piping penetration area located outside of containment. Solenoid valve SOV-1280 vents the space between the inside containment vent valve and the outside valve that is closest to containment.

The inspector noted that if the inside containment isolation valve were to fail to close during a venting evolution, an interlock would prevent valve SOV-1280 from closing. The resulting postulated single failure would result in a open pathway from containment through the inside containment vent valve and the one-inch solenoid valve 1280 to an area outside containment.

The inspector raised this concern to the system engineer on the evening of January 21, 1999. By the morning of January 22, 1999, the licensee confirmed the inspector's observation, determined that this was a condition outside the plant design basis and made a 10 CFR 50.72 notification to the NRC. In addition, the licensee administratively precluded containment vent activities until a method of venting could be determined and implemented which would not be susceptible to a single failure vulnerability.

Immediate corrective actions included a temporary procedure change to SOP-CB-3, "Containment Pressure Relief and Purge System Operation." This procedure change included a step to close the manual isolation valve 1110-7 between the SOV-1280 and the containment vent valves. Also, the power leads for SOV-1280 were lifted during the pressure relief to assure the solenoid remained in the position to supply weld channel to the isolation valves 1190 and 1191. The licensee developed and implemented a temporary modification to address this issue until a permanent solution can be implemented. The temporary modification added an additional section of piping onto the vent portion of the solenoid valve. This section of piping has a manual isolation valve and a cap at the end of the pipe. The manual isolation valve was leak tested in accordance with the technical specification requirements prior to installation and will be credited as an isolation valve to meet the containment integrity technical specification. The inspector reviewed the temporary modification and its associated nuclear safety evaluation and noted that they were appropriate to the circumstance. The licensee's long term corrective actions remain open and a licensee event report is planned for submittal to the NRC.

The inspector determined that the single failure vulnerability of the containment vent system reflected inadequate design control and is a potential violation of 10 CFR 50, Appendix B, Criterion 3 **(EEI 98010-03)**. The assessment of the licensee's long term corrective actions, causal analysis and safety significance determination remains open, pending the submittal of a licensee event report on this design issue.

c. Conclusions

The licensee's immediate actions to a NRC identified single failure vulnerability of the containment vent system was appropriate. However, this vulnerability reflected inadequate design control and is a potential violation of 10 CFR 50, Appendix B, Criterion 3 **(EEI 98010-03)**. The assessment of the licensee's long term corrective actions, causal analysis and safety significance determination remains open, pending the submittal of a licensee event report on this design issue.

E2.2 Seismic Grating in the Emergency Diesel Generator Room

a. Inspection Scope (37551)

The inspector reviewed the reasonable expectation of operability (REO) performed for the loose grating on the 32' elevation of the emergency diesel generator room.

b. Observations and Findings

On January 27, 1999, a maintenance worker identified that the grating on the 32' elevation of the diesel platform did not appear to be seismically restrained in the north/south direction. Engineering provided a reasonable expectation of operability (REO) to the shift manager to provide assurance that the loose grating met the seismic requirements of the emergency diesel generator room. The REO assumed a damping factor of 1%, which provided a peak acceleration factor of 0.5g. The REO concluded that since this was less than the steel to steel friction factor of 0.7, the seismic capability of the emergency diesel grating on the 32 foot elevation was adequate. The inspector noted that the seismic assumptions in the REO were appropriate, and the determination was provided to the shift manager in a timely manner.

c. Conclusions

The reasonable expectation of operability performed by engineering in response to a seismic qualification concern of grating in the diesel generator room was appropriate. The assumptions used in the evaluation were appropriate to the expected seismic conditions specified in the plant design.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Unresolved Item 50-286/98007-04: Service Water Leak to 32 FCU

a. Inspection Scope (49001)

The inspector reviewed the licensee's service water improvement action plan and the circumstances surrounding the need for a notice of enforcement discretion on November 6, 1999.

b. Observations and Findings

On November 4, 1998, a system engineer discovered a service water leak on the outlet of the 32 fan cooler unit (FCU) between the containment penetration and the outlet isolation valve. NYPA requested and the NRC approved a notice of enforcement discretion (NOED) to extend the allowed outage time (AOT) for the fan cooler units as specified in technical specification 3.3.B.2.a. The AOT for the fan cooler units was extended for an additional 24 hours to facilitate completion of weld repairs. The weld was repaired and the FCU was restored to an operable condition.

Unresolved item (URI) 98007-04 was opened in NRC inspection report 50-286/98007 for evaluation of conditions related to the cause and corrective actions for the leak. This included a review of the root cause of the service water leak to determine if any regulatory requirements were not met.

