U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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EXECUTIVE SUMMARY

Haddam Neck Station
NRC Inspection Report No. 50-213/96-10

This inspection included aspects of licensee operations, engineering, maintenance, and plant support, and the licensee recovery from a significant operational event. The report covers a 6 week period of resident inspection.

Plant Operations:

Licensee preparations for adverse weather conditions were adequate and generally thorough. The loss of an offsite distribution line resulted in a reduction in the independent power supplies, but did not impact the operation of equipment needed to cool the reactor. Plant operators responded in accordance with plant procedures.

Operators and plant staff showed poor sensitivity to the control of shutdown risk during the period from August 28 - September 1, as several events occurred which significantly reduced plant safety margins. The events included the inadvertent decrease in reactor vessel level during plant shutdown conditions, and the degradation in plant equipment needed for decay heat removal system. For approximately four days, control room operators were unaware that nitrogen gas was leaking into the reactor vessel and causing level to decrease. The NRC identified several significant problems related to the event: inadequacies in plant operations procedures, operator failure to follow procedures; inappropriate operator decisions; and inadequate pre-job briefings, resulting in inappropriate equipment manipulations. Poor material conditions were revealed in valve leakage through systems that interfaced with the RCS.

Maintenance:

The licensee took actions to restore equipment needed to restore margins and redundancy for the shutdown cooling function. Several performance deficiencies were identified in the actions to correct conditions adverse to quality. Maintenance support for the timely restoration of inoperable decay heat removal equipment was poor. The availability of quality parts, vendor specifications, and repeat test failures resulted in excessive delays in restoring RHR to an operable status.

Following the loss of control of outage work activities resulting in an inadvertent drain of the reactor vessel on September 4, the licensee implemented a system of controls that were adequate to protect key safety functions and allow the gradual resumption of outage work.

Engineering:

The engineering support for the control of special nuclear materials was inadequate, as demonstrated in the failure to follow special nuclear material (SNM) inventory procedures and in the failure to complete a timely physical inventory of irradiated fuel in the spent fuel pool. The housekeeping and cleanliness conditions in the spent fuel pool were poor and unacceptable to support the conduct of SNM inventory checks. The licensee position on the recording of fuel bundle serial numbers in addition to completing a piece count while

performing an SNM inventory in accordance with 10 CFR 70.51 is a matter that requires further NRC review. The materials of construction and the analytical approach used to assure acceptable material strengths for the new spent fuel pool storage racks was acceptable.

Plant Support:

Radiological and security controls were maintained in accordance with licensee and regulatory requirements.

Safety Assessment & Quality Verification:

The plant management and staff failed to appreciate the significance of the nitrogen bubble event and systematically under reacted to the event initially, resulting in delays in initiating and integrated event response. There were deficiencies in the quality of information during the event response, in the maintenance and engineering support provided to mitigate the event, and in the availability of quality parts for safety related equipment. There was inadequate integration of plant resources and support activities to effectively respond to degraded plant conditions in a recovery type of forum. The licensee review of a precursor event was not thorough. Delays were encountered in establishing vessel level indication, and backup core cooling methods. These events revealed fundamental weaknesses and broad concerns in the overall conduct of operations of Haddam Neck.

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REPORT DETAILS

Summary of Plant Status

Prior to this period, the plant was shutdown with extensive engineering activities in progress to address design basis issues that would allow a plant restart. On August 8 the licensee decided to start the cycle 19 refueling and maintenance outage early. The plant was operated in cold shutdown during the period as preparations were made to prepare the reactor for core offload. The reactor was partially drained and vented when the reactor disassembly sequence was placed on hold on August 30, 1996. Several significant events occurred which challenged reactor cooling and inventory control: a through wall leak was identified on the inlet valve for the "A" residual heat removal (RHR) heat exchanger on August 30; the "B" RHR pump failed as the operator tried to start it on September 1; and, the operators discovered on September 1 that about 5000 gallons of reactor coolant system (RCS) inventory was inadvertently drained when a nitrogen bubble accumulated in the reactor head. The causes and the management response to these events were reviewed by an NRC augmented inspection team.

During a partial work stoppage intended to preclude a loss of RCS inventory, about 300 gallons of RCS inventory were inadvertently drained from the reactor to the containment sump on September 4. The Unit Director and the Nuclear Safety and Oversight group implemented a full stop work order in response to the September 4 event. This action brought plant outage activities to a halt, except those activities needed to restore the RHR system to an operable status. For the remainder of the inspection period, the licensee focused on measures to complete an independent review of the events, and to implement a recovery plan that would restore the "B" RHR pump to an operable status, evaluate the degradation on the "A" RHR heat exchanger, and prepare the reactor for core offload. This inspection focused on the licensee activities to recover from these events. The results of the augmented team inspection are described in NRC Inspection 50-213/96-80.

Organization Changes

The licensee announced the following business and organizational changes during the inspection period. On August 29, the licensee announced that the economic analysis for Haddam Neck would be updated. The study was designed to determine whether the plant remained an economic producer of electricity, and was scheduled take about two months to complete. On September 3, Mr. Bruce Kenyon became the President and Chief Executive Officer of Northeast Utilities, replacing Robert Busch. On September, 18, the licensee announced management changes as part of a Recovery Organization for the four NU nuclear plants. The new organization included the position of Recovery Officer for Haddam Neck, which would remain unfilled pending completion of the economic analysis. During the interim, T. C. Feigenbaum was designated as the Recovery Officer for the Haddam Neck Site. The plant staff was reorganized under three Directors as follows: Mr. Jere LaPlatney was designated as Unit Director with oversight for the functional areas of operations, maintenance/construction, work planning/outage management, corrective action, I&C/electrical and operational readiness; Mr. J. Haseltine was designated as Engineering Director, with oversight for the functional areas of Engineering systems, design engineering, and configuration management; and, Mr. Gary Bouchard was designated as the Director of Work Services, with oversight for the functional areas of security/medical, HP/radwaste, stores, information technology, facilities and chemistry. The new organization would become effective on October 1, 1996.

I. Operations

O1 Conduct of Operations[1]

Using Inspection Procedure 71707, the inspectors conducted periodic reviews of plant status and ongoing operations. Operator actions were reviewed during periodic plant tours to determine whether operating activities were consistent with the procedures in effect, including the alarm response procedures.

O1.1 Preparations for Severe Weather

a. Inspection Scope (71707)

The inspection scope was to review licensee actions to prepare the site for severe weather.

b. Observations and Findings

The licensee took actions to prepare the site and to track the progress of Hurricane Edouard during the period of August 29 - September 1, 1996. The operators entered abnormal operating procedure (AOP) 3.2-5, "Natural Disasters" at 5:00 a.m. on August 31 as the storm came within 800 miles of the site with a forecast of hitting New England. The forecast at one point was for the hurricane to pass over Bristol, Connecticut by mid day on September 2, making it a direct threat to the Haddam Neck site.

The licensee took actions well in advance of the predicted landfall to inspect the site and secure loose debris which could generate flying debris that could impact personnel, structures or the station power supplies. Site inventories were also reviewed and replenished as necessary to assure water, chemical and fuel oil supplies were adequate. Both emergency diesels were confirmed to have been recently tested and in a standby ready condition. Emergency diesel EG-7 had been installed and connected to Bus 1-2, but was not considered available to the operators pending actions to complete testing and turnover from construction. The reactor containment building was in a status that was less than full integrity, but conditions for closure were established: the equipment hatch was installed and fastened sufficiently to handle partial pressurization; both personnel access doors were closed; the containment purge valves were open with ventilation in progress and capable of closure; and any other containment penetration not fully secured was tagged and identified for rapid closure. Outage work activities inside containment had been suspended for the period of August 30 - September 2.

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

The inspector toured the site periodically during August 29 - 30, to review the activities in progress and during the back shift hours on August 31 to assess the completion of licensee preparations. The inspector identified loose materials and sheet metal debris stored in dumpsters which could become airborne and pose a potential hazard for the 115 KV distribution lines. These areas were discussed with the duty shift manager, who took action to better secure the materials.

