U.S.NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-213/90-15

License No. DPR-61

Licensee: Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, CT 06141-0270

Facility: Haddam Neck Plant

Location: Haddam Neck, Connecticut

Inspection Dates: September 12 to October 31, 1990

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Inspection Summary: Inspection on September 12, to October 31, 1990
(Inspection Report No. 50-213/90-15)

Areas Inspected: Routine safety inspection by the resident inspectors. Operational areas reviewed included the reactor trip on September 20; auxiliary feedwater pump steam binding; and plant operations with steam generator primary-to-secondary leakage, a failed nuclear instrumentation channel, an observed difference between reactor coolant system loops differential temperatures, and a leak in the refueling water storage tank. Maintenance activities were reviewed including auxiliary feedwater pump repairs, nuclear instrumentation replacement, and steam generator level controller repairs. Surveillance observations included monitoring of the steam generator leakage, reactor fuel clad failures, containment air recirculation fan cooler performance and plant freeze protection systems. Security system modifications and response to employee fitness for duty concerns were reviewed. Engineering evaluations of loose parts in the feedwater system, pressurizer level indication anomalies, auxiliary feedwater instrumentation, and emergency diesel generator performance were also reviewed.

Results: See Executive Summary

EXECUTIVE SUMMARY

Haddam Neck Plant

NRC Region I Inspection No. 50-213/90-15

Plant Operations

Failure to follow procedures when isolating a condensate pump resulted in a manual reactor trip on September 20. This was licensee identified.

Operators responded quickly in taking manual control of the No. 3 feedwater regulating valve when it went full open on October 3. Likewise, in an incident on October 10, operator action in response to steam binding of an auxiliary feedwater pump was in conformance with NRC Generic Letter 88-03 and NRC Bulletin 85-01.

The licensee's self-assessment indicated that poor judgement was made in maintaining reactor power operation after opening of the service water filter bypass valves associated with the supply to the containment air recirculation (CAR) fan coolers. These valves were installed for use in post-accident situations in the event that the service water filters become blocked. An error was also made in opening the CAR service water outlet throttle valves without an engineering basis. The reactor was shutdown when cooler flow performance degraded. These issues were licensee identified.

Radiological Controls

Good radiological controls were observed. Particularly in the controls for steam generator work and also in monitoring and containment of the refueling water storage tank leakage.

Maintenance and Surveillance

Good performance was noted in the discovery of freeze protection deficiencies concerning outdoor safeguards system piping. Licensee review of the problem led to improvements in temperature monitoring.

Strong programs were observed in monitoring primary to secondary steam generator leakage and in the monitoring of reactor fuel clad failure.

An unqualified component was installed in the No. 3 steam generator level controller without initiating a nonconformance report and performing an engineering evaluation of the affect it may have had on the reactor protection system. This issue was licensee identified.

The surveillance of CAR fan coolers for service water hydraulic performance is considered to be a significant improvement in the ability to assess emergency safeguards equipment performance.

EXECUTIVE SUMMARY

Haddam Neck Plant

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Security and Safeguards

Replacement of security system computers and its associated process communication equipment was noted to be a good initiative. However, an issue concerning the failure to carry out for-cause testing revealed inadequacies in supervisor training. This is addressed further in NRC Inspection Reports 50-213/90-17 and 90-18 along with information concerning an enforcement conference contained in a letter from the NRC Region I dated November 6, 1990.

Engineering and Technical Support

Good performance was noted in the engineering analysis and associated setpoint changes associated with the observed differences in indicated pressurizer level. Thorough engineering assessments were made of the effects of loose parts which were remnants from a failed feedwater regulating valve.

Safety Assessment and Quality Verification

The licensee has responded much more deliberately to the emergency diesel generator starting problems during this inspection period than the response noted in the previous resident inspection.

Self-assessment made by the licensee following the above referenced issues relating to plant operations have resulted in changes to administrative controls and operator training. Additionally, following the auxiliary feedwater pump steam binding incident of October 10, licensee evaluation led to a temperature alarm setpoint change.

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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. Summary of Facility Activities

The reactor was operating at 80% power at the beginning of the inspection period. Repairs had been completed to the No. 1 steam generator feedwater regulating valve following its stem failure on September 3; based on the results of the failure analysis, all four regulating valve stem and disc assemblies were modified and nondestructively examined. However, the valve failure had resulted in metallic debris entering the feedwater system. This material, which was not located during inspections, was assumed to have entered the No. 1 steam generator through its feedwater ring. A technical evaluation of the potential effects of this debris justified the return to power operation for one week pending completion of the final assessment. That assessment, which justified completion of the operating cycle, was reviewed and accepted by the Plant Operations Review Committee on September 13.

During power ascension, plant maximum power was found to be limited to approximately 87% due to instabilities in the performance of the feedwater and condensate system. At this power level, feedwater pump suction pressure became unusually low and the condensate pump motor current was observed to oscillate. The reactor was manually tripped in response to an automatic low suction pressure trip of the two main feedwater pumps on September 20. Prior to the trip, reactor power was at 50% and one of the two condensate pumps was being isolated for investigation of the performance problem.

The reactor remained shutdown and the licensee started a planned maintenance outage earlier than expected. Major work included; replacement of the lower power range detector of nuclear instrumentation system channel No. 1; repair of the "B" auxiliary feedwater pump, which experienced a shaft seal failure on September 18; investigation into the deviation between pressurizer level channels; and, replacement of the "B" condensate pump along with the investigation into the system flow capacity problem.

The condensate flow problems were found to be caused by the delamination of the inside surface of rubber expansion joints at the condensate pump suctions. Apparently, suction flow area became increasingly restricted by the collapse of the rubber boot as system flow demand increased. The rubber boots were replaced with stainless steel expansion devices.

The reactor was made critical at 3:31 a.m. on September 25 and reached full power on September 27. While at full power, on October 3 at 9:15 a.m. a failure occurred in the No. 3 steam generator level controller causing the corresponding feedwater regulating valve to open fully. Operators responded, placed the valve in manual control and restored steam generator level. The maximum level was 76% of the narrow range.