The service water improvement action plan (IDSE-SIP-96-005) provided actions to identify and correct degrading service water piping conditions. Specifically, extensive replacement of small bore piping with corrosion resistant 6% Molybdenum stainless steel material, periodic visual examination (VT) of system components for leaks, nondestructive examination by radiography (RT) and ultrasonic testing (UT) of a sample of service water piping to determine the extent of active corrosion in the vicinity of pressure boundary welds.

The inspector walked down accessible parts of the service water system outside of the containment boundary to inspect the condition of the system, the location of radiographic and ultrasonic examinations, and to see the extent of replaced piping. The scope of VT, RT and UT from the past few years was reviewed as were the NYPA Generic Letter 89-13 Implementation Summary of November 1997, the Service Water Improvement Plan of August 1998, the Current Service Water System Corrosion Monitoring Program Implementing Procedure and related Action Tracking System items. The inspector concluded that NYPA had taken extensive actions to prevent service water leaks and to properly address those leaks that do occur. In addition the inspector concluded that there was no violation of NRC requirements associated with this service water piping leak and therefore URI 50-286/98007-04 is closed.

c. Conclusions

The scope of the licensee's service water action plan was extensive and appropriately addressed equipment performance issues associated with the service water system. Based on the service water system improvements made by the licensee, the circumstances surrounding need for the notice of enforcement discretion was determined not to be a violation of NRC requirements. Therefore, unresolved item 50-286/98007-04 is closed.

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls-External and Internal Exposure

a. Inspection Scope (83750-01)

The inspector evaluated the effectiveness of selected aspects of the applied radiological controls program. The evaluation included a selective review of the adequacy and implementation of the following radiological controls program elements:

- access controls to radiologically controlled areas
- use and adequacy of personnel occupational exposure monitoring devices
- maintenance of personnel occupational radiation exposures (external and internal) within applicable regulatory limits and as low as is reasonably achievable (ALARA)
- implementation of the radiation work permit program including the effectiveness of work planning
- oversight of the National Voluntary Laboratory Accreditation Program (NVLAP)-accredited personnel dosimetry program of the vendor providing whole-body TLDs
- operation and maintenance of a whole-body-counting program

The inspector evaluated performance in the above selected areas via observation of activities, tours of the radiologically controlled area (RCA), discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The licensee implemented effective access controls to the radiologically controlled areas of the station including use of radiation work permits (RWPs), bar code readers, and computerized log-in stations. No access control deficiencies were identified. A pre-job RWP briefing for a vapor containment (VC) entry at 100% power was conducted in a thorough manner.

Appropriate personnel monitoring devices for access to the RCA were supplied and used. Thermoluminescent dosimeters (TLDs) and personnel alarming dosimeters were observed to be properly worn to measure external dose. Access controls for high radiation areas (HRAs) were effective. Radiological postings and labels throughout the areas toured provided additional administrative controls and information to the workers. The radiological posting program was improved by using new signs which were all of the same size, placed at a consistent height and location, and used consistent, descriptive wording. Survey maps with radiological data were posted at the main health physics (HP) control point.

The licensee maintained personnel occupational radiation exposures (external and internal) within applicable regulatory limits and as low as is reasonably achievable (ALARA). A review of personnel exposure data for 1998 identified that individual exposure results for total effective dose equivalent (TEDE), lens of the eye dose equivalent (LDE), shallow-dose equivalent (SDE), and extremity dose equivalent were well below regulatory requirements. Further, the maximum individual committed effective dose equivalent (CEDE) for any one individual was well within applicable NRC limits, and, in fact, was less than the current licensee recording level of >10 millirem. The occupational exposure of declared pregnant women and the dose to the embryo/fetus were controlled in accordance with Title 10 Part 20.1208 of the Code of Federal Regulations (10 CFR 20.1208).

The licensee used TLDs provided by an outside organization, which operated and maintained a NVLAP-accredited personnel dosimetry program. The licensee had a copy of a NVLAP assessment of this program that was performed in the fourth quarter of 1998, which supported continued accreditation. Quarterly blind spiking of TLDs provided acceptable quality control results. The licensee operated and maintained a whole-body counter, which was currently calibrated for energy and efficiency in accordance with procedures. The respirator fit room facilities had been upgraded to include respirator cleaning capacity.

c. Conclusions

The licensee implemented effective applied radiological controls. Access controls to radiologically controlled areas were effective, and appropriate occupational exposure monitoring devices were provided and used. Personnel occupational exposure was maintained within applicable regulatory limits and as low as reasonably achievable (ALARA). The radiation work permit program was properly implemented.

R1.2 Radiological Controls-Radioactive Materials, Contamination, Surveys, and Monitoring

a. Inspection Scope (83750-01)

The inspector evaluated the effectiveness of the licensee's surveys, monitoring and control of radioactive materials and contamination. The evaluation include a selective review of the adequacy and effectiveness of the following radioactive material and contamination control program elements:

- surveys and monitoring of radioactive material and contamination
- the calibration status of survey and monitoring equipment
- the proper use of personal contamination monitors and friskers
- the tracking of personnel contamination events and goals

The inspector evaluated performance in the above selected areas via observation of activities, tours of the RCA, discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The licensee implemented an effective radioactive material and contamination control program. Continuous air monitors were calibrated and in use in the RCA. Hand-held contamination monitors (friskers) and radiation survey meters exhibited current calibration stickers and were appropriately used by personnel. Personnel were properly frisking at the RCA exit using whole body contamination monitors.

An HP technician was observed conducting routine periodic area surveys for radiation and contamination in a capable and effective manner. Survey records contained appropriate radiological information. Several radiation areas were now remotely monitored at a central HP control point using electronic dosimeters and wireless transmission. Radiological housekeeping conditions in the primary auxiliary (PAB), fuel storage (FSB), and radioactive machine shop (RAMS) buildings were good. Radioactive material and radioactive waste were clearly labeled, segregated, and stored in an orderly manner. Receptacles for used anti-contamination clothing, radiologically contaminated trash, and radiologically clean trash were available in the RCA and were clearly labeled. Additional RCA floor area was recovered as uncontaminated and painted to allow easier decontamination in the future.

Goals to assist in monitoring and tracking the effectiveness of personnel and area contaminations continued to be maintained and used to gauge the overall effectiveness of the station's programs.

c. Conclusions

The licensee implemented overall effective surveys, monitoring, and control of radioactive materials and contamination. HP technicians performed proper surveys and properly documented survey results. Radiological housekeeping conditions were noted to be good. The personnel contamination rate was tracked and trended against a goal. The radiological surveys, monitoring, and controls were implemented with calibrated and properly used devices.

R1.3 Radiological Controls-As Low As Is Reasonably Achievable (ALARA)

a. Inspection Scope (83750-01)

The inspector evaluated the effectiveness of the licensee's program to maintain occupational radiation exposure as low as is reasonably achievable. The evaluation included a selective review of the adequacy and effectiveness of the following ALARA program elements/documents:

- RWP dosage report for 1998
- 1998 dose savings initiatives
- RE-REA-4-1/12 Flowpath for RWP/ALARA reviews
- Spent Resin Transfer-Lessons Learned, 06/19/98
- Monthly ALARA Reports for 1998

- Post-Job ALARA Reviews for manipulation of the SP-952 valve on the 34 reactor coolant pump (RCP) platform at 100% power and for repair of the CH-342 valve leak in vapor containment at 2-4% power
- Selection as World Class ALARA Performer for 1998 for North America by the Nuclear Energy Agency of the International Atomic Energy Agency

The inspector evaluated performance in the above selected areas via observation of activities, tours of the RCA, discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

A Monthly ALARA Report was distributed to management of each department and included an analysis of person-rem performance on past-month and year-to-date bases for each department and for the station against their respective monthly and annual goals. These reports also included discussion of radiologically significant work during the past month, ALARA suggestions, number of RCA entries and contamination events, and a RWP dosage report for the past month and for year-to-date.

Other ALARA efforts included the replacement of a long-standing temporary lead blanket shielding arrangement for a primary system letdown line on the 41-foot level of the PAB with a permanent shield wall, resulting in the reduction of the general area dose rate by a factor of three. The ALARA awareness and suggestion programs received greater attention and procedural improvements.

The person-rem goal for 1998 was <22. The year included 343 days of operation, and the remaining days were spent in several forced outages. The post-Job ALARA Reviews for the manipulation of SP-952 on the 34 RCP platform at 100% power and for repair of the CH-342 leak in vapor containment at 2-4% power demonstrated a strong commitment to dose savings. The ALARA functions were incorporated into the routine task requirements, and trigger points for pre- and post-job evaluations were made more conservative. Through these and other efforts to achieve dose savings, the person-rem for 1998 was held to 14.756. The dose-saving efforts and results were indicative of a strong ALARA performance.

Annual goals for person-rem and personnel skin contaminations for 1999 were established. A refueling outage including 10-year in-service-inspection (ISI) work was scheduled for 1999. The annual person-rem goal was determined based on achieving an overall improvement in dose savings over past results for the tasks identified in the preliminary outage scope.

c. Conclusions

The licensee implemented a very effective program to maintain occupational radiation exposure as low as is reasonably achievable, and the ALARA efforts and results for 1998 were exceptionally good, including the management of radiologically significant work and a station record for lowest annual person-rem.