Hurricane Edouard's projected landfall over central Connecticut with 115 miles per hour winds made it a significant threat to the plant and an additional operational distraction to the operators as the weekend events unfolded. Shift personnel continued to make preparations for the hurricane through the mid shift on August 31 - September 1. As of 7:30 a.m. on September 1, the hurricane preparations had been completed with building exterior doors secured and strongbacks in place on the turbine building roll up doors. By September 1, the projections were for the storm to pass over Watch Hill, Rhode Island, such that the effects on the site were expected to be less severe. Hurricane Edouard tracked east of New England on September 2 and ultimately had no impact on the site. No deficiencies were noted.

c. Conclusions

Licensee preparations for adverse weather conditions were adequate and generally thorough.

O1.2 Degraded Reactor Inventory and Core Cooling

a. Inspection Scope(71707)

The inspector reviewed the licensee actions in response to a series of events that challenged reactor vessel inventory control and core cooling. This inspection focused on licensee recovery actions in response to degraded plant and equipment conditions.

b. Observations and Findings

The plant was in cold shutdown as preparations were made to prepare the reactor for core offload as part the refueling and maintenance outage. As part of the vessel disassembly sequence, the missile shield had been removed in preparation for lifting the vessel head. Core exit thermocouples and reactor vessel level instruments had been disconnected since the associated cabling connections had been removed with the shield blocks. The reactor was vented and partially drained to the vessel flange reference level when the reactor disassembly sequence was placed on hold on August 30, 1996. The operators refilled the reactor to a level near the top head and left the reactor coolant system vented to the vent header at the pressurizer and the head. The instrument used by the operators for vessel level control was the pressurizer cold calibrated level instrument, LI-402. The cavity level instrumentation system (CLIS) was also available and used by the operators as an aid to trend vessel level changes. However, there was no procedure or formal guidance available to correlate the pressurizer and CLIS indications. The reactor coolant

system loops were filled, but isolated (loop stop valves closed). Reactor coolant temperature as measured by instruments in the RHR system was about 105 degrees fahrenheit (F).

Several events occurred which challenged reactor cooling and inventory control: about 500 gallons of water were diverted from the reactor coolant system on August 22 while placing the RHR purification system in service (adverse condition report (ACR) 96-926); a nitrogen leak path into the reactor cooling system (RCS) was inadvertently created on August 28 while aligning boration flow paths (ACR 96-946); a through wall leak was identified on the inlet valve for the "A" residual heat removal (RHR) heat exchanger on August 31 (ACR 96-968); the "B" RHR pump failed as the operator tried to start it on September 1 (ACR 96-964); the operators discovered on September 1 that about 5000 gallons of RCS inventory was inadvertently drained when a nitrogen bubble accumulated in the reactor head (ACR 96-966); and, RCS water was diverted to the containment sump via the spray header on September 4 during preventive maintenance on valves RHR-MOV-23 and 34 (ACR 96-978). A brief summary of each event along with the licensee's immediate corrective actions follows.

(1) ACR 96-926, Water Diverted from RCS While Restoring Purification

The reactor was in cold shutdown on August 22 with core cooling provided by the "A" RHR pump aligned to the "A" RHR heat exchanger. RHR purification was in service by providing a slip stream of about 120 gallons per minute (gpm) through a flow path established from the heat exchanger through RHR-V-874A to the ion exchangers and back to the vessel. The operators secured the purification system per procedure NOP 2.7-4 (Attachment 4 - Temporary Shutdown of RHR Purification) to accommodate surveillance testing of a valve in the flow path. When the testing was completed, the operators returned the purification system to service, but inadvertently skipped step 1.4 in normal operating procedure (NOP) 2.7-4 during the restoration sequence. Step 1.4 directed the operators to complete a valve lineup in preparation for the restoration. Failing to complete the lineup resulted in leaving valve PU-V-261A open, which created a flow path from the RHR system to the refueling water storage tank. The consequence of this error was that about 500 gallons of reactor coolant was diverted to the refueling water storage tank (RWST) when RHR purification was placed back in service at 9:10 p.m. on August 22.

The operator noted the diversion after receiving a low level alarm from the cavity level indication system (CLIS). The operators immediately closed RHR-V-874A to terminate the event. The licensee determined that the error occurred because the operator failed to follow NOP 2.7-4; a contributing cause was that the procedure was inadequate in that step 1.4 did not require a sign off like other steps in the restoration sequence. The licensee imposed a stop work order for the operators, which prohibited routine operator activities except as needed to minimize shutdown risk and to assure compliance with the technical specifications. The specific procedure deficiency was addressed by a change to the procedure and other procedures were reviewed to identify other potential deficiencies. The other operating shifts were briefed regarding the event to review lessons learned.

Management expectations regarding pre-job briefs were reviewed with the operators. The licensee lifted the stop work order on operations on August 24 after completion of a review to assure the corrective actions addressed the causes of the diversion event. This corrective action was found to be inadequate to prevent subsequent operational events.

(2) ACR 96-946, Nitrogen Intrusion Into RCS August 28

On August 28 with the plant in Mode 5 and the RCS vented to the primary event header, the operators identified the need to change the boration flow path. The "B" emergency diesel generator was scheduled to be run for surveillance. The diesel would be inoperable when run in parallel with the grid. Since the metering pump is powered from the "B" emergency diesel generator (EDG), it would be technically inoperable during the surveillance, thus necessitating a change in the credited boration flow path to meet technical specification (TS) 3.1.2. While realigning the boration flow path in accordance with surveillance procedure (SUR) 5.1-159B, the nuclear side operator (NSO) inadvertently opened valve BA-V-355 at 8:15 a.m. This created an unintended flow path from the isolated volume control tank (VCT) via the charging system and the flow of nitrogen to the RCS. The operators noted an increase in cavity level and a decrease in VCT pressure. BA-V-355 was closed. Several deficiencies contributed to this event, including: the auxiliary operator had not performed the task before; the procedure was confusing; and, no pre-job brief was performed prior to the evolution.

The operators realized that nitrogen had been introduced into the RCS and took actions to vent the suction header of the "B" charging pump since it was presumed to be gas bound. The operators wrote a procedure to perform the vent per administrative control procedure (ACP) 1.2-5.3 and the venting was completed at about 1:30 p.m. on August 28, 1996.

The operators had correctly diagnosed the nitrogen intrusion and determined that about 500 gallons of RCS inventory had been displaced out of the reactor. The event was considered a significant precursor since the charging pump likely would have failed had it been operating when the nitrogen entered the header. Unbeknownst to the operators, nitrogen continued to flow into the RCS (later determined to be at a rate of 7 cubic foot per minute) through valve BA-V-355.

(3) ACR 96-968, Leak on RHR Valve 791A August 31, 1996

While performing operational rounds on August 31, a NSO noted boron deposits on the body of valve RHR-V-791A. The operator wiped the boron off and noted a small amount of weepage from a through wall leak in the valve body. The valve was the RHR inlet isolation valve for the "A" RHR heat exchanger. The valve was an 8 inch diameter double wedge gate valve, and was a type 316 stainless steel casting manufactured by Alloyco. The leak rate was very low with an operator estimate of 0.1 ml/minute; it would take several seconds for wetness to reappear once the areas was wiped clean. The licensee declared the valve inoperable at 1:27 p.m. on August 31.

The reactor was in cold shut down at the time with reactor cooling provided by the operating "A" RHR pump and with both heat exchangers in service. The through wall defect in the safety class 2 pressure boundary rendered that portion of the RHR system inoperable even though the heat exchanger remained functional for reactor decay heat removal. The licensee reported the issue to the Nuclear Regulatory Commission (NRC) per 50.72 (b)(2)(i) as a degradation in a primary safety barrier that placed the plant in an unanalyzed condition.

The shift manager called maintenance personnel and engineering personnel to the site for support in evaluating the valve and initiating a corrective action plan. The leak was initially considered to be not isolable until an evaluation was completed by the system engineer (SE). The initial plans were to complete the evaluations necessary to apply a soft patch (leakage limiting) and to arrange the non destructive examinations (NDE) needed to accomplish a temporary repair. The NDE evaluation of the flaw was needed to obtain additional assurance of the continued integrity of the component and RHR piping, and further to determine whether it was acceptable to follow the planned core offload sequence. At the time, the vessel was partially disassembled in support of core offload, with the core deluge lines and vessel level instrumentation disconnected. Further, on August 31, the operators considered that the loops were filled but isolated, and thus reactor heat removal with the steam generators was available on a delayed basis if needed.

The system engineer had evaluated the valve status by 3:00 p.m. on August 31 and recommended that the valve be shut to isolate the leak and to continue core cooling with one heat exchanger. The leak was in the disk guide area in the upper portion of the valve body, which could be isolated form the 120 psi RHR pressure boundary by closing the valve. Even though the leak was considered very minor and to not pose a flood concern, it was part of the RHR pressure boundary and should be isolated as soon as possible. The plan was to isolate the leak in case the flaw was worse than it appeared; this contingency would preclude a further degradation of the operating RHR loop. Actions to isolate the "A" heat exchanger were completed on the swing shift on August 31. The SE assisted the operators by providing guidance on the allowable component cooling water flow needed to support core cooling using a single heat exchanger.

At least three other RHR valves were immediately identified as the same make and model as RHR-V-791A. The licensee planned to include these components in the evaluation of RHR system integrity. The inspector held several conferences with licensee and NRC management and technical personnel on August 31 regarding the NRC guidance and requirements (in Generic Letter (GL) 90-05 and 10 CFR 50.55a) for degradation in safety class 2 components. The licensee intended to use the guidance of GL 90-05 and 50.55a to place the inoperable "A" RHR loop back in service after characterizing the flaw. The licensee intended to seek formal NRC approval to grant either a relief from the 50.55a code or to approve a temporary repair of the valve as necessary.

The licensee's initial contingency plan on loss of RHR was to unisolate a loop and to steam a generator. The "A" RHR loop was considered inoperable with the defect in RHR-V-791A. The applicable technical specification was TS 3.4.1.4.2, which required for Mode 5 with the loops not filled (isolated) that both RHR be operable with one RHR pump operating. The associated action was to return the loop to an operable status as soon as possible (which was in progress). Operations management was at the site to support the operators and identified alternative actions available to the operators: if the loop stop valves were opened with the loops still filled, then TS 3.4.1.4.1 applied and only one RHR loop need be operable. The time to core boil could be lengthened from about 85 minutes (loops isolated) to about 157 minutes by opening the loops stop valves. The licensee concluded on September 1 that it was not prudent to lower vessel level to the flange reference level (planned for September 3) for head lift with the existing RHR status.

The inspector responded to the site on August 31 and independently confirmed the status of licensee response actions. The operations staff with support from engineering and maintenance continued to make preparations to perform NDE performed on RHR-V-791A throughout the evening of August 31. The radiographer contractor sent personnel to the site. The work center prepared the work orders needed to support the radiography. Additional personnel were called to the site to support rigging the radiography equipment down into the RHR pit, which involved a 40 foot drop over safety related equipment. The radiographer was at the site by 11:00 p.m. with a 90 curie (Ci) Co-60 source shielded in a 600 pound pig. Health physics personnel briefed the contractor personnel on the area where the work would be done and the associated radiological conditions. The inspector raised concerns with the shift manager and the Duty Officer that adequate resources (number of personnel, appropriate technical discipline) be applied to a movement of heavy load over operating RHR equipment as a shutdown risk issue. The inspector noted the recent history at Haddam Neck involving licensee and NRC identified problems with the implementation the heavy loads program, and the problems with the primary auxiliary building (PAB) floor block lift rig in particular. The shift manager deferred proceeding with the rigging activity pending resolution of these issues and the development of additional support by maintenance and engineering personnel.

Site staffing continued to be the normal operations crew plus the radiographer and support personnel thorough the day shift on September 1. Operations management was also present at the site to support the activities. The radiographer evaluated the valve as necessary to make preparations for the radiographic test (RT). The SE prepared additional evaluations needed to pull the floor blocks that would address potential flooding issues as well as the control of heavy loads over safe shutdown equipment. The rigging path selected was over the "A" RHR pump (north end of the pit). Plans were made to swap from the "A" to the "B" RHR pump for core cooling so that the rigging would be done over the "inoperable" "A" RHR train.

Licensee actions to complete the NDE on RHR-V-791A were completed after the end of this inspection period. While the leak and continued structural integrity questions remained safety significant, the work to restore a second operable RHR pump became the higher priority activity after the "B" RHR pump was found to be inoperable on September 1.

(4) ACR 96-966, Nitrogen Intrusion Into RCS September 1, 1996

Starting at about 9:00 a.m. on September 1, the operators noted indications of an apparent drain of reactor coolant inventory from the RCS. The operators had been investigating over the period from August 28 - September 1 an excessive use of nitrogen. When the nitrogen supply to the VCT was isolated as part of the troubleshooting on September 1, a slow loss of pressurizer level occurred over the next several hours until 1:00 p.m. The operators responded to the level decrease by making up to the reactor from the refueling water storage tank to keep the level indication on scale on LI-402. The operators made up to the RCS four times over the period of four hours, as pressurizer level varied between about 5% and 10%. The crew responded to an apparent RCS leak (estimated to be 20 gpm) between 9 and 10 a.m. by implementing AOP 3.2-31A, Reactor Coolant System/Refueling Cavity Leak, but found no leakage in the containment or RHR area.

The operators correctly diagnosed that the apparent leakage was a relocation of the RCS inventory from the pressurizer to the RCS as pressure in the reactor head decreased after the nitrogen was secured. By about 1:00 p.m., the operators had assembled plant data and reconstructed events to identify the inadvertent introduction of nitrogen into the RCS and build-up of a bubble in the head. The leak path was from the nitrogen system to the VCT to the RCS via charging system. The nitrogen leaked from the VCT through valve BA-V-355, which had been closed during the event on August 28, but had continued to leak by its seat because it was not totally closed. The operators initially estimated that several thousand gallons of inventory had been displaced by the nitrogen; the volume was later calculated to be 5000 gallons.

Although the RCS was vented to the vent header, the operators initially suspected that a slug of water had accumulated in the vent tube between the reactor head and the charging floor, and a loop in tube down to the cavity floor provided a vent seal until pressure built up to clear the line of water. The licensee was initially not sure what caused the vent system obstruction, and the event sequence was reviewed over the next several hours using log sheets to correlate pressurizer and cavity levels. The information regarding the presence of a loop seal in the cavity portion of the vent was later found to be inaccurate. Ultimately, the license concluded that the vent system had been inadequately designed and operated.

The reactor remained adequately cooled throughout the nitrogen intrusion event and during the period on September 1 when level between the reactor and pressurizer equalized. RHR temperatures remained stable at about 105 degrees F and pressurizer level remained between 2% and 10%. However, while it was clear that level in the reactor had decreased while the nitrogen bubble had existed, there was

some uncertainty on just how low level had been and whether level had been restored to the desired level inside the head.

The immediate corrective action was to secure the nitrogen addition by closing two series valves between the VCT and the RCS. Valves BA-V-355 and BA-V-354 were tagged closed (reference clearance 96-895). Further, the nitrogen supply to the VCT was secured to remove the motive force driving nitrogen into the charging system. Finally, the operators recognized that conditions were similar to the nitrogen injection event on August 28, and actions were completed to vent nitrogen from the charging system.

For approximately four days, control room operators were unaware that nitrogen gas was leaking into the reactor vessel and causing level to decrease. The licensee later determined that by September 1, 1996, reactor vessel level had decreased to approximately 3 feet below the reactor vessel flange. The licensee subsequently reported this event to the NRC on September 11, 1996 in accordance with 10 CFR 50.72(b)(2)(iii)(B) as an event that could have affected the operability of the RHR system.

The events revealed deficiencies in plant material conditions that allowed the leak path to occur: inadequate followup to the August 28 event; a poor sensitivity to the control of shutdown risk; and, a lack of a questioning attitude to pursue operational concerns in a timely manner.

(5) ACR 96-964, "B" RHR Pump Seized

The operators planned to shift from the operating "A" RHR pump to the standby "B" RHR pump in preparation for the radiography on valve RHR-V-791A. The operators attempted to start the "B" RHR pump at 2:19 p.m. and discovered that the pump was mechanically bound. After two start attempts in which the operators noted excessive and lengthy in rush currents, the pump was declared inoperable and apparently bound. The pump could not be turned by hand. The motor was subsequently disconnected from the pump. The motor was found to turn freely and plant electricians subsequently meggered the motor with satisfactory results. The priority in the licensee corrective actions shifted to restoring the "B" RHR pump to and operable status.

By mid afternoon on September 1, the licensee had opened the loop isolation valves for reactor coolant system loops #3 and #4 by 6:00 p.m.; all four loops were open to the reactor vessel by 9:30 p.m. Technical Specification Definition 1.27 allows any combination of operable RHR pump and RHR heat exchanger to constitute an operable RHR loop. Thus the plant operated in compliance with Technical Specification 3.4.1.4.1 with the operable "A" RHR pump and the "B" RHR heat exchanger when at least two RCS loops were filled and connected to the RCS. The action to unisolate the loops effectively doubled the time to core boiling (from about 60 minutes to 120 minutes). The licensee also expected that, in the event of a further degradation in the RHR system and the complete loss of RHR cooling, the reactor could be cooled by natural circulation. The status of electric power supplies

was good with both emergency diesel generators and both 115 KV offsite distribution lines available. The shutdown risk diesel EG-7 was considered available to the operators but had yet to be turned over from construction. The assurances of adequate core cooling and stable inventory were discussed. The licensee reviewed the emergency plan and concluded that no emergency action levels were met for the existing plant conditions.

The inspector responded to the site on September 1 and independently confirmed stable plant conditions and adequate core cooling based on observations of control board indications and plant equipment status. The licensee briefed NRC management regarding plant conditions and planned recovery actions during teleconferences during the evening of September 1 and on September 2. The focus of the briefings was to understand the sequence on how nitrogen got into vessel, how it was isolated and what was the status of the reactor vent. The NRC also reviewed the status and appropriateness of licensee corrective actions to stabilize the plant, correct degraded equipment conditions, and to recover margins to core cooling and inventory control. The NRC expressed concerns to the licensee on the significance of bubble event, the adequacy of resources and the timeliness of response actions.

(6) ACR 96-978, RCS Water Diverted to Containment Sump

The Unit Director issued a partial stop work order on September 2 to halt work activities on the primary side plant. The stoppage was intended to preclude any activity that created the potential for an inadvertent loss of RCS inventory as actions continued to restore margins to core cooling and vessel inventory control. Despite the stop work order, authorized work orders (AWO) 96-6265 and 96-6266 were issued to perform electrical maintenance on the motor operators for RHR system valves RHR-MOV-23 and RHR-MOV-34, respectively. Even though the work control center (WCC) had implemented the stop work order, these AWOs were issued for implementation because they were coded against equipment outage window 40 (miscellaneous reactor plant systems), the planned work was not considered a challenge to shutdown cooling (electrical PM), and a manual valve (RH-V-23A) downstream of the MOVs was closed. The shift operations crew used a similar rationale when the AWO was presented to the shift for approval to start work.

As part of the motor preventive maintenance, the operator is required to signal the valve open to pull each valve off the seat. This was done in coordination with an electrician stationed at the valve breakers, who opened the breaker as soon as the valve was signaled open. The MOVs were opened at about 1:00 p.m. on September 4. Valve RH-V-23A was not fully closed which created a flow path from the RHR system to containment spray header through automatic drain valve RH-V-888 to the containment sump. The operator noted a decrease in cavity and pressurizer level and immediately contacted the electricians to close the valve motor breakers so the MOVs could be closed. Valves RH-MOV-23 and RH-MOV-34 were reclosed about 15 minutes after they were opened. During this period, pressurizer level decreased from 3% to 0% and cavity level decreased from 337 to 332 inches.

The licensee estimated that about 300 gallons of RCS inventory was inadvertently drained from the reactor to the sump on September 4.

The licensee completed other short term corrective actions. The operators checked valve RH-V-23A and found that it could be closed by about one-eighth of a turn. To further preclude any activity that could impact shutdown risk, the Unit Director issued a stop work order for all outage activities. This event revealed an inadequate control of work activities, and a poor sensitivity to the control of shutdown risk.

(7) Recovery Actions

On September 2 at 3:00 p.m., the Unit Director was onsite and conducted a meeting with the event response team leaders. This is the first time the CY response efforts came together in an integrated fashion for the events that occurred over the period from August 30 - September 1. This review brought together the history of events; the present plant status and the state of degraded equipment or systems; and, the shutdown risk status (including available contingencies for the operators). Assignments were made for major work items along with priorities. This meeting defined roles and responsibilities for event investigation and corrective action, and identified the need to establish, organize and staff a defined recovery organization. The licensee's response to the operational events was also summarized in a letter to the NRC dated September 9, 1996.

The licensee established a plan to regain margins to core cooling and to prepare for a resumption in the core offload sequence. This plans was not formalized until September 6, 1996. It was appropriate to proceed with core offload because that condition represented the lowest level of shutdown risk. However, the reactor plant must pass through a relatively high risk condition (reactor drained to vessel reference level) in order to proceed to core offload. The licensee established the following criteria which must be satisfied prior to proceeding to core offload: (i) both RHR trains are available for service, including the securing of regulatory relief as needed; (ii) the conduct of an independent review team (IRT) to investigate and determine the root cause of the major events that challenged reactor safety margins; and, (iii) the completion of appropriate corrective actions identified from the IRT as related to the initiation of core offload.

The licensee considered the plant staff to be in a recovery mode until the completion of these actions and the start of core offload. Finally, the licensee requested the NU Safety Analysis Branch to complete an analysis of the nitrogen intrusion event to assess the adequacy of the available compensatory measures and the potential plant vulnerabilities.

(a) Core Cooling

The licensee reviewed the status of contingency plans for loss of core cooling, and the strategies outlined in AOP 3.2-12, in particular. The AOP was in effect at the time on the events on August 28 - September 1 and provided the following mitigation strategies for the loss of RHR: RCS pressurization and natural circulation

using the steam generators as a heat sink; forced circulation using a reactor coolant pump and the steam generators as a heat sink; and, the use of the low pressure safety injection pumps in a recirculation path back to the refueling water storage tank, with core cooling provided by the RHR heat exchangers.

After taking actions to unisolate the reactor coolant loops, the licensee took actions necessary to prepare a reactor coolant pump (RCP) for operation, as follows: the #3 RCP was selected for service and grounding carts were removed to allow insertion of the breaker in Bus 1-1B; RCP seal water cooling supply and return flow path was aligned but not placed in service; the control power fuses for the #3 RCP were made available but were not installed; the tie breaker from Bus 1-2 to Bus 1-1B was available to assure the prompt restoration of power to the #3 RCP bus; and, the alignment of feedwater using the electric auxiliary feedwater pump from the demineralized water storage tank was verified to be in standby readiness. The RCS integrity was verified to be intact except for the vent valves on the pressurizer and reactor head. The "B" charging pump was verified to be available for RCS pressurization up to 100 psig. Thus, the plan was established to use an RCP for core cooling in the event of a loss of the "A" RHR pump for any reason. The alternate core cooling path was available to the operators on September 3.

