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REGION I

Report No. 50-213/88-02

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 Neck, Connecticut

Inspection at: Haddam Neck Plant

Inspection dates: January 13 - February 29, 1988

Inspectors: Andra A. Asars, Resident Inspector
John T. Shedlosky, Senior Resident Inspector

Approved by: E. C. McCabe, Jr. 3/11/88
E. C. McCabe, Chief, Reactor Projects S· .ion 1B Date

Summary: Inspection 50-213/88-02 (1/13-2/29/87)

Areas Inspected: This was a routine safety inspection (191 hours) by the resident inspectors. Areas reviewed included outage activities, radiation protection, fire protection, security, maintenance, surveillance testing, events occurring during the inspection period and preparations for start-up.

Results: No violations were identified and no unresolved items were opened.

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DETAILS

1. Summary of Facility Activities

During this inspection period the licensee completed modifications to the core barrel and thermal shield. Preparations for restart were also initiated including reperformance of selected surveillance tests. Tests selected included those which would come due before the next outage and those more complex tests which involve interaction of many plant systems. Reactor core reload began on February 20 and was completed on February 22. Mode 5 was entered on February 27.

2. Plant Operations

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

- Control Room
- Primary Auxiliary Building
- Vital Switchgear Room
- Diesel Generator Rooms
- Control Point
- Security Building
- Fence Line (Protected Area)
- Yard Areas
- Turbine Building
- 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. In addition to normal working hours, the review of plant operations was conducted during the following midnight shifts, weekends, and holidays:

- January 30, 1988, 2:00 PM to 4:00 PM
- February 15, 1988, 6:00 AM to 10:00 AM
- February 28, 1988, 12:00 Noon to 3:30 PM

No unacceptable conditions were identified. Operators were alert and displayed no signs of inattention to duty or fatigue.

3. Plant Operations Review Committee (PORC)

The inspector attended several Plant Operations Review Committee (PORC) meetings. Technical specification 6.5 requirements for required member attendance were verified. The meeting agenda included procedural changes, proposed changes to the Technical Specifications and field changes to design change packages. The meeting was characterized by frank discussions and questioning of the proposed changes. In particular, consideration was given to assure clarity and consistency among procedures. Items for which adequate review time was not available were postponed to allow committee members time to review and comment. Dissenting opinions were encouraged.

The inspectors have noted that, immediately following the daily morning meeting, a short meeting with the department supervisors is routinely held to review Plant Information Reports (PIRs). PIRs are a means of informing station management of any station conditions, significant equipment failures, or events. This includes matters which may be reportable to NRC. PIRs are generally reviewed by management within one working day of occurrence. At this time reportability to NRC and the need for follow-up or additional actions are evaluated. The inspectors have observed thorough reviews of PIRs and also noted that the PIR process has proven an effective method of bringing management attention to station conditions.

4. 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:

4.1 Reactor Core Support Barrel Thermal Shield Repair

The repair and modification to the reactor core support barrel thermal shield attachments was completed during this inspection period. A set of six keyways were installed to the upper rim of the thermal shield. This modification was made to limit the thermal shield tangential motion which was responsible for the previous failure of the thermal shield to core support barrel fasteners. These activities have previously been discussed in NRC Inspection Reports 50-213/87-25, 87-27 and 87-31. Additionally, the licensee has provided information on the repair program by letter dated February 25, 1988 (Serial B12808).

The inspectors observed the work on the core support barrel assembly on a day to day basis during the inspection period. This included coverage of diving evolutions and milling machine operation. The inspectors found that the work activities conformed with the procedures, extensive Quality Assurance surveillance coverage was conducted and the work received a high level of licensee management oversight.

The underwater diving made to perform manual evolutions required by the repair process was of special interest to the inspectors. Radiological protection of diving personnel was initially examined in NRC Inspection 50-213/87-25 during the initial dives. Inspection coverage of diving evolutions continued through the current inspection period. The inspectors routinely observed pre-dive surveys of the work area, the placement of barriers which prevented the diver from leaving the designated work area, the placement of personnel dosimetry, the pre-dive briefings, radiological safety and remote dosimetry monitoring during the dives, and the control of contamination as divers or tools and equipment exited the reactor cavity pool.