R7 Quality Assurance in RP&C Activities

a. Inspection Scope (83750-01)

The inspector evaluated the effectiveness of the licensee's self-identification and corrective action processes. The evaluation include a selective review of the adequacy and effectiveness of the following program elements and documents:

- Audit Report A98-19-I by the Quality Assurance Division
- self-assessments by the radiation protection organization
- corrective action program

The inspector evaluated the performance in the above area via observation of activities, tours of the RCA, discussions with cognizant personnel, review of applicable documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The scope, depth, and level of detail of Audit Report A98-19-I by Quality Assurance Division was extensive and reviewed the program for strengths and weaknesses. The audit team utilized two subject matter experts from outside organizations. The audit found that the program was implemented satisfactorily with respect to compliance with NRC regulations. The audit report identified eight deficiencies, of minor safety significance involving inconsistent practices and procedures and incomplete records. These deficiencies met the licensee's threshold for entry as deviation event reports (DERs) into the corrective action program. The audit report also noted seventeen improvement items, separate from the DERs, which were entered as items into the action commitment tracking system (ACTS). The identified corrective actions and due dates for these items were appropriate.

Self-assessments by the radiation protection organization addressed radiation detection instrumentation, the respiratory protection program (two improvement items identified), and record keeping for decommissioning (two improvement items identified). An annual review of the radiation protection self-assessment program identified that timely completion of scheduled self-assessments and use of proper procedure to close related ACTS items required improvement. In response, an ACTS item was generated to revise the controlling procedure for these self-assessments to clarify both the process for closing related ACTS items and responsibility for identifying self-assessment topics, their assignment, and the tracking of progress and completion. Overall, the self-assessments evidenced generally effective efforts to identify procedural compliance, strengths, and weaknesses.

c. Conclusions

The licensee's self-identification and corrective action processes in the area of radiation protection were effective. A quality assurance (QA) audit, self-assessments, and the corrective action program continued to be effective in identifying, at a low threshold, deficiencies and improvement opportunities. Effective corrective actions were implemented for findings.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

Region based inspectors presented inspection findings in the area of radiological controls to members of the licensee's management on January 29, 1999, and in the area of engineering on February 5, 1999. The resident inspectors presented the integrated inspection results to members of the licensee's management at the conclusion of the inspection on February 11, 1999. The licensee acknowledged the findings presented.

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

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Barrett, Site Executive Officer
F. Dacimo, Plant Manager
J. Comiotes, General Manager-Operations
D. Quinn, General Manager, Support Services
J. Russell, General Manager-Maintenance
G. Mavrikus, Director, IP3 Engineering
K. Peters, Manager, Licensing

INSPECTION PROCEDURES USED

IP 37550:	Engineering
IP 49001:	Inspection of Erosion/Corrosion Monitoring Programs
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 83750:	Occupational Radiation Exposure
IP 92700:	Event Reports
IP 92901:	Followup - Operations
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NCV	98010-01	Backup Nitrogen to the Control Room Ventilation System
NCV	98010-02	Containment Isolation Valve Leak Testing
EEI	98010-03	Containment Vent Design Deficiency

Closed

NCV	98010-01	Backup Nitrogen to the Control Room Ventilation System
NCV	98010-02	Containment Isolation Valve Leak Testing
URI	98007-04	Service Water Leak to 32 FCU

LIST OF ACRONYMS USED

ACTS	action commitment tracking system
ALARA	as low as is reasonably achievable
AOT	allowed outage time
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CVCS	chemical and volume control system
DER	deviation event report
ECN	engineering change notice
EDG	emergency diesel generator
FCU	fan cooler unit
FSB	fuel storage building
HP	health physics
HRA	high radiation area
I&C	instrument and controls
ISI	in-service inspection
IVSW	isolation valve seal water
LCO	limiting condition for operability
LDE	lens of the eye dose equivalent
NCV	non-cited violation
NPO	nuclear plant operator
NVLAP	National Voluntary Laboratory Accreditation Program
PAB	primary auxiliary building
PM	preventive maintenance
QA	quality assurance
RAMS	radioactive machine shop
RCA	radiologically controlled area
RCP	reactor coolant pump
REO	reasonable expectation of operability
RP&C	radiological protection and chemistry
RT	radiograph testing
RWP	radiation work permit
SDE	shallow dose equivalent
TEDE	total effective dose equivalent
TLD	thermoluminescent dosimeter
TM	temporary modification
TPC	temporary procedure change
URI	unresolved item
UT	ultrasonic testing
VC	Vapor Containment
VT	visual test
WCCPPS	weld channel and containment penetration pressurization system