(b) Head Vent System

The licensee gradually recognized and addressed inadequacies with the vessel head vent system over the course of several days following the nitrogen intrusion event. Actions were taken to make changes to eliminate design discrepancies and improve the installation. The improvements to the head vent system included the use of larger diameter tubing, the use of new fittings, the elimination of sharp bends, the use of a new digital manometer to replace the u-tube manometer, and the relocation of connection points to the vent header to avoid flow restrictions. The operation of the vent system was also improved by changing normal operating procedure (NOP) 2.9-6 (TPC 96-402) on September 4 to better establish and monitor the vacuum compensation equipment. The proper functioning of the vent system boundary valves was also demonstrated by a test conducted on September 5. The licensee developed the correlation between the pressurized level and cavity level indication systems, and trained the operators on the new procedures.

(c) Reactor Level

The licensee took actions to provide confirmation that the challenge to core cooling provided by the bubble had passed. Plant workers took readings on September 3 using a test box connected into the reactor vessel level indication system (RVLIS) probe at the head connection point. These readings showed that level was near the top of the hemispherical head. On September 3, workers used a bypass jumper 96-51 to provide readout for three core exit thermocouples (CETs). This jumper provided readouts in the control room for CETs positioned at core locations P9, P10 and M10. All three CETs showed core exit temperatures in the range of 96-99 degrees F, which provided confirmation of adequate core recovery and subcooled conditions. Finally, the licensee implemented bypass jumper 96-55 on September

11 to provide a temporary connection for RVLIS train "B," with continuous readout in the control room. Both the CETs and RVLIS indications remained available to the operators to provide continued confirmation of adequate core coverage for the remainder to the recovery effort. Finally, on September 5, the licensee completed tests on the reactor vent system to confirm its adequacy (following modifications) and also completed a test to provide absolute confirmation of level.

(d) Refueling Sequence

Based on an analysis of the September 1 nitrogen bubble event, the licensee recognized that the refueling sequence defined in Refueling Procedure FP-CAW-R19 contained windows of vulnerability where indications of core temperature and vessel level were reduced and fc eriods that were unnecessarily long. The refueling sequences was reviewed and revised to optimize availability of level indication for operators.

(e) Work Control

Following the inadvertent drain down event on September 4, the licensee issued a complete stop work order for outage activities at Haddam Neck. The order was issued by the Unit Director on September 4, and was supplemented by a second order issued by the Nuclear Safety and Oversight Organization in accordance with nuclear generation procedure (NGP) 3.19. The intent of the order was to provide additional protection of the reactor vessel inventory and the key safety functions. The effect of the NSO order was to preclude the resumption of any activity without the prior review and approval by the oversight group.

c. Conclusions

Operators and plant staff showed poor sensitivity to the control of shutdown risk during the period from August 28 - September 1, as several events occurred which significantly reduced plant safety margins. The events included the inadvertent decrease in reactor vessel level during plant shutdown conditions, and the degradation in plant equipment needed for decay heat removal system challenges and equipment failures. For approximately four days, control room operators were unaware that nitrogen gas was leaking into the reactor vessel and causing level to decrease. By September 1, 1996, reactor vessel level had decreased to approximately three feet below the reactor vessel flange.

The series of events revealed several deficiencies in the conduct of operations. Procedures used for extended operation in Mode 5 were either lacking, poorly written or incomplete. Operators failed to follow procedures completely and pre-job briefs were not thorough. Operators made inappropriate decisions and did not show a questioning attitude. Poor material conditions were revealed in valve leakage through systems that interfaced with the RCS. Licensee corrective actions for past operational concerns were not adequate to preclude subsequent events. Potential enforcement actions associated with these concerns will be addressed as part of NRC Inspection 50-213/96-80.

The plant management and staff failed to appreciate the significance of the nitrogen bubble event and systematically under reacted to the event initially, resulting in delays in initiating an integrated event response. There were deficiencies in the quality of information developed as part of event briefings and completeness of communications regarding plans, contingencies and available procedures. There was an inadequate number and type of resources applied to plant problems and degraded equipment, and an inadequate integration of plant resources and support activities to effectively respond to degraded plant conditions in a recovery type of forum. The licensee review of a precursor event on August 28 was not thorough. Delays were encountered in establishing vessel level indication, and backup core cooling methods.

O1.3 Loss of Offsite Power Distribution Line

a. Inspection Scope (71707)

The inspection scope was to review licensee actions in response to a partial loss of the 115 KV distribution system.

b. Observations and Findings

On September 11, 1996, a fault occurred in a substation located offsite (Middletown, Connecticut) in the 115 KV distribution system. The fault resulted in the loss of line 1772 at 6:24 p.m., which is one of two 115 KV distribution lines that provides power to Haddam Neck. The loss of line 1772 resulted in the automatic switching of breakers and relay protective actions in the 115 KV system, and the receipt of control board alarms in the Haddam Neck control room. The operators implemented emergency operating procedure EOP 3.1-10, Partial Loss of AC, and SUR 5.1-153B to verify the status of station power supplies.

The impact of the electrical transient on the plant was minimal following a momentary interruption of power. Power is normally supplied to the plant distribution system by two independent transmission lines (1772 and 1206), which are cross-tied in the 115 KV switchyard by breaker 12R-1T-2. Protective relays are provided to detect faults and initiate automatic switching to isolate the fault and maintain power to the plant distribution system. The operators verified that this occurred on September 11. Following the loss of line 1772, the 12R-1T-2 tie breaker was opened to separate lines 1206 and 1772, tie breaker 2T3 was closed to cross tie Bus 1-2 with Bus 1-3, and the transformer low side supply breaker 12R-21S3-2 was opened. The switching scheme functioned as designed to maintain power on the site buses, and the power interruption was very brief such that neither the degraded grid voltage nor the emergency bus undervoltage setpoints were reached. The emergency diesel g nerators were not required and did not start.

Although the power transient on the site buses was of short duration, several 480 volt contactors opened causing the tripping of some loads supplied from Bus 1-2. The affected equipment included the momentary loss of some ventilation fans, and the operating spent fuel pool cooling pump, P-21-1A, which was powered from Bus

4 via MCC-2. The operators restored the affected equipment and the spent fuel pool cooling pump was restored, within 20 minutes of the initiating event, at 6:41 p.m. There was no significant change in spent fuel pool (SFP) bulk temperature. The inspector reviewed the operating specifications for the 480 volt contactors and the bus undervoltage protection schemes and noted that the equipment operated as designed.

The plant continued to meet the requirements of Technical Specification 3.8.1.2 during the event, which required a single offsite line and one emergency diesel generator be operable for operation in Mode 5. The event was significant in view of the plant status at the time. The reactor decay heat cooling was provided with the "A" residual heat removal (RHR) pump in service. The RHR system was degraded from the normal two train redundancy because the "A" RHR heat exchanger was out of service due to a through wall leak in the inlet isolation valve, and the "B" RHR pump was inoperable. Although the "B" emergency diesel generator was operable, it could only power the "A" RHR pump.

The licensee implemented additional contingency measures. Although diesel generator EG-7 had been installed and connected to Bus 1-2, it was not available to the operators on September 11 pending turn over from construction. The licensee took actions to clear the test related tagouts on EG-7, which made the diesel available as a backup power source for the "A" RHR pump at 1:04 a.m. on September 12. Procedures were available to use EG-7 to power Bus 1-2. Design engineering was requested to expedite the turnover process. The licensee also prepared a written contingency plan to document actions that have been taken or will be taken if electrical power supplies were further degraded. This plan included markups of existing emergency operating procedure (EOP) 3.1-10 to identify the expected flow path through the procedure network, and included the use of EG-7. Finally, the licensee recognized the need to evaluate electrical lineups that would allow supplying the RHR pumps from the alternate power train. This matter was the subject of an adverse condition report for work assignments as a long term corrective action.

The load dispatcher restored line 1772 at 4:46 a.m. on September 12, which returned two independent power feeds to Haddam Neck. The load dispatcher deferred to a later planned outage of line 1772 to further address the offsite substation problems.

c. Conclusions

The loss of an offsite distribution line resulted in a reduction in the independent power supplies for Haddam Neck, but did not impact the operation of equipment needed to cool the reactor. Plant equipment operated as expected during the transient, and the operators responded in accordance with plant procedures. Licensee contingency plans were appropriate to assure further actions to preserve core cooling would be taken if electrical supplies degraded further.

II. Maintenance

M1 Conduct of Maintenance

M1.1 LPSI Flow Test

a. Inspection Scope

The inspector observed the following surveillance activity:

- ST 11.7-193, Low Pressure Safety Injection System Integrated Flow Test (Original)

b. Observations and Findings

The inspector noted during a pre-test functional verification of the communication system, that the licensee appropriately postponed the low pressure safety injection (LPSI) system integrated flow test due to "quality" of the radio communications within various locations within the facility. The communications were changed to sound-power telephones and verified acceptable during a test run of the procedure.

During the test, operators appropriately entered into the technical specification requirements during temporary securing of shutdown cooling, and appropriately monitored reactor coolant system inventory during draindown, and refill operations with the low pressure safety injection pumps. Portions of ST 11.7-193 observed by the inspector (sections 6.2 through 6.5) were appropriately controlled by the shift engineer, the system engineer, and the management test lead.

The inspector independently confirmed selected procedural prerequisites such as: reactor coolant temperatures using core exit thermocouples, initial refueling water storage tank levels and temperatures, operability of pressurizer level indication and cavity level indication, and tracking of reactor coolant and containment integrity.

The test results were acceptable to gather flow measurements to support a design change.

c. Conclusions

The inspector noted appropriate control of the LPSI Integrated flow test was exhibited by operations and engineering personnel.

M1.2 Event Followup - Degraded Safety Equipment

The inspector reviewed maintenance and testing activities completed in support of the recovery from events involving a degradation in the reactor residual heat removal cooling function, and the presence of a nitrogen bubble in the reactor head.

a. Inspection Scope (62703)

The inspectors observed all or portions of the work activities listed below. The major focus of the inspection this period was to review the licensee test and maintenance activities conducted to restore systems required for shutdown core cooling.

- RHR valve 791A through wall leakage, August 31
- RHR Pump "B" failure to start on September 1
- Reactor Head Vent System Functional Test on September 5
- Determination of Reactor Vessel Level on September 5
- "B" RHR Pump Testing per ST 11.7-198 on September 15-16
- "B" RHR Pump Motor Testing & Evaluation on September 17

b. Observations and Findings

RHR Valve 791A

During operational rounds on August 31, a nuclear side operator (NSO) noted a through wall leak in the body of valve RHR-V-791A. The valve was the RHR inlet isolation valve for the "A" RHR heat exchanger. The valve was an 8 inch diameter double wedge gate valve, and a type 316 stainless steel casting manufactured by Alloyco. The through wall defect in the safety class 2 pressure boundary rendered that portion of the RHR system inoperable even though the heat exchanger remained functional for reactor decay heat removal.

The initial plans were to complete the evaluations and the non destructive examinations (NDE) needed to assure the continued integrity of the component and RHR piping. The NDE was completed after repair to the "B" RHR pump and a radiographic (RT) source was rigged down in to the RHR pit. The RT examination was completed on September 20. The examination did not identify any structurally significant defects. The casting appeared to be of good quality. No wall thinning in the vicinity of the defect was apparent. The licensee concluded that the defect was due to the lineup of casting shrinkage flaws. Based on the results of RT and liquid penetrant examinations, the license concluded that no flaws were present that would challenge the structural integrity of the valve or piping system. The licensee used these results as the basis of a relief request under 10 CFR 50.55a(g)(5)(3), that would allow classifying the valve as operable, but degraded. The licensee intended to place the "A" RHR system in service pending NRC approval of the relief request.

This action to submit the relief request was in progress at the conclusion of the inspection.

RHR Pump "B" failure

The operators attempted to start the "B" RHR pump on September 1 and discovered that the pump was mechanically bound. The pump could not be turned by hand and was determined to be mechanically bound. The motor was subsequently disconnected from the pump. The motor was found to turn freely and plant electricians subsequently meggered the motor with satisfactory results. The pump was disassembled for repair. The licensee continued with the repair of the "B" pump on September 4 after completing actions to restore vessel level indication.

The "B" RHR pump was successfully used to remove reactor decay heat when the plant was shutdown on July 24 for the maintenance outage. The licensee determined that the pump failed after it was shutdown on August 19, when it was run for about 45 minutes to reduce core exit temperatures. Upon disassembly, the licensee found that the casing and pump wear rings had excessive wear. The shaft sleeve was damaged in the area of the throttle and the disaster bushings. The throttle bushing tac welds were broken and the bushing had been displaced from its area of installed fit. The disaster bushing was worn around its circumference. The shaft was severely bowed.

The licensee's formal root cause evaluation was still in progress at the conclusion of the inspection. The preliminary results are described here. The cause of the failure was attributed to a combination of a marginal design and manufacturing defects in the rotating assembly. There was occasional contact between the shaft and the stationary throttle bushing at the impeller end of the shaft. After hours of operation with occasional contact, the throttle bushing tac welds broke allowing the bushing to move from its area of fit. The shaft deflected and started to rub during the pump start on July 30, 1996. The bushing eventually cocked and locked between the stuffing box and the shaft sleeve, presumably during the 45 minute run on August 19. The locked bushing cut the shaft sleeve, heated the shaft and caused severe bowing and shaft rubs. The rotor seized when the pump was stopped on August 19.

The licensee initiated a repair plan with two paths to success: a rebuild of the damaged rotating assembly; and, the installation a new rotating assembly using a new element available from plant stores. The licensee ultimately repaired the pump using the spare element; however, the licensee discovered significant problems in the spare assembly, which had parts with critical clearances that did not meet the manufacturer's tolerances. The spare assemble failed a receipt inspection on September 4, 1996 (ACR 96-979). The final restoration of the "B" RHR pump was delayed as the licensee obtained specifications from the pump vendor to rebuild the assembly in accordance with the recommended tolerances.

After replacing the pump, the licensee identified flaws in the pump motor casing which required further evaluation (see below). Delays were also encountered due to problems in establishing oil level in the pump bearings. The stationary oil baffle ring had to be replaced following the rebuild of the rotating assembly. The resolution of this problem revealed further dimensional problems with the pump parts. Further

delays were experienced as the licensee encountered problems during post repair testing in assuring that the pump casing was completely vented. Licensee actions to repair, test and restore the "B" RHR pump were in progress at the end of the inspection period.

"B" RHR Pump Motor Testing & Evaluation

Post maintenance testing was completed on the "B" RHR pump and motor following the repair and rebuild of the rotating assembly. The testing included the measurement of pump and motor vibration data, as well as the measurement of motor currents, which were evaluated as part of the motor current analysis program. Two anomalous conditions were identified during the testing conducted on September 16: the motor current analysis identified indications that were indicative of air gaps or eccentricity in the rotor; and, a crack several inches long was identified in the motor casing (ACR 96-1049). The crack was several inches long and was located on the west end of the motor in the lower end of the upper casing. Further examination identified a second crack in the lower end of the upper casing.

The initial plan to replace the "B" motor with a spare unit from the warehouse was abandoned when the licensee determined that the spare motor leads would have to be replaced to permit a proper termination of the motor leads with an environmentally qualified splice. The licensee disposition of these problems was summarized in a memorandum dated September 20, 1996 (CY-TS-096-0432), which provided the engineering evaluation of the deficiencies. The licensee concluded it was acceptable to use the installed "B" RHR pump motor. The cracks were determined to be pre-existing defects that were evaluated by the pump vendor and found acceptable during a previous overhaul. The cracks did not effect the function of the pump. The motor current data was evaluated in conjunction with vibration data as the pump was tested in several flow regimes. The results were not indicative of abnormal bearing movement or operation.

c. Conclusions

Licensee actions were in progress at the completion of the inspection period to restore equipment needed to restore margins and redundancy for the shutdown cooling function. As described above and in NRC Inspection 50-213/96-80, several performance deficiencies were identified in the licensee actions to correct conditions adverse to quality. Maintenance support for restoring inoperable decay heat removal equipment was poor. The availability of quality parts, vendor specifications, and repeat test failures resulted in excessive delays in restoring RHR to an operable status. Potential enforcement actions associated with these concerns will be addressed as part of NRC Inspection 50-213/96-80.

M3 Maintenance Procedures and Documentation

M3.1 Work Stoppage and Control of Outage Work

a. Inspection Scope

The purpose of this inspection was to review the licensee control of outage work activities following the inadvertent drain down event on September 4.

b. Observations and Findings

Following the drain down event on September 4, the licensee issued a complete stop work order for outage activities at Haddam Neck. The intent of the work stoppage was to provide additional protection of the reactor vessel inventory and the key safety functions. The effect of the order was to preclude the resumption of any activity without the prior site management review. The work hold applied to all station activities that had the potential to affect plant systems; thus activities to maintain the buildings and grounds, or other work in outside areas (for example, on the protected area perimeter) were allowed to continue. The only exception to the work stoppage inside the power block was the ongoing work to test or repair the RHR pumps and the RHR heat exchanger inlet valve.

The inspector reviewed the licensee's plans and controls for the phased resumption of outage activities. A primary side work ban would remain in effect until the "B" RHR pump was restored and the "A" RHR heat exchanger inlet valve flaw was dispositioned. A plan was developed to resume work activities in a controlled manner. All outage work orders were withdrawn from the field to the work control center. All work orders were reviewed against the shutdown risk management key safety functions (reactivity control, inventory control, decay heat removal, electrical supplies, and containment), which was used to rebuild the outage work schedule. Work was released in accordance with defined system out of service windows, and any window released for work was done with a process that included reviews by the work control center and operations (by an individual with a senior operator license), and approved by station management and the nuclear safety oversight organization. The site management approval was assured by the generation of an outage sequence change request (OSCR) for each activity released (with grouping according to outage windows allowed). The resumption of work was allowed using the following criteria:

- no work affecting key safety functions would be released until both trains of RHR were available for service, the reactor vessel level issues were resolved and the root cause investigation for the September 1 events were complete and corrective actions were taken.

- other outage work would be released at a pace to allow a slow resumption of work activities.

The licensee's plan was described in a letter to the NRC dated September 5, 1996. Management expectations on the resumption of outage work were addressed in a letter (NUD-96-44) to the Haddam Neck staff dated September 6, 1996. The inspector reviewed the implementation of the augmented work control plan and found it was completed as described in the September 5 letter. The inspector reviewed the resumption of work activities on September 6 and 7 that were issued as part of eleven "system out-of-service" windows (01, 03, 39, 51, 54, 55, 56, 58, 59, 66, 69), which allowed the resumption of activities on the turbine and secondary plant.

c. Conclusions

Following the loss of control of outage work activities resulting in an inadvertent drain of the reactor vessel on September 4, the licensee implemented a system of controls that were adequate to protect key safety functions and allow the gradual resumption of outage work.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) LER 96-005, Degraded Spent Fuel Cooling System

This licensee event report (LER) described an event whereas the spent fuel pool cooling system was shut down due to the discovery of loose parts. This event was discussed in Inspection Report 50-213/96-02 detail 4.4. No new issues were revealed by the LER.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Special Nuclear Material Inventory Controls (UNR 96-10-01)

a. Inspection Scope

The purpose of this inspection was to review the licensee's process for the control of special nuclear materials per 10 CFR 70.51

b. Observations and Findings

On September 3-4, 1996, the licensee completed a physical inventory of special nuclear material (SNM) in the spent fuel pool in accordance with procedure SNM 1.4-11. The serial numbers (SNs) for 15 fuel assemblies could not be read during the inventory because accumulated debris caused by the rerack job covered the numbers. The inventory was completed by the reactor engineering group. The physical inventory was signed off as complete and approved on September 4, 1996. The licensee issued ACR 96-1046 on September 13, 1996 to document the inability to read the serial numbers, and issued a request to vacuum the affected a bundles at a later date.

The inspector noted that these actions were not consistent with the requirements of SNM 1.4-11, Step 6.2.3 which required the technician to record the fuel assembly serial number in each fuel location using an underwater camera. When questioned how this step was completed for the affected fuel assemblies, the technician stated that the inability to read the SNs was resolved in accordance with step 6.3.1 which requires that the results of the present inventory be compared to the previous inventory and for the technician to resolve any discrepancies. The technician "resolved" the differences by using the refueling tracking board and the SNM transfer records to "identify" which assembly was in the storage locations.

The inspector discussed this matter and concerns regarding how the inventory was completed with the engineering manager for Reactor Engineering. The manager reviewed the inventory activities and identified the following concerns:

- the technicians did not follow the requirements of SNM 1.4-11 when completing the physical inventory;

- 10 CFR 70.51 as implemented by SNM 1.4-11 requires that a physical inventory of spent fuel be conducted at an interval not to exceed 12 months. Contrary to this requirement, the 1996 inventory was due by April 1996, but was not completed until September. The inventory was completed within 45 days following the rerack project which had resulted in a major movement of fuel.

These discrepancies were described in an ACR dated September 20, 1996. The followup actions were described in memorandum CY-TS-96-455 dated September 26. The licensee confirmed that the subject irritated bundles met the burnup requirements for storage in the newly configured spent fuel racks. The licensee planned to verify the affected serial numbers by lifting each fuel assembly and reading the SN on the side of the top nozzle. This would be done after the core offload is completed. The affected fuel assemblies would be vacuumed to remove the debris.

The licensee short term actions to address the failure to follow SNM inventory controls were adequate. However, the licensee stated that although the requirements of SNM 1.4-11 were not met, the SNM physical inventory requirements of 10 CFR 70.51 and CFR 74.31 were met through the conduct of a piece count and without the recording of SNs. In support of this position, the licensee polled SNM inventory practices at 10 other reactor plant sites and noted that a piece count is used.

The inspector noted that 10 CFR 70.51 requires that a physical inventory of spent fuel at fixed reactor sites be conducted at intervals not to exceed 12 months. Section 70.51(a)(8) defines a physical inventory to be a determination on a measured basis of the quantity of SNM on hand at a given time. The methods for conducting a physical inventory are set out in Section 70.51(f). Section 50.71(f) requires the licensee to establish physical inventory procedures to assure that (i) the quantity of SNM associated with each item on inventory is a measured value; and,

(ii) each item on inventory is listed and identified to assure all items are listed and that no item is listed more than once. Finally, CFR 74.4 defines an item to be any discrete quantity of SNM having a unique identity and an assigned element and isotope quantity. Each spent fuel assembly at a fixed reactor site has a unique identity and an assigned element and isotope quantity, as reported by the licensee periodically in the SNM material status reports (NRC Form 742).

Based on the above, the inspector concluded that the requirements of 10 CFR 70.51 for a physical inventory of spent fuel at a reactor site could not be satisfied solely on the basis of a piece count and further, that the regulations would require the recording of fuel assembly serial numbers. The inspector noted that the industry defined criteria for the conduct of physical inventories of irradiated fuel at reactor sites in American National Standards Institute (ANSI) Standard N15.8-1974, Nuclear Material Control Systems for Nuclear Power Plants. Section 6.4.2 of the standard states that a piece count, together with a verification of assembly serial numbers shall be taken during physical inventories. The licensee is not committed to the requirements of ANSI N15.8.

The inspector discussed this issue at the exit interview on September 27, 1996. The inspector described the NRC position that fuel assembly serial numbers and a piece count should be recorded to complete a physical inventory in accordance with 10 CFR 70.51. The Unit Director acknowledged the inspector's comments but concluded, based on a position developed by nuclear licensing, that only a piece count is required. This matter is considered unresolved pending (i) confirmation of licensee corrective actions as described above; and, (ii) further review by NRC management **(UNR 96-10-01)**.

c. <u>Conclusions</u>

The engineering support for the control of special nuclear materials was inadequate, as demonstrated in the failure to follow SNM inventory procedures and in the failure to complete timely physical inventory of irradiated fuel in the spent fuel pool. The housekeeping and cleanliness conditions in the spent fuel pool were poor and unacceptable to support the conduct of SNM inventory checks. The licensee position on the recording of fuel bundle serial numbers in addition to completing a piece count while performing an SNM inventory in accordance with 10 CFR 70.51 is a matter that requires further NRC review.

E8 Miscellaneous Engineering Issues (92902)

E8.1 <u>Spent Fuel Rack Materials</u>

a. <u>Scope (37751)</u>

The scope of this inspection was to review the design documents for the spent fuel storage racks in use in the Haddam Neck spent fuel pool. The review was performed to determine the type of materials used in the construction of the racks and what was credited for the analysis of rack structural integrity. The findings

below were based on interviews with licensee personnel and information in the licensee design specifications and the NRC safety evaluation.

b. Observations and Findings

CY installed new storage racks in the spent fuel pool in 1996 as part of a design change completed under Plant Design Change Record (PDCR) 1592, and thereby increased the storage capacity from 1168 cells to 1480 cells. The prior SFP design was approved by the NRC as part of license Amendment No. 7. The changes completed under PDCR 1592 included the removal of 30 of the 75 existing racks, and the installation of 13 new high density racks. The 13 new racks were the "free standing" type which do not require lateral restraints. The remaining 45 older racks retained in the pool have lateral seismic restraints, and are tied together through inter-ties. There is no physical interconnection between the old and new racks. The free standing modules are stable against a seismic event that is 150% of the postulated safe shutdown earthquake load.

The PDCR 1592 design specifications require the racks structural members meet the limits in Section III of the 1992 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The rack modules are designed as cellular structures such that each fuel assembly rests on a flat horizontal load bearing surface, with lateral support provided by the side walls of the cell. The structural materials used to fabricate the spent fuel racks are stainless steel, including type ASME SA240-304 and ASME SA564-630, which is a hardened stainless steel (reference Rack Purchase Specification SP-ME-909 and PDCR 1592). The racks also use Boral as a neutron absorber material, but this material is not used for structural integrity. Boral is composed of boron carbide (fine granulated powder) and 1100 alloy aluminum. Boron carbide is homogeneously dispersed throughout the central layer of the boral panels. The boral panels are welded to the stainless steel sheet metal side walls of the module cells. The boral material is not credited for the structural integrity of the racks.

The spent fuel racks, pool and building structure were analyzed to demonstrate structural adequacy as required by NRC requirements. The pool structural loads were evaluated using combinations of discrete loads, including dead weight of racks and fuel assemblies, hydrodynamic loads, thermal loads, rack support dynamic loads and seismic loads. The rack mechanical structures were evaluated to include loads from seismic, thermal, and mechanical forces. The loads and load combinations and acceptance criteria were based on the ASME code and the standard review plan (SRP) NUREG-0800 Section 3.8.4. The NRC staff approved the rack design and analytical methodologies as part of License Amendment 188 dated January 22, 1996 (reference the NRC Safety Evaluation Sections 2.6 and 2.9).

c. Conclusions

No inadequacies were identified regarding the materials of construction of the new racks or in the analytical approach to assure adequate material strength.

E8.2 (Closed) LER 96-09: Loss of Component Cooling Water

The licensee reported in LER 96-09 that procedure AOP 3.2-10 was inadequate in that had the procedure been implemented, it could have resulted in plant operation outside of the design basis. Procedure section 4.5 for a loss of component cooling water provided a method for the operators to align service water to the RHR heat exchangers by cross tieing the service water and the component cooling water systems. The licensee determined that if an accident occurred while in the alignment allowed by the AOP, then the post accident sump recirculation function could be compromised. The configuration described in the AOP (and LER) never existed in the plant.

The corrective actions were to change AOP 3.2-10 (TPC 96-238 dated 4/30/96) to delete the section that would cross tie the service water and component water systems. The cause of the event was inadequate procedure review in the failure to consider the occurrence of an accident while in the configuration. Weaknesses in the procedure review process and in translating the design basis into procedures and practice were previously identified problems that were being addressed by the licensee's configuration management program. No new issues were revealed by the LER. This item is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Health Physics Controls for Shutdown Operations

a. Inspection Scope (83729)

The scope of this inspection was to review radiation protection controls for operating activities.

b. Observations and Findings

The inspector observed the implementation of selected portions of the licensee's radiological controls program during routine inspections of the accessible plant areas. The inspector observed activities for compliance with radiation work permits (RWPs) to assure that descriptions of radiological conditions were appropriate as compared to posted conditions and that personnel adhered to RWP requirements. Access to various radiologically controlled areas and the use of personnel monitors and frisking methods upon exit from those areas was also observed.

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. This review included inspector measurement of radiation source terms within the radiologically controlled areas using licensee survey instruments. The inspector reviewed licensee controls to move a barrel of sludge removed from the containment to a radwaste storage area on August 23. The sludge was obtained as part of activities to decontaminate portions of the refueling canal. The barrel contact radiation levels as high as 100 R/hour and was appropriately marked to inform workers of the radiological hazards. The licensee took appropriate measures to reduce the source terms and to minimize the exposure to plant workers during transport of the barrel within the plant yard.

The implementation of the physical security program was reviewed during inspection tours of the plant. The security controls for the access to the protected and vital areas were maintained.

c. Conclusions

Radiological and security controls were maintained in accordance with licensee and regulatory requirements.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on September 27, 1996. 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.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Jere LaPlatney, Unit Director
Gerry Waig, Maintenance Manager
Jack Stanford, Operations Manager
James Pandolfo, Security Manager
Ron Sachatello, Radiation Protection Manager
Tom Cleary, Nuclear Licensing Engineer

NRC

Stephen Dembek, Haddam Neck Project Manager

INSPECTION PROCEDURES USED

IP 40500:	Effectiveness of Controls in Identifying, Resolving, and Preventing Problems
IP 62703:	Maintenance Observation
IP 71707:	Plant Operations
IP 73051:	Inservice Inspection - Review of Program
IP 83750:	Occupational Exposure
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events
IP 92902:	Followup - Engineering
IP 92903:	Followup - Maintenance
IP 93702:	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPEN, CLOSED, AND DISCUSSED

Open

96-10-01	UNR	Completion of Recovery Actions Prior to Entry into Mode 6
96-10-02	UNR	Recording Serial Numbers for a Physical Inventory per 10 CFR 70.51.

Closed

None

Discussed

None

LIST OF ACRONYMS USED

ACR	Adverse Condition Report
ANSI	American National Standards Institute
AOP	Abnormal Operating Procedure
ASME	American Society of Mechanical Engineers
AWO	Authorized Work Orders
Ci	Curie
CLIS	Cavity Level Indication System
CYAPCo	Connecticut Yankee Atomic Power Company
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
F	fahrenheit
GL	Generic Letter
gpm	gallons per minute
IRT	Independent Review Team
LER	Licensee Event Report
LPSi	Low Pressure Safety Injection
NDE	Nondestructive Examinations
NGP	Nuclear Generation Procedure
NOP	Normal Operating Procedure
NRC	Nuclear Regulatory Commission
NSO	Nuclear Side Operator
OSCR	Outage Sequence Change Request
PAB	Primary Auxiliary Building
PDCR	Plant Design Record
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RVLIS	Reactor Vessel Level Indication System
RWPs	Radiation Work Permits
RWST	Refueling Water Storage Tank
SE	System Engineer
SNM	Special Nuclear Material
SNs	Serial Numbers
SRP	Standard Review Plan
SUR	Surveillance Procedure
TS	Technical Specification
VCT	Volume Control Tank
WCC	Work Control Center