The refueling water storage tank (RWST) was found to be leaking through its bottom plate on September 14 at an approximate rate of five gallons per day. Licensee surveillance of the tank continued throughout the inspection period; the leak rate remained constant and in the range of five to eight gallons per day.

Primary to secondary leakage through the No. 2 steam generator was observed to increase following the return to power in September. It had increased to about 68 gallons per day (gpd) through October. The leakage rate had been approximately 30 gpd prior to the September 20 reactor trip. Leakage rates were calculated every four (4) hours based on radiochemical analysis. Additionally, instrumentation was installed on October 18 to monitor nitrogen-16 gamma decay in the No. 2 main steam line to continuously calculate leak rate. The leak rate was trended and planning started for an outage to make repairs.

Steam binding occurred in the "B" auxiliary feedwater pump on October 10 due to back leakage through its discharge check valve and the flow injection check valves to the No. 2 and 4 feed water lines. Operator actions were prompt in returning the pump to service and seating the check valves.

The calibration of auxiliary feedwater flow instruments was found in error on October 12 in that the water temperature used for the calibration point was 450 degrees F (normal feedwater temperature) instead of 80 degrees F (nominal auxiliary feed water temperature). The density difference resulted in a calculated nine gallon per minute (gpm) error at the minimum required flow of 80 gpm per steam generator. The qualification of the auxiliary feedwater flow instruments to operate in the harsh environment following a postulated main feedwater line break has also been under review.

The reactor remained at full power until October 27 when a shutdown was required by Technical Specifications 3.0.3 and 3.6.2. The performance of the containment air recirculation (CAR) fan coolers was observed to degrade due to an accumulation of foreign material within the coolers. This material was carried through the service water system after the intake structure trash rakes were cleaned using high pressure water. Service water filters to the CAR fan coolers became plugged and were bypassed when it became increasingly difficult to maintain flow. Because the filter bypass valve was opened during a period of heavy river debris, surveillance of cooler performance was started. The No. 3 CAR fan was declared inoperable because of excessive hydraulic resistance at midnight on October 26; and, the No. 4 CAR fan declared inoperable at 2:20 p.m. October 27. The reactor was cooled to Operational Mode 5 at 3:00 p.m. October 28 and the licensee began an outage to clean the cooler coils and to repair the leaking steam generator tube. The reactor remained shutdown through the end of the inspection period.

NRC Commissioner James R. Curtis visited the site on October 12 to conduct discussions with licensee management and to make an inspection of the facility.

A delegation of visitors from the Soviet Union met with the licensee during the week of October 15 to discuss the symptom based emergency operating procedures. Plant tours and simulator demonstrations were made.

The inspection ac ivities during this report period included 118 hours of inspection during normal utility working hours. In addition to normal utility working hours, the review of plant operations was routinely conducted during portions of backshifts (evening shifts) and deep backshifts (weekend and night shifts). Inspection coverage was provided for 51 hours during backshifts and 26 hours during deep backshifts.

2. Plant Operations

2.1 Operational Safety Ve ification

The inspectors observed plant operation and verified that the plant was operated safely and in accordance with licensee procedures and regulatory requirements. Regular tours were conducted of the following plant areas:

-- control room
-- primary auxiliary building
-- radiological control point
-- electrical switchgear rooms
-- auxiliary feedwater pump room
-- security access point
-- protected area fence
-- intake structure
-- diesel generator rooms
-- turbine building

There were no significant observations noted.

2.2 Engineered Safety Features System Walkdown

In addition to routine observations made during regular plant tours, the inspectors conducted walkdowns of the accessible portions of selected safety related systems. The inspectors verified system operability through reviews of valve lineups, control room system prints, equipment conditions, instrument calibrations, surveillance test frequencies and results, and control room indications. During this inspection period, walkdowns of the following systems were performed:

-- Auxiliary Feedwater System

-- Emergency Diesel System

-- Portions of the Low Pressure Safety Injection System

-- Portions of the High Pressure Safety Injection System

-- Containment Air Cooling System

There were no significant observations noted.

2.3 Follow-up of Events Occurring During Inspection Period

During the inspection period, the inspectors provided on-site coverage and follow-up of unplanned events. Plant conditions, alignment of safety systems, and licensee actions were reviewed. The inspectors confirmed that required notifications were made to the NRC. During event follow-up, the inspectors reviewed the corresponding plant information report (PIR) package, including the event details, root cause analysis, and corrective actions taken to prevent recurrence.

2.3.1 Reactor Trip on September 20

A manual reactor trip was performed by control board operators on September 20 at 7:20 a.m. in response to the automatic trip of both main feedwater pumps because of low pump suction pressure. This occurred as the discharge valve of the "B" condensate pump was shut to obtain a value for pump discharge pressure.

The licensee had been investigating an apparent loss of condensate pump capacity and suspected that the "B" pump had degraded. Power was found to be limited to about 87% of full power with two condensate, two main feedwater and one heater drain pump running. This was the normal full power alignment.

During the turbine plant startup following the 1990-91 refueling outage, the plant operators discovered a need for the second condensate pump at a lower power level than normally needed, and also observed feedwater pump suction pressure begin to drop off at about 87% power. Oscillation of main control board ampere meters indicated the pumps alternately attempting to pick up the supply of condensate water flow.

In in effort to verify pump performance curve values prior to a plant shutdown, reactor power was reduced to 50% of full power. A plant shutdown was in progress to allow for condensate pump replacement.

As the discharge valve of the "B" condensate pump was shut fully, "A" pump discharge pressure dropped dramatically from 400 psig to less than 200 psig. At this point, the steam generator main feedwater pumps tripped on low suction pressure. These pumps will automatically restart if suction pressure is restored pressure was not restored. The manual trip was performed in anticipation of low steam generator water level.

Emergency safeguards equipment, including auxiliary feedwater, performed properly following the feedwater pump trip and the manual reactor trip.

The licensee's post trip evaluation identified two significant concerns. First, Normal Operating Procedure NOP 2.2-1, Changing Plant Load, Section 6.2.12.b allows removing the second condensate pump from service between 250 and 230 Mw(e) (or approximately between 41% and 38% reactor power). Shutting the "B" condensate pump discharge valve at 50% reactor power therefore violated that procedure. Secondly, because one of the two auxiliary feedwater pumps was out of service and the feedwater/condensate system was known to have an abnormal condition affecting its flow capacity, the manipulation of the pump discharge valve at this power was not prudent. The inspector concurs with the licensee's evaluation.

The licensee's investigation into the condensate system following the trip revealed that the inside surface of both pump suction flexible couplings had delaminated. Those rubber couplings were replaced during the refueling outage and were just placed into initial service. The "B" condensate pump was also replaced (with a spare) during the outage.

The rubber couplings were replaced with solid stainless devices; also, the "B" condensate pump was replaced with a recently rebuilt spare. The condensate pumps were flow tested and found to meet their head/flow performance curve prior to placing the turbine plant in service following the trip.

The inspector concurred with the licensee evaluation of the incident referenced above; there were no other significant observations. Although this incident involved a procedure violation, no Notice of Violation is being issued in accordance with the provisions of 10 CFR Part 2, Appendix C, Section V.G.1, Exercise of Discretion (NC 90-15-02).

2.3.2 Primary to Secondary Steam Generator Leakage

Primary to secondary leakage was observed in the No. 2 Steam Generator though the inspection period. The leak rate, which is calculated by using the results of radiochemical analysis, increased following the plant return to power in September. Where it had been approximately 30 gallons per day (gpd) prior to the September 20 manual reactor trip, it had increased to about 68 gpd just prior to a reactor shutdown required by technical specifications for reasons of containment air recirculation fan cooler operability on October 27. The technical specification for leakage through an individual steam generator is 150 gpd.

During this period of reactor operation, the calculations of leak rate were made based on radiochemical analysis made about every four (4) hours. The data was plotted and the rate of increase (or decrease) was projected in time, based on a least-squares fit of datapoints.

Instrumentation was installed and placed in service on October 18 to monitor nitrogen-16 (N-16) gamma decay in the No. 2 main steam line. That device was calibrated to display a calculated leakage rate and was beneficial because of its continuous on-line display of that value. Correlation was observed between this displayed value of leakage and that calculated by radiochemical analysis every four (4) hours.

The licensee had been planning to make repairs to the steam generator and commenced that evolution following the October 27 reactor shutdown. Just prior to that shutdown, the leakage rate projections to 100 gpd and 1'0 gpd were February 1 and April 1, 1991, respectively.

The inspector followed the licensee's monitoring and trending programs throughout the inspection period. The calculations and trending performed using surveillance procedure SUR 5.4-44 met the licensee's response to NRC Bulletin 88-02, dated March 24, 1988. The installation of the N-16 monitor was a good initiative, and the licensee's decision to perform steam generator inspection and repair was conservative.

2.3.3 Steam Binding of the "B" Auxiliary Feed Water Pump

The "B" auxiliary feedwater (AFW) pump became steam bound on October 10 due to back leakage of three system check valves. At 9:30 a.m. the auxiliary operator reported that the "B" AFW pump casing was very hot. Venting of the pump casing produced steam and AFW header temperature was 183 degrees F. The "B" AFW pump was declared inoperable and technical specification (TS) action statement 3.7.1.2.a. was entered. The "A" AFW pump was satisfactorily started to verify operability.

Venting of the "B" AFW pump continued until 10:00 a.m. at which time the pump was started and its ability to attain 1000 psig discharge pressure was verified. The pump was declared operable and the TS action statement exited.

In accordance with ANN 4.13-21, "Auxiliary Feedwater Header High Temperature", operations personnel inspected the AFW discharge piping to the feedwater (FW) system. FW-CV-156-2 and FW-CV-156-4 in the normal AFW discharge path to the FW system were identified as stuck open, causing the AFW discharge lines to heat up. The piping adjacent to these valves was measured at 300 degrees F. This, coupled with back leakage of the "B" AFW pump discharge check valve (FW-CV-153), caused the "B" AFW pump to steam bind. The "A" AFW pump discharge check valve did not similarly leak back and the pump remained operable.

The FW check valves were isolated and the "B" AFW pump was run (as directed by ANN 4.13-21) throughout the day to cool down the affected piping. At 8:20 p.m. on October 10 the check valves had been reseated, the AFW discharge lines satisfactorily cooled, and the normal system lineup restored.

In response to this event the licensee initiated an AWO to inspect the "B" AFW pump discharge check valve during a plant outage to determine the cause of its back leakage. Additionally, operators continued to monitor AFW discharge piping temperature during routine tours.

The "Aux. Feed Header High Temperature" annunciator in the control room had not alarmed to alert operators to the high temperature condition. The alarm setpoint for this annunciator is 195 degrees F. This event demonstrated that the AFW pumps can become steam bound at piping temperatures less than this setpoint. Setpoint change request 90-P-0037 was initiated and completed under AWO 90-10513 on October 23. Under this work order, the calibration of temperature indicator TIA-1300 was verified to be correct and the alarm setpoint was lowered to 160 degrees F.

The inspector observed the verification of the "A" AFW pump operability and activities involved with venting the "B" AFW pump and seating the affected check valves. Operators quickly responded to the event and took appropriate immediate corrective actions in accordance with station procedures to return the AFW system to service. The inspector confirmed that the response to steam binding incident was in conformance with NRC Generic Letter 88-03 and NRC Bulletin 85-01. The change of annunciator temperature setpoint was a reasonable conservative action and promptly implemented.

2.3.4 Feedwater Flow Excursion No. 3 Steam Generator

On October 3, a feedwater flow excursion occurred in the No. 3 steam generator when the associated feedwater regulating valve (FRV) automatically opened fully. Operators managed to stop the transient before the steam generator level exceeded 76% by placing the FRV in manual control. Normal steam generator levels were recovered.

Troubleshooting activities and repair of the No. 3 steam generator level controller are discussed in section 4.1.4 of this report.

2.3.5 Reduced Reactor Coolant System Differential Temperature

During this operating cycle the licensee has noted that the reactor coolant system (RCS) loops differential temperatures (delta-T) are lower than observed during previous operating cycles. This difference is attributed to the increased RCS flow rates resulting from the removal of the reactor vessel thermal shield.

The delta-T values are a factor in the variable low pressure trip (VLPT) equation. This equation is currently valid only with a delta-T of 45 degrees F. Operators have observed that RCS loop 4 delta-T varies between 44 and 45 degrees F.

On September 28, while this condition was under evaluation a trip signal was inserted into the reactor protection system for RCS loop 4 VLPT.

Corporate engineers conducted a review of the VLPT equation and determined that the minimum acceptable delta-T with the current VLPT equation is 44 degrees F. A proposed Technical Specification change is being prepared to change the delta-T multiplier in the VLPT equation to account for the observed changes and provide additional safety margin.

The Plant Operations Review Committee (PORC) met on September 28 and October 2 to discuss this situation. It was concluded that continued operation is acceptable and the trip signal on RCS loop 4 VLPT was removed.

The inspector attended the PORC meeting on October 2. The calculations performed in determining the acceptability of this reduced delta-T were reviewed with NRC Headquarters personnel. No unacceptable conditions were identified.

2.3.6 Containment Air Recirculation Fan Cooler Operability

The containment air recirculation (CAR) fan coolers are required for post accident containment pressure control. Heat removal from the air coolers is made through service water cooling. The service water is processed through pump discharge strainers and self cleaning filters prior to entering the CAR fan coolers and fan motor coolers.

The plant experienced a period of heavy debris in the Connecticut River during the week of October 21. The river provides the source of service water. This condition requires that the plant intake trash rakes be periodically cleaned, an evolution which is performed by divers.

The plant service water system experienced an increase of debris in the CAR fan cooler self cleaning filters following trash rake cleaning on October 26. The debris loading was so severe that the filters required disassembly and manual cleaning.

There are two (2) self cleaning filters, both of which supply all four (4) CAR fan cooler units.

A filter bypass valve was opened by shift personnel at 3:05 p.m. on October 26 in response to continual filter clogging. The valve was shut after a filter was returned to service. However, bypass valves were reopened several additional times during the evening of October 26 and the early morning of October 27.

With the filter bypass valve open debris may foul the internals of the CAR fan coolers and fan motor coolers. Because the coolers are required to be operable for post accident containment cooling and pressure control, a surveillance on the coolers was started to monitor their hydraulic resistance. The surveillance, performed by procedure SUR 5.7-118, had established acceptance criteria for each of the four (4) cooler units.

As the surveillance was performed, personnel trended cooler performance and declared No. 3 CAR fan cooler out of service at 11:30 p.m. on October 26; and, No. 4 CAR fan cooler out of service at 2:40 p.m. on October 27.

Technical specification 3.6.2 requires all four (4) CAR fan cooler units to be operable in Operational Modes 1 through 4. Seventy two hours is allowed to return one inoperable cooler to service; two inoperable coolers cause a reactor shutdown and cooldown in accordance with technical specification 3.0.3 to cold shutdown, Mode 5 within specified time increments.

A reactor shutdown from full power was commenced at 3:00 p.m. October 27; the reactor was shutdown at 8:45 p.m. and was cooled to less than 200 degrees F, Mode 5, at 4:00 p.m. on October 28. The plant entered an outage to allow cleaning of the coolers and also to inspect steam generators and repair a leaking tube.

The licensee's evaluation of this event identified a lapse in conservative action by remaining at full power with the filter bypass valves open. Additionally, that it was also incorrect to reposition the service water return valves from the CAR fan coolers. These valves were placed in a throttle position which was determined through a special test procedure to insure adequate flow to all safeguards equipment cooled by service water.

Those valves had been opened fully at 3:14 p.m. for about 36 minutes after the filter bypass valve was opened. Shift personnel had also decided this was required to maintain cooler flow after opening the filter bypass valve.

The inspector followed the licensee actions, evaluation and corrective actions which included additional instruction to operators in the form of additional training, procedure enhancements and improvements to posted signs. The coolers were disassembled and cleaned, and the service water system performance was verified through a full flow test, SUR 5.7-148B, a special test to reset the throttle valves, ST 11.7-7; and measuring the cooler hydraulic resistance, SUR 5.7-118.

The inspector concurs with the licensee evaluation of the event and their operator's actions. There were no additional significant observations.

2.3.7 Interruption of Spent Fuel Storage Pool Cooling Flow

The licensee removed the spent fuel pool storage pool cooling system from service on October 23 to allow the replacement of a pump shaft seal. The "B" pump seal required replacement and the system has no means of individual pump isolation

Prior to maintenance the pool temperature was monitored and heatup rate calculated. The pool temperature was recorded periodically during the maintenance action. The NRC duty officer and resident inspector were informed prior to securing the system.

The inspector monitored the licensee's actions; there were no significant observations noted.

3. Radiological Controls

During routine inspections of the accessible plant areas, the inspectors observed the implementation of selected portions of the licensee's radiological controls program. Utilization and compliance with radiation work permits (RWPs) was reviewed to ensure that detailed descriptions of radiological conditions were provided and that personnel adhered to RWP requirements. The inspectors observed controls of access to various radiologically controlled areas and the use of personnel monitors and frisking methods upon exit from those areas. Posting and control of radiation areas, contaminated areas and hot spots, and labelling and control of containers holding radioactive materials were verified to be in accordance with licensee procedures. During this inspection period, radiological controls for the following activities were observed:

-- Inspection and Repair of Leaking Tube in the No. 2 Steam Generator

-- Inspection and Repair of Leaking Steam Generator Primary Manways

-- Replacement of the Nuclear Instrumentation System Channel No. 1 Lower Power Range Detector Fission Chamber

-- Replacement of "B" Charging Pump Outboard Bearing

Health physics technician control and monitoring of these activities were determined to be adequate.

3.1 Radiological Assessment of Refueling Water Storage Tank Leak

On September 14, 1990, leakage of the refueling water storage tank (RWST) was discovered. The engineering aspects are described in Section 6.3 of this report. The licensee's radiological assessment of this leakage was reviewed by a NRC Region I specialist inspector. The inspector toured the area of the RWST and verified the licensee's effort to prevent leakage into the storm drain system (yard sewer system). It appeared that all leakage was contained in the concrete dike around the RWST. The inspector also noted that there was no contact between the leakage and soil around the RWST. The leak rate was determined to be about 5 gallons per day as of September 24, 1990.

The inspector reviewed analytical techniques and weekly analytical results of the RWST grab samples for gamma emitters and tritium. The total gamma and tritium activities in the RWST were 9.57E-3 micro Ci/cc and 7.28E-3 micro Ci/cc, respectively. The inspector had no further questions about the measurement techniques and analytical results for gamma and tritium analyses.

The inspector also reviewed the licensee's hypothetical radiological dose assessment results based on the worst case (assuming a complete drain of the RWST into the environment over a six hour period). The licensee performed a dose assessment using the Off-site Dose Calculation Manual methodology. The results of the hypothetical radiological dose assessments for the whole body and organ doses were 5 mRem and 5.75 mRem, respectively. The dose limits of the Technical Specifications for the whole body and organ doses are 1.5 mRem/quarter (3.0 mRem/year) and 5 mRem/quarter (10 mRem/year), respectively. However, it should be noted that the dose assessment results were based on the worst case, and there was no release to the environment. The inspector had no further questions in this area.

The licensee has planned the following actions in the event of any release to the environment.

- Continue monitoring of the leak progression pathway to ensure RWST liquid is not discharged to the yard sewer system.

- Monitor #4 and #5 yard sewers daily as a check against discharge. These two sewers are downstream of the RWST yard sewer (#3).

- Dose calculations will be done for any RWST water that is released to the environment through the liquid pathway.

Based on the review of the licensee's measurement technique analytical results, and actions, the inspector determined t following.

- The licensee has the capability to accurately measure gamma emitters and tritium in the RWST.

- The licensee's actions to monitor leakage to the environment were good.

- There were no negative impacts on the environment and/or the public health and safety as a result of this event.

- The licensee has the capability to perform the necessary radiological dose assessment.

4. Maintenance and Surveillance

4.1 Maintenance Observation

The inspectors observed various corrective and preventive maintenance activities for compliance with procedures, plant technical specifications, and applicable codes and standards. The inspectors also verified the appropriate quality services division (QSD) involvement, use of safety tags, equipment alignment and use of jumpers, radiological and fire prevention controls, personnel qualifications, post-maintenance testing, and reportability. Portions of activities that were reviewed included:

-- Replacement of the Nuclear Instrumentation System Channel No. 1 Lower Power Range Detector Fission Chamber

-- Inspection and Repair of Leaking Tube in the No. 2 Steam Generator

-- Inspection and Repair of Leaking Steam Generator Primary Manways

-- Rebuilding of the Spare Condensate Pump

-- Repair of Pressurizer Level Instrument Tubing Leak

-- Repair of "B" Auxiliary Feedwater Pump Turbine Following Shaft Seal Failure and Subsequent Bearing Failure

-- Inspection of Condenser Hotwell and Condensate Pump Suction Lines

-- Replacement of Condensate Pump Suction Line Flexible Couplings

-- Replacement of the "B" Condensate Pump

-- Repair of No. 3 Steam Generator Level Control System

-- Replacement of Auxiliary (Heating) Steam Supply Line to the Control Room Air Handling Unit

-- Repair of Reactor Vessel Loose Parts Monitor Following Lightning Damage

-- Repair of Leak "B" Main Feedwater Pump Discharge Flange

-- Repair of No. 1 Steam Generator Level Control System

-- Replacement of "B" Spent Fuel Storage Pool Cooling Pump Shaft Seal

-- Repair of Safeguards System Outdoor Piping Freeze Protection

-- Cleaning of Service Water System Filters for Supply to the Containment Air Recirculation (CAR) Fan Coolers, Fan Motor

Coolers and Spent Fuel Storage Pool Cooling Heat Exchangers

-- Cleaning of the CAR Fan Coolers and Fan Motor Coolers

-- Repair of Containment Isolation Trip Valve CC-TV-920

-- Replacement of "B" Charging Pump Outboard Bearing

-- Modifications to Auxiliary Feedwater Flow Instruments for Protection Against Harsh Environment

There were no significant observations noted.

4.1.1 Nuclear Instrumentation Fission Chamber Replacement

A failed fission chamber associated with the nuclear instrumentation system (NIS) channel No. 1 lower detector was replaced. Time domain reflectometry performed during troubleshooting of the instrument located a fault at the location of a factory splice in the detector cable near the detector. The failure occurred shortly after commencing reactor power operations with the newly installed NIS.

The inspector followed the licensee actions in the investigation and repair of the problem. The licensee complied with the provisions of technical specification 4.2.4.3 for reactor core quadrant power tilt ratio monitoring with the detector out of service.

4.1.2 Repair of the "B" Auxiliary Feedwater Pump

The "B" auxiliary feedwater (AFW) pump was declared inoperable on September 18 at 8:50 a.m. During performance of SUR 5.1-13B, "Auxiliary Feed Pump (P-32-1B) Functional Test," operator observed that the oil in the turbine lubricating oil system was milky and water was spraying from the turbine outboard bearing. The "A" AFW pump was verified to be operable.

Technical Specifications (TS) 3.7.1.2 requires that both AFW pumps be operable with the plant operating. The associated action statement allows 72 hours of continued plant operation while repairs are made to the inoperable pump.

The inspectors observed troubleshooting and repair activities under AWO 90-9684. The pump turbine was disassembled and inspected; a broken spring was found in the carbon seal assembly and the shaft was scored. The turbine was overhauled in accordance with CMP 8.5-184, "Overhaul of Auxiliary Feed Pump Turbine". This included replacement of the shaft, bearing, and seal housing. A realignment was performed and the governor and overspeed trip were adjusted.

The "B" AFW pump was reassembled and successfully tested in accordance with SUR 5.1-13B on September 21. The pump was declared operable at 4:07 a.m. and the TS action statement exited prior to elapse of the 72 hours.

The inspectors observed the maintenance pre-job briefing, tagout verification, job staging, and portions of the repair, reassembly, and testing. Good communications and coordination between operations, maintenance, and engineering personnel were noted. Procedures were determined to be adequate and procedural compliance was good.

4.1.3 Restoration of High Energy Line Break Jet Impingement Shield

The licensee discovered bolts missing from a jet impingement shield surrounding an auxiliary (heating) steam pipe in the service building cable spreading area. The licensee effectively dealt with the analysis of the pipe and shield and took prompt corrective action to restore the device.

The inspector monitored the licensee's actions; there were no significant observations noted.

4.1.4. Troubleshooting and Repair of No. 3 Steam Generator Level Controller

As described in section 2.3.5 of this report, a feedwater flow excursion occurred on October 3 in the No. 3 steam generator (SG) when the feedwater regulating valve (FRV) opened fully.

Instrumentation and Controls (I&C) personnel conducted troubleshooting activities under AWO 90-10001. The No. 3 SG level and flow controllers were inspected. The level controller level setpoint was observed to oscillate. This oscillation was attributed to a defective input filter capacitor. The input and output filter capacitors are categorized as Quality Assurance (QA) category 1. A suitable input filter capacitor was not available; a non-QA capacitor was installed under AWO 90-10212. The output filter capacitor was also replaced at this time. The No. 3 SG level controller was returned to service and the No. 3 FRV was placed in automatic control at 3:03 p.m. on October 3.

On October 3, I&C personnel initiated an upgrade of the input filter capacitor in accordance with ACP 1.2-4.2, "Commercial Grade Procurement, Upgrade and Dedication Process."

The following day, station engineers determined that an unevaluated condition existed with the SG level controller in service containing an unqualified component. This condition was of concern because the SG level control circuitry is not isolated from the reactor protection system (RPS). It was unknown what effect a failure of the input filter capacitor could have on the RPS. An evaluation of this condition should have been conducted through the Nonconformance Report (NCR) process prior to placing the SG level controller in service. This concern was brought to station management attention on October 4.

Technical Specification Table 3.3-1 requires that a trip signal be inserted into the RPS within one hour of declaring a channel of SG level inoperable. A RPS trip signal was inserted for the No. 3 SG level input to the feed/steam flow mismatch coincident with low SG level reactor trip on October 4 at 1:30 p.m. A plant information report (PIR) and NCR were written to document the installation of an unqualified component. A Material Equipment and Parts List (MEPL) evaluation was also initiated to determine if a capacitor failure could affect the RPS.

The MEPL evaluation concluded that a failure of the capacitor cannot prevent a SG low level condition from being correctly interpreted by the RPS. It was determined to be acceptable for the input capacitor to be unqualified.

The Plant Operations Review Committee (PORC) met on October 4 to discuss the circumstances surrounding the PIR and the conclusions of the MEPL evaluation. PORC concluded that the SG level channel was operable and that the RPS trip signal could be removed. Station management directed the I&C Manager to reinstruct I&C personnel in the importance of performing an engineering evaluation prior to installation of unqualified components into QA Category 1 systems.

The trip signal was removed putting the No. 3 SG level controller back in service on October 4 at 4:13 p.m.

The inspector observed the installation of the RPS trip signal and attended the PORC meeting on October 4. The AWOs associated with this event were reviewed and the event was discussed with station personnel from several disciplines.

ACP 1.2-15.1, "Nonconformance Reports," requires that nonconforming conditions involving materials, parts, components, or installed plant equipment and systems are reported using an NCR. In this case, the licensee identified that an unqualified component was installed into a QA Category 1 system. The system had been placed into service without initiation of an NCR to determine the affects on system operability. Adequate immediate corrective actions were taken and the error was determined to be of minor safety significance. However, the failure to utilize the NCR process prior to placing the No. 3 SG level controller in service constitutes a violation of ACP 1.2-15.1. Although this is a procedural violation, no Notice of Violation is being issued in accordance with the provisions of 10 CFR Part 2, Appendix C, Section V.G.1, Exercise of Discretion (NC 90-15-01).

4.1.5 Steam Generator Tube Leak Repair

Visual inspections were performed of the No. 1 and No. 2 steam generators by closed circuit television for leakage from the secondary into the primary channel head following the October 27 plant shutdown.

One tube at location row 12, column 3 was found to leak at 19 to 22 drops per minute with the secondary at atmospheric pressure. That tube which was found to be dented during refueling outage inspections was again inspected by eddy current. Because of the size of the dent, which was located at the third tube support plate, profile measuring probes could not pass through the tube. A 0.460 inch probe, which is the minimum acceptable size, passed during 1989 refueling outage inspections.

All tube support plates of these steam generators are solid plates.

The licensee 's initial evaluation is that the failure occurred through primary water stress induced cracking at the dent location. The tube was plugged using a mechanically rolled plug. The licensee is evaluating past data for other tubes with potentially similar defects.

Slight leakage was observec a several previously plugged tubes. One had been pluggec ith an early design explosive plug; two others with the Westinghouse mechanical plugs. The two with mechanical plugs had stabilization devices installed. Those devices are designed to prevent plug failure. These were evaluated by the licensee and no further action taken.

The inspector followed the licensee inspection and repair actions. Positive controls were in place to reduce personnel radiation exposures. The same practices and equipment used to support steam generator work during refueling outages was used. However, in this case they were successfully put in place with far less time for planning.

4.1.6 Steam Generator Primary Manway Leak Repair

Leakage was observed at the primary manways of the No. 3 steam generator during a containment inspection conducted on October 25. The reactor was at full power.

The licensee was investigating an observed increase in reactor coolant system calculated leak rate and an increase in containment particulate activity. Calculated leak rates had increased to about 0.75 gallons per minute (gpm) from about 0.5 gpm over several weeks. To facilitate this inspection and obtain as much information as possible within the reactor coolant loop area while maintaining as low as possible personnel radiation exposure, photography was used.

Following detailed inspections during the maintenance outage which began with the October 27 reactor shutdown, leakage was found at all eight primary manways. This was apparently related to a change in gasket material which was done in an effort to avoid the use of asbestos.

All eight of the steam generator primary manways were opened and the gaskets replaced with the type originally specified. Accumulations of boric acid were removed from studs, stud holes flange areas and flange covers. These areas were inspected and in several instances small defects evaluated.

The inspector followed the licensee's actions and found positive efforts in place to reduce radiation exposure and also to address the effects of boric acid accumulation.

4.2 Surveillance Observation

The inspectors 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; and, test results satisfied acceptance criteria or were properly dispositioned. Portions of activities associated with the following procedures were reviewed:

-- Recording of In-Core Neutron Flux Maps

-- Non-destructive Analysis of Steam Generator Feedwater Regulating Valves

-- Calibration of Pressurizer Level Instruments

-- Calibration of Control Rod Position Indication

-- Calibration of Neutron Flux Axial Offset Indication

-- Calibration of Main Steam Line High Flow Trip Setpoint

-- Calibration of Reactor Coolant System Flow Instruments

-- Calibration of Steam Generator Feed Flow/ Steam Flow Mismatch Alarms

-- Emergency Diesel Functional Test

-- Containment Air Recirculation (CAR) Fan Cooler Service Water Hydraulic Resistance Test

-- Inservice Vibration Testing of:
    Charging Pump

    Emergency Diesel Generator

-- "B" Auxiliary Feedwater Pump Functional Test

-- CAR Fan Damper Test

-- Primary to Secondary Steam Generator Leak Monitoring

-- Calibration of Auxiliary Feedwater Flow Instruments for Post-Accident Feedwater Temperature Conditions

-- Evaluation of Reactor Fuel Clad Failures

There were no significant observations noted.

4.2.1 Primary to Secondary Leakage Monitoring

Primary to secondary leakage was observed in the No. 2 Steam Generator though the inspection period. The calculations of leak rate were made based on radiochemical analysis made about every four (4) hours. The data was plotted and the rate of increase (or decrease) was projected in time based on a least-squares fit of datapoints.

Instrumentation was installed and placed in service on October 18 to monitor nitrogen-16 (N-16) gamma decay in the No. 2 main steam line. That device was calibrated to display a calculated leakage rate and was beneficial because of its continuous on-line display of that value. Correlation was observed between this displayed value of leakage and that calculated by radiochemical analysis every four (4) hours.

The inspector followed the licensee's monitoring and trending programs throughout the inspection period. The calculations and trending performed using surveillance procedure SUR 5.4-44 met the licensee's response to NRC Bulletin 88-02, dated March 24, 1988. The installation of the N-16 monitor was a good initiative, and the licensee's decision to perform steam generator inspection and repair was conservative.

4.2.2 Reactor Fuel Clad Failure Monitoring

The inspector followed the results of the licensee's fuel clad integrity monitoring program. That program which is based on traditional and enhanced techniques indicated about 20 to 25 of the small pin hole defects experienced during the last operating cycle.

The licensee's program was established to identify this unique type of clad failure.

The inspectors will continue to monitor it during future inspections.

4.2.3 Containment Air Recirculation Fan Cooler Performance

The inspector followed the periodic monitoring of containment recirculation (CAR) fan cooler performance. That surveillance which calculates cooler hydraulic resistance is performed in accordance with surveillance procedure SUR 5.7-118.

Data and trends were periodically reviewed by the inspector; there were no significant observations noted.

4.2.4 Freeze Protection Performance Monitoring

A deficiency in the station freeze protection system was discovered by licensee personnel inspecting the system. A temperature sensing bulb, associated with freeze protection thermostatic controls, was found to have been repositioned such that it was not exposed to outdoor temperature. The sensor was associated with emergency safeguards pump suction lines from the refueling water storage tank.

During the licensee review of this finding, it was decided that the ability to directly monitor pipe wall temperature would be far superior to the installed means of monitoring freeze protection, which mainly consists of ammeters on freeze protection heater circuits.

The inspector followed the licensee actions in restoring and improving the monitoring of this system. There were no significant observations noted.

5. Security

During routine inspection tours, the inspectors observed implementation of portions of the security plan. Areas observed included access point search equipment operation, condition of physical barriers, site access control, security force staffing, and response to system alarms and degraded conditions. These areas of program implementation were determined to be adequate.

5.1 Fitness for Duty Issue of September 13

The licensee discovered that an employee reported to duty on September 13 in a potentially unfit-for-duty condition. He was observed by his supervisor and a co-worker. Based on behavioral observations, both had reason to believe that the individual was potentially unfit-for-duty. His supervisor was not fully aware that he should have been tested for-cause because he had not received supervisor fitness for duty training. Two other supervisors, who had received the required training, were made aware of the situation but did not advise the employee's supervisor that for-cause testing was appropriate. The employee's supervisor sent the individual home without having him tested on the advice of one of those supervisors.

The licensee informed the inspector of this issue on September 13. Additional details are stated in inspection reports 50-213/90-17 and 90-18.

5.2 System Modifications

The licensee is in the process of replacing the security system process computers and their data and control communication equipment. This project is a licensee self initiative.

6. Engineering and Technical Support

The inspectors reviewed selected engineering activities. Particular attention was given to safety evaluations, plant operations review committee approval of modifications, procedural controls, post-modification testing, procedures, operator training, and UFSAR and drawing revisions.

6.1 Assessment of Power Operation with Loose Parts in the Feedwater System on the No. 1 Steam Generator

The inspector reviewed the licensee's safety evaluation of continued reactor operation with out having retrieved a loose part from the No. 1 steam generator feedwater regulating valve. That evaluation, document No. PSE-EM-90-329, dated September 13, 1990, justified operation through the remainder of the present operating cycle, Cycle No. 16. The evaluation was reviewed and accepted by the Plant Operations Review Committee.

6.2 Evaluation of Observed Differences in Indicated Pressurizer Level

The inspector followed the licensee actions in evaluating an observed difference in displayed pressurizer level. The licensee evaluated the deviations between channels in an effort to determine the most accurate channel. Additionally, steps were taken to insure that actual pressurizer level was maintained as close as possible to the required programmed level. The inspector also followed the licensee actions in reference leg filling and transmitter calibration.

The inspector found the licensee actions to be conservative; of note was the use of statistical analysis of the instrument data to evaluate their performance.

6.3 Evaluation of Leaking Refueling Water Storage Tank

The inspector followed the licensee actions following the discovery of an apparent leak form the plant refueling water storage tank. This included the identification and confirmation of its source through chemical and radiochemical analysis along with their efforts to quantify the leak rate which was about five to eight gallons per day through the inspection period. The tank leak rate is estimated once per shift while removing the collected leakage.

The inspector also reviewed the licensee's analysis of the tank structural integrity, document No. PSE-CE-90-558, dated October 1, 1990 and their evaluation of the off-site dose consequences in the case of a tank failure, document No. CH-90-991, dated September 24, 1990. An evaluation of the radiological assessment is contained in report section 3.1.

The tank bottom was fabricated from one-quarter inch thick pieces of aluminum plate which has been overlapped and fillet welded. The licensee is evaluating methods for inspection and repair. It has been estimated that a one inch long crack in a bottom plate we'd would result in this leak rate.

The inspector will continue to follow the licensee's actions

6.4 Evaluation of Auxiliary Feedwater Flow Instruments

The licensee commenced an evaluation of the qualifications and calibration of the auxiliary feedwater flow instruments after that variable was recently upgraded to a Type A variable defined by Regulatory Guide 1.97 following recently discovered deficiencies in the auxiliary feedwater system (reference inspection report 50-213/90-13, section 6.2).

The inspector followed the licensee's actions which included replacement of terminal box terminal strips with environmentally qualified splices and the recalibration of the instruments for cooler water. The instruments had been calibrated for normal feedwater temperature. However under post accident conditions, the cooler water was observed to result in indicated flow unconservatively high. At the assumed 80 gpm post accident auxiliary feedwater flow, indicated flow would have been 89 gpm. A 30 gpm error would have been present at full instrument scale of 300 gpm.

The inspector followed the licensee investigation and corrective actions. There were no significant observations noted.

6.5 Diagnostic Tests of the "A" Emergency Diesel Generator

During the previous inspection period (NRC Inspection Report 50-213/90-13), the "A" emergency diesel generator (EDG) experienced a start failure and several start failure alarms while being surveillance tested. These occurred while the air start system west bank was selected and the engine was started from the excitation panel. Several tests were performed, but no cause for the start failure and alarms was determined. The licensee did observe that in tests immediately following a slow start the EDG performed satisfactorily and was declared operable.

During this inspection period, diagnostic testing of the "A" EDG was performed on September 17. Troubleshooting of the start failure alarm was conducted under AWO 90-9648 and simulated the start configuration in which the problems were experienced. Maintenance, Instrumentation and Controls, Engineering, and Quality Services personnel participated as well as the EDG vendor representative. The engine was run twice with recorders installed to measure the response of selected start circuitry relays, engine speed, and air start system pressure. Start failure alarms were received for both starts. Test data was analyzed and test personnel concluded that the air start system and start circuitry are satisfactory. Further testing of the governor booster oil system was recommended.

The monthly surveillance test was performed as the post maintenance test; no start failure alarm was received. The engine was returned to service on September 17.

On October 9, the monthly surveillance test was successfully performed on the "A" EDG. The engine attained idle speed in 6.95 seconds and no start failure alarm was received. The inspectors noted that this starting time meets the acceptance criteria of the surveillance procedure, and historically the engine has started in about 5 seconds during this surveillance test.

6.6 Auxiliary Steam System Modifications

Based on the results of studies of the plant auxiliary (heating) steam system, portions which supplied control room ventilation heaters were taken out of service. The concern was potentially undesirable effects of a pipe break.

During this inspection period, a new auxiliary steam line was installed to supply heating steam to the control room air handling unit. The inspector observed portions of the work in progress. There were no significant observations noted.

7. Safety Assessment and Quality Verification

7.1 Plant Operations Review Committee

The inspectors 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. 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.

7.2 Review of Written Reports

Periodic and Special Reports, and Licensee Event Reports (LERs) were reviewed for clarity, validity, accuracy of the root cause and safety significance description, and adequacy of corrective action. The inspectors determined whether further information was required. The inspectors also verified that the reporting requirements of 10 CFR 50.73, Station Administrative and Operating Procedures, and Technical Specification 6.9 had been met. The following reports were reviewed:

LER 89-06-01 Heating Steam Containment Isolation Valves Failed Surveillance Test

LER 90-15 Failure To Add New Fire Door To Surveillance Procedure

LER 90-16 Design Deficiency Identified In Auxiliary Feedwater Auto Actuation System

LER 90-17 Auxiliary Feedwater System Actuated While Troubleshooting Feed Flow Recorder

LER 90-18 Manual Plant Trip Due To Feedwater Control Valve Failing Open

LER 90-19 Two of Three Pressurizer Level Channels Declared Inoperable

LER 90-20 Manual Plant Trip Due to Condensate Pump Degradation and Procedural Noncompliance

LER 90-21 Surveillance Frequency Exceeded for Turbine Building Heat Detector Test

Haddam Neck Plant Monthly Operating Report 90-09, covering the period September 1 to September 30, 1990

Haddam Neck Plant Cycle 16 Startup Physics Test Report, dated October 19, 1990

There were no significant observations noted.

8. Exit Interviews

During this inspection, periodic meetings were held with station management to discuss inspection observations and findings. At the close of the inspection period, an exit meeting was held to summarize the conclusions of the inspection. No written material pertaining to this inspection was given to the licensee and no proprietary information related to this inspection was identified.

In addition to the exit meeting for the routine resident inspection, the following meetings were held for inspections conducted by Region I based inspectors.

Report No.	Inspection Dates	Reporting Inspector	Areas Inspected
213/90-80	September 10 - September 21	L. Prividy	Maintenance Team Inspection

Report No.	Inspection Dates	Reporting Inspector	Areas Inspected
213/90-17	September 24 - September 28	E. King	Fitness For Duty Program
213/90-18	September 25 - September 26	G. Smith	Fitness for Duty Event
213/90-16	October 1 - October 5	L. Cheung	Compliance with NRC Regulatory Guide 1.97
213/90-81	October 9 - October 17	J. Prell	Emergency Operating Procedures Team Inspection
213/90-82	October 1 - October 12	G. Kelly	Employee Safety Concerns Program Team Inspection