The inspectors found that this specialized aspect of the Radiological Protection Program remained strong throughout the period of the extended outage. During each dive observed there was good coordination between those directing the work evolution, the dive support personnel and the supervising Health Physics Technician. In every instance it was obvious that prime responsibility for diver safety rested with the diver support personnel and that radiological safety rested with the station personnel. There were no unacceptable conditions identified.

4.2 Spent Fuel Building Sluice Gate Repairs

During preparations for core reload, the licensee identified that the spent fuel pool sluice gate valve would not fully close easily due to interference between the valve shaft and shaft guide bushings. The sluice gate valve is on the Fuel Building side of the fuel transfer tube and is used for isolation of the spent fuel pool from the refueling cavity should there be a loss of cavity inventory. The valve is a Chappman gate valve with a long shaft which is guided by three bushing assemblies attached to plates in the spent fuel pool liner. This gate valve is air operated to open and close, however, under emergency situations, it is designed to gravity close. During refueling preparations with the transfer tube gate valve (valve on the containment side of the transfer tube) closed, the licensee had cycled the sluice gate valve several times. It was during these evolutions that the licensee identified that the valve would not close unassisted. Underwater videos of valve operation were taken using the mini-submarine utilized for the core support barrel thermal shield repairs. From these videos, the licensee determined that the valve shaft was slightly bent and its movement was restricted by the lower bushing assembly. Through the Jumper/Lifted Lead/Bypass process, the licensee analyzed and PORC approved the disengaging of the lower bushing assembly. This assembly was then lowered

to rest on the valve gate. With this bushing bypassed, it is possible for the shaft to bend further if the valve were to be closed by air. To prevent further shaft degradation, the licensee hung, on the valve controls, a sign stating that the valve is to be closed solely by gravity. The valve was successfully stroke tested several times; air to open and gravity to close. Inspector review of these activities and the bushing disengagement identified no deficiencies.

4.3 Cable Vault CO2 System Testing

On February 23, the inspectors observed licensee conduct of SPL 10.7-336, Containment Cable Vault CO2 Suppression System Concentration Test. The purpose of this test was to demonstrate that the cable vault CO2 system conforms to the original design requirement of providing a CO2 concentration of 30% within two minutes and 50% within seven minutes on both levels in the vault. This test was conducted because the licensee was unable to obtain original station records detailing previous test results. During test performance, extensive personnel protective actions were taken due to the danger involved with this test. The test was performed successfully, however evaluation of test data indicated that the CO2 concentrations in the lower level fell slightly below the acceptance criteria. Because of this, the CO2 system was declared inoperable on February 27. In accordance with Technical Specification (TS) 3.22.B.2, a continuous fire watch was established. The TS also requires that a backup fire suppression system be provided; the licensee has provided a local hose station in the Service Building hallway and can manually discharge both the normal and backup CO2 banks should there be a fire. Currently, the licensee is evaluating methods to ensure a sufficient concentration of CO2 is provided. These corrective actions will be reviewed during future inspections.

5. NRC Information Notice 87-44, Thimble Tube Thinning in Westinghouse Reactors

This Information Notice (IN) was issued to alert licensees to potential problems with thimble tube thinning in the area between the lower core plate and the bottom of the fuel assembly guide tubes. The licensee's actions in response to this notice were previously discussed in NRC Inspection Report 50-213/87-31.

In response to this notice the licensee performed eddy current testing (ECT) on the thimble tubes. Three tubes were identified as having wear in the area of concern. Two tubes displayed wear of 20% and one had wear of 30%. The wear has been characterized as localized pitting which is different from the fretting observed at other nuclear plants. The licensee has elected not to take any immediate actions based on these observations. However, ECT will be performed on the tubes in the next refueling outage. At that time, the new data will be evaluated and compared to the observations from this outage. Any necessary corrective actions will be determined at that time.

6. Events Occurring During the Inspection

6.1 Licensee Event Reports [LERs and Safeguards Event Reports (SERs)]

The following LERs and SERs 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, 10 CFR 73.71, and Station Administrative and Operating, and Security 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.

88-01 Design Error Found In Steam Generator Blowdown Isolation Circuit

88-02 Fire Detection Subsystem Declared Inoperable Due to Damaged Heat Detectors

No unacceptable conditions were identified.

6.2 Steam Generator Blowdown Isolation Valve Modification Errors

On January 14, during licensee evaluation of station drawings as part of the update/verification of the Master Equipment and Parts List, it was identified that an error existed in a design change made [Plant Design Change Record (PDCR) 362] to the Containment Isolation System in 1980 as part of the post-TMI modifications. This change was in response to NRC concerns about equipment automatic actions after an Engineering Safety Features System reset. Specifically, the Steam Generator Blowdown isolation valves (BD-TV-1312-1, 2, 3, and 4) would not trip closed in response to a High Containment Pressure (HCP) signal preceded by a loss of voltage on 4160V buses 1-2 and 1-3 or 480V buses 4, 6, or 7. Additionally, these four valves would not have closed upon receipt of an undervoltage signal on either of these buses. LER 88-01 was submitted and PDCR-930, Steam Generator Blowdown Isolation Circuitry Changes, was created to correct this condition. The PDCR relocated the logic for the bus undervoltages to be in series with the HCP trip logic and modified the relays to correspond to the de-energize to open actuation logic. Post-modification resting was successfully performed under SPL 10.7-335, Preoperational Test of PDCR-930, to verify that the circuitry is now corrected.

The licensee reviewed other containment isolation valve trip circuitry modified by PDCR-362 to verify that the other valves would function as designed. No further deficiencies were found. Also a task force was formed to review post-TMI modifications involving electrical design changes to verify correctness of the design and its implementation. The task force reviewed the design requirements, the functional impact of

the design change on the modified system(s), and the design drawings to verify that the requirements were met. No further deficiencies were identified.

The inspector had no further questions on this matter.

6.3 Containment Penetration Local Leak Rate Testing Failures

Due to the long duration of the current refueling outage, the licensee elected to reperform several surveillance tests to minimize the potential need for a mid-cycle shutdown for overdue surveillances. The repeated surveillances included snubber testing, system functional testing, and containment isolation valve local leak rate (LLRT) testing.

LLRTs were not required to be redone because the plant had not left Mode 5 since they were last performed. However, since containment integrity has been a sensitive issue, these tests were included. Eleven containment penetrations which have historically exhibited high as-found leak rates and repetitive LLRT failures were chosen. Four of these valves are for the reactor coolant pump seal supply. LLRTs on these penetrations would result in unnecessary personnel exposure. For this reason these penetrations were not retested. Of the seven penetrations tested, two failed: P-65, Containment Air Sample; and P-71, Primary Vent Header. In addition, P-80, Auxiliary Spray From Fire System, was observed to be leaking after successful stroke testing; it subsequently failed a LLRT.

P-71, Primary Vent Header, failed the penetration LLRT with leakage of 429 lb-mass/day, exceeding the Inservice Inspection (ISI) acceptance criteria of 5 lb-mass/day at 40 psig. Technical specification (TS) 4.4 requires total containment leakage to be less than 650 lb-mass/day. At the beginning of the refueling outage (July 19[illegible]) this penetration was tested and exhibited a leakage of 0.16 lb-mass/day. During station outages only, this penetration is used to vent air from the primary plant during fill and vent evolutions. The penetration has two in series carbon steel, angle globe stop valves which serve as containment isolation valves, VH-V-522 inside containment and VH-V-525 outside containment. Both of these valves are kept locked closed during station operation. During performance of the LLRT, operations personnel noted a small water discharge when the test connection valve (VH-V-523) was opened. The LLRT was performed, in accordance with SUR 5.7-74, Primary Vent Header (P-71), with unsatisfactory results. To determine the leakage path, the test equipment was relocated to an alternate test connection between the isolation valves (VH-V-524). Leakage was determined to be 170 lb-mass/day. This reduced leakage suggests that test boundary valve VH-V-521 greatly contributed to the initial leakage measurement. Both penetration isolation valves were cleaned and the system was thoroughly flushed with primary water. A second LLRT was performed with unsatisfactory results, 798 lb-mass/day. The majority of this leakage was determined to be through the outboard boundary valve (VH-V-521). This valve was also disassembled and cleaned. A satisfactory as-left LLRT was then performed

with leakage measured at 0.05 lb-mass/day. The licensee concluded that normal operation of the vent system during the outage resulted in the passage of Reactor Coolant System fluid through these valves. Any sediment present could easily have become lodged in the valve seat/disc area because these valves are installed upside down (VH-V-525) and sideways (VH-V-522) in the lines. To ensure minimal penetration leakage when containment integrity is required, the licensee has established a permanent start-up LLRT to be performed after completion of fill and vent evolutions and before containment integrity is set. The valves would then be locked closed for the duration of the operating cycle.

The Containment Air Sample penetration (P-65) also failed its LLRT retest with leakage of 19 lb-mass/day (its ISI acceptance criterion is 10 lb-mass/day). The as-left LLRT performed in July 1987 was 0.17 lb-mass/day. Two check valves, VS-CV-1103 and 1104, one on each side of the containment wall, serve as containment isolation valves for this penetration. The as-found test performed in July 1987 identified 41 lb-mass/day leakage from VS-CV-1104 and no leakage from VS-CV-1103. VS-CV-1104 was disassembled and the presence of loose dirt and dust was identified. The valve was not reworked; it was reassembled and the LLRT retest was satisfactory. This most recent LLRT identified a leakage of 19 lb-mass/day for VS-CV-1103 and no leakage for VS-CV-1104. When VS-CV-1103 was disassembled and inspected, no seat or disc degradation or dirt accumulation was observed. The valve was reassembled and retested satisfactorily with a leakage of 0.61 lb-mass/day. The licensee concluded that dirt or dust may have been lodged between the valve seat and disc but fell away when the valve was disassembled. Currently, the licensee is evaluating several possible long-term corrective actions to prevent future LLRT failures for this penetration. These potential solutions include installation of heat tracing to eliminate buildup of condensation and altering the check valve design.

P-80, Auxiliary Spray From Fire System, failed its LLRT under different circumstances. This penetration provides river water from the Fire Protection System to the containment spray header and would only be used under extreme emergency conditions. After successfully stroke testing MOV-RH-31 in a routine operability verification, operations personnel noted unexpected water leakage from the penetration drain valve, RH-V-36. Under Engineering recommendation, a LLRT was performed. The penetration exhibited a leak rate of 1237 lb-mass/day. This alone exceeds the TS 4.4 limit for total containment leakage of 650 lb-mass/day. The valve was disassembled and inspected. An accumulation of silt and sand was identified between the valve seat and disc. The valve was cleaned, reassembled and successfully leak tested (1.5 lb-mass/day).

This valve has been leak tested three times during this refueling outage. The In-Service Testing (IST) Program requires that it be stroke tested during each cold shutdown and, when outages are extended as this one was, quarterly. The penetration was successfully leak tested in July 1987. The first quarterly stroke test was successfully performed in November

1987; leakage was observed from the drain line. When the second quarterly stroke test was performed in February 1988, similar leakage was again observed. In both cases, after flushing and cleaning, the penetration passed the LLRTs. This valve is not operated for purposes other than operability verifications because this would result in a spray down of containment. Containment integrity can be assured by LLRT after valve manipulation. The licensee is currently evaluating alternative testing sequences to preclude LLRT failures after operability stroke testing.

Licensee maintenance and testing activities in response to these LLRT failures was thorough. Long-term corrective actions for P-80 and P-65 will be reviewed during future inspections.

6.4 Inadequacies Identified In Containment Penetration Modifications

During this refueling outage, the licensee made changes to several containment penetrations under Plant Design Change Record (PDCR) 878, Appendix A and J Penetration Modifications. Included in this modification was the installation of nine solenoid-operated valves (SOVs). The original PDCR specified SOVs with Elastomer seating surfaces. After the purchase order for these valves was processed, the licensee determined it prudent to substitute SOVs with Viton seats. (The Elastomer seats tended to heat up when energized for long periods and then had the potential to stick in the closed position.) The vendor supplied kits for the seat replacements after the purchased valves were installed. During this valve seat changing process the licensee identified deficiencies with the initial valve installation.

The PDCR had been performed by a contractor (C.N. Flagg) earlier in the outage. When site maintenance personnel were changing the valve seats, they identified an error in the size of terminal lugs which were installed. Furthermore, wire braiding was not removed before installation of the Raychem splices. This was brought to management attention through the Plant Information Report (PIR) process. All nine valves were removed for seat replacement and the deficiencies were corrected.

Generation Construction has informed the contractor of these deficiencies. Preliminