U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-213/85-19

Docket No. 50-213

License No. DPR-61

Licensee: Connecticut Yankee Atomic Power Company
P. O. Box 270
Hartford, CT 06101

Facility: Haddam Neck Plant, Haddam, Connecticut

Inspection at: Haddam Neck Plant

Inspection conducted: September 4 - October 15, 1985

Inspector: Paul D. Swetland 10/31/85
Paul D. Swetland, Senior Resident Inspector Date Signed

Approved by: E C McCabe 10/31/85
E. C. McCabe, Chief, Reactor Projects Section 3B, Division of Reactor Projects Date Signed

Summary: Routine resident inspection (99 hours) of plant operations, radiation protection, physical security, fire protection, maintenance and surveillance testing, previous inspection findings, and events occurring during the inspection.

No violations were identified. Five outstanding inspection items were closed. Follow-on inspection action was opened for review of licensee corrective actions responding to an unexpected gaseous radioactivity release on September 19, 1985.

DETAILS

1. Review of Plant Operations

The inspector observed plant operation during regular tours of the following plant areas:

-- Control Room	-- Security Building
-- Primary Auxiliary Building	-- Fence Line (Protected Area)
-- Vital Switchgear Room	-- Yard Areas
-- Diesel Generator Rooms	-- Turbine Building
-- Control Point	-- Intake Structure and Pump Building

Control room instruments were observed for correlation between channels and for conformance with Technical Specification requirements. The inspector observed various alarm conditions which had been received and acknowledged. Operator awareness and response to these conditions were reviewed. Control room and shift manning were compared to regulatory requirements. Posting and control of radiation and high radiation areas was inspected. Compliance with Radiation Work Permits and use of appropriate personnel monitoring devices were checked. Plant housekeeping controls were observed, including control and storage of flammable material and other potential safety hazards. The inspector also examined the condition of various fire protection systems. During plant tours, logs and records were reviewed to determine if entries were properly made and communicated equipment status/deficiencies. These records included operating logs, turnover sheets, tagout and jumper logs, process computer printouts, and Plant Information Reports. The inspector observed selected aspects of plant security including access control, physical barriers, and personnel monitoring.

a. Plant Operations Review Committee (PORC)

The inspector attended the Plant Operations Review Committee (PORC) meeting on September 6, 1985. Technical specification requirements for required member attendance were met. The meeting agenda included a review of Revision 1 to the design change package for replacement of the plant process computer (PDCR-713). The meeting was characterized by frank discussions and questioning of the proposed change. All questions were satisfactorily resolved and the PDCR was approved. The inspector had no further comments.

2. Observation of Maintenance and Surveillance Testing

The inspector observed various maintenance and problem investigation activities for compliance with requirements and applicable codes and standards, QA/QC involvement, safety tags, equipment alignment and use of jumpers, personnel qualifications, radiological controls, fire protection, retest, and reportability. Also, the inspector witnessed selected surveillance tests to determine whether properly approved procedures were in use, test instrumentation was properly calibrated and used, technical specifications were satisfied, testing was performed by qualified personnel, procedure details were adequate, and test results satisfied acceptance criteria or were properly dispositioned. The following activities were reviewed:

-- Reactor Trip Breaker Testing (Preventive Maintenance procedure 9.5-40) on September 27, 1985.

-- Emergency Core Cooling Tests (Surveillance procedure 3.1-4) on October 2, 1985.

-- Auxiliary Feedwater Initiation Tests (Surveillance procedure 9.2-45) on September 11 and 16, 1985.

3. Followup on Previous Inspection Findings

During the course of the inspection, four NRC open items were reviewed. The inspector found licensee actions with regard to these areas to be sufficient to close these items. Details follow:

3.1 (Closed) Followup item (213/84-03-02) Applicable operating experience information was transmitted to training without verification of its inclusion as initial or re-qualification training material. The licensee was to evaluate measures for control and followup for operations critical information implemented through operator training. In conjunction with ongoing upgrades and accreditation in the training department, the licensee has initiated task oriented training goals. Under this new system, training is conducted to provide appropriate knowledge and skills to perform a specified set of formalized tasks. The licensee has determined that control of new operating experience information will be accomplished through the formal task review/change process. Thus new information can be included and tracked by new or revised training tasks and implemented under existing training department task verification processes. Operations critical information will be transmitted to the training department, and the task action response will be followed by the licensee's controlled routing system. The inspector had no further questions in this area.

3.2 (Closed) Followup Item (213/84-28-06) Inadequate preventive maintenance instructions resulted in the failure of several power distribution circuit breakers. The licensee was to revise the preventive maintenance procedure for these breakers to include inspection and cleaning of sticky closing coil hold-in (X) relays which caused intermittent breaker operations. Revision 6 to procedure 9.5-16, 50DHP-250 Breakers was implemented on August 29, 1985. The inspector verified that instructions for removal, inspection, cleaning, and replacement of the breaker X-relays had been included. This item is closed.

3.3 (Closed) Violation (213/85-03-01,02, and 03) Inadequate calibration procedure acceptance criteria resulted in plant operation with non-conservative loss of flow trip setpoints. The licensee was to revise the applicable calibration procedures, implement a program for monitoring normal loop differential pressure (dp) and screen all safety-related surveillance procedures to determine that no other faulty acceptance criteria existed. Procedures 5.2-23 and 5.2-3 were revised on March 22, 1985, to clarify that the loss of flow setpoints must be calculated using the currently measured value of loop dp. In addition, the three loop operation procedure (NOP 2.2-3) was revised to specify that the reactor protection system (RPS) loss of flow setpoints must be reset for normal 3-loop dp when operating in this mode. The licensee's program for monitoring the drift of loop dp was detailed in NRC Meeting Report 85-05. The inspector verified that this weekly monitoring activity had been incorporated in the licensee's automated planned maintenance system. The licensee also completed the screening of other safety-related surveillance procedures. Only one other procedure (RPS high main steam flow trip) had similar acceptance criteria. The inspector verified that the actual setpoints for high main steam flow were set conservatively with respect to the Technical Specification limits. The licensee has similarly controlled the calibration setpoint for the steam flow instruments in procedure 5.2-38, Steam Line Break Channel Calibration. The inspector had no further questions in this area.

3.4 (Closed) Unresolved Item (213/85-07-01) The licensee reported a fire protection system wiring discrepancy and the failure of previous, required surveillance tests to identify this problem. The licensee was to clarify the surveillance requirement for separate verification of fire damper and ventilation fan actuation. Also, the licensee was to resolve a difference between the configuration of the Cable Vault Ventilation System and Fire Hazards Analysis (FHA) description of this system. The licensee determined that the FHA description of the cable vault ventilation system was incorrect. The referenced exhaust duct damper is not installed in the system. However, because the exhaust duct vents from the top of the cable vault, and because the CO_2 released by the fire protection system is heavier than air, there is no motive force which would dilute the CO_2 concentration after the fans shutdown. Therefore, the absence of the exhaust duct damper does not prevent inerting the cable vault atmosphere. The conclusions of the FHA remain valid. The licensee revised surveillance procedure 5.5-20, Cable Vault CO_2 System... Test, on August 10, 1985, to include specific guidance on the verification of the fan and damper actuations separately. The inspector had no further questions in this area.

3.5 (Closed) Unresolved Item (213/85-15-03) Licensee actions did not effectively correct an identified deficiency in the control room fire detection system. The licensee was to conduct training for plant engineers regarding correct implementation of quality assurance program corrective action procedures. The licensee conducted corrective action program training sessions on September 17 and 24. The inspector reviewed the lesson plans and attendance records and discussed the presentation content with selected individuals. The specified objectives were verified to have been accomplished. In addition, these training sessions included a discussion of post-modification testing requirements and philosophy as committed to by the licensee in NRC Region I Meeting Report 50-336/85-26 regarding testing problems at Millstone Unit 2. The inspector had no further questions in these areas.

4. Followup on Events Occurring During the Inspection

4.1 Licensee Event Reports (LERs)

The following LERs were reviewed for clarity, accuracy of the description of cause, and adequacy of corrective action. The inspector determined whether further information was required and whether there were generic implications. The inspector also verified that the reporting requirements of 10 CFR 50.73 and Station Administrative and Operating Procedures had been met, that appropriate corrective action had been taken, and that the continued operation of the facility was conducted within Technical Specification Limits.

85-18 -- Inoperable Fire Door

85-20 -- Potential Unauthorized Access to A High Radiation Area

85-21 -- Cable Spreading Area Fire Barrier Problems

85-22 -- Inoperable Fire Barrier

4.2 Safety Analysis Uncertainty Associated with Misaligned Control Rods

During a review of the next reactor core reload (Cycle 14) design and safety analyses on September 16, 1985, the licensee identified that the effect of the allowable control rod misalignment on appropriate safety analyses had not been reverified as required since fuel cycle 11. The plant was operating at full power with all control rods aligned to within 8 steps of the bank position indication. The total allowable rod misalignment is accounted for in safety analyses using a 5 percent peaking penalty in the axial power distribution calculations. Thus, the worst local hot spot in the core is increased by 5% in safety analysis calculations to account for allowable rod misalignment. Local peaking in the core is most severe early in core life. The present core has 291 effective full power days (about 86% of core life) expended. Therefore, the actual core peaking expected during the remainder of this fuel cycle will be less than the safety analysis assumed worst case values. In addition, the licensee committed to the following interim actions to maintain closer control of core peaking factors to insure no core thermal limits could be exceeded until complete re-evaluation of the core peaking calculations is completed. These actions included:

a. Maximum allowable negative axial offset was reduced administratively by 15%.

b. Quadrant power tilt ratio monitoring was increased from a weekly to a daily frequency and is limited to less than 2 percent deviation.

c. Maximum allowable rod misalignment was reduced administratively to ± 16 steps from the rod bank position.

The inspector verified that these actions had been implemented in plant procedures using temporary procedure changes for procedures 5.3-23 Excore-Incore Axial Offset Correlation; 2.2-2, Steady State Operations and 5.1-26, Incore Power Distribution Monitoring; respectively. In addition, plant operators were appraised of the identified analysis problems by memorandum dated September 16, 1985. The licensee has completed the re-evaluation of rod misalignment penalties and has determined that the 5% assumed penalty provides adequate margin for all plant operation after 200 EFPD using only the current technical specification limits (± 40 steps). Therefore, the more restrictive administrative controls have been suspended. Since no plant operation prior to 200 EFPD occurred with rod misalignment approaching the TS limit, the licensee has stated that no actual operation in an unanalyzed condition occurred. The licensee promptly notified NRC of this potential unanalyzed condition identified on September 16, 1985. The licensee is taking action to correct the rod misalignment penalty problems in the fuel cycle 14 analyses and to establish controls to insure that all specified fuel design parameter assumptions are properly verified in all subsequent reload submittals. NRC licensing is following licensee actions in this area to support NRC approval of the fuel cycle 14 reload package. The inspector had no further questions at this time.

4.3 Unexpected Gaseous Radioactivity Release

During preparations for maintenance on the reactor coolant valve stem leakoff header pressure control valve on September 19, 1985, a portion of this system including the pressure control valve (DH-PV-1170) and the leakoff header cooler were isolated and vented. These components are located in the pipe trench area in the primary auxiliary building which is ventilated through high efficiency particulate and charcoal filters prior to release directly to the plant vent stack (PVS). The isolated vent header components were vented directly to the pipe trench area. Control room operators had prepared for and approved the potential release expected from this short duration, low-level venting activity. Upon commencement of the venting, the PVS radiation monitor rose sharply to a 7000 microcuries per second release rate. The venting activity was ordered to be terminated, because this release rate exceeded the expected and reportable release activities. The release was terminated about ten minutes later by closing another header isolation valve and reconnecting the vent fitting. The licensee determined that the source of the higher than expected release was the on-service waste gas decay tank which provides cover gas to the primary drain tank. The gas leaked backward through two check valves and a closed vent header isolation valve into the vent header cooler which was being vented. The unexpected release terminated when a second valve in this flow path was closed. The licensee calculated that 5.5 Curies of primarily noble gases were released to the environment. While the actual release rate was only 13% of the allowable Technical Specification limit, the calculated release temporarily exceeded the maximum permissible concentration at the site boundary and represented a potential worst case dose of 0.1 millirem if an individual had been downwind of the release on the site boundary. The inspector reviewed the licensee's documentation and calculations for this release. No discrepancies were identified. The event was properly reported to NRC and state officials. Upon isolation of the vent cooler with a second header valve, the original maintenance activity was completed without further incident. The leaking isolation valve identified by this event has been added to the outage maintenance work package. Inspector review of the isolation scheme specified for the original repair included only one closed manual isolation valve between the vent path and the low pressure waste gas system. Although plant operators generally use a double valve isolation policy for maintenance activity where possible, this policy is not formalized and was not used in this case against the low pressure waste gas system. The licensee's failure to employ two valve isolation in this area resulted in an unnecessary release of fission product gases to the environment. The inspector brought this concern to the licensee's attention. The licensee will review the need for new guidance in this area during routine followup of this event. NRC will follow the applicable licensee corrective actions during a subsequent inspection. (IFI 50-213/85-19-01)

4.4 Licensee Response to Hurricane Gloria

On September 27, 1985, Hurricane Gloria swept up the eastern seaboard of the United States making landfall at Long Island, New York and Southern Connecticut. The licensee and NRC tracked the path of the hurricane and accurately predicted the projected path and potential consequences. Preparations for the storm were satisfactorily implemented including site inspection, cleanup and securing, emergency power source testing, fuel top-off, and stationing of two emergency response organization shifts onsite. The licensee decided, conservatively, to shut down the plant shortly before the arrival of the storm. The plant shutdown at 8:20 a.m. on September 27, 1985. The hurricane passed to the west of the site between 1:00 and 3:00 p.m. on September 27. Peak winds experienced onsite were about 75 mph with mild precipitation. The inspector toured the facility during the storm. No site damage was experienced other than the buckling of a turbine building roll-up door. Offsite power was maintained throughout the storm. The licensee activated the emergency response organization at the alert level due to potential winds exceeding the site design basis levels (115 mph). By 4:00 p.m. September 27, the storm degraded and passed the site, and the licensee terminated the emergency condition. Plant startup began, and after testing and chemistry conditioning holds, the plant returned to power on September 28, 1985. The inspector had no further questions in this area.

5. Review of Periodic and Special Reports

Upon receipt, periodic and special reports submitted pursuant to Technical Specification 6.9 were reviewed. This review verified that the reported information was valid and included the NRC required data; that test results and supporting information were consistent with design predictions and performance specifications; and that planned corrective actions were adequate for resolution of the problem. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following periodic reports were reviewed:

-- Monthly Operating Report 85-08
-- Monthly Operating Report 85-09

This report covered plant operation during the period August 1, 1985 to September 30, 1985.

6. Exit Interview

During this inspection, meetings were held with plant management to discuss the findings. No proprietary information related to this inspection was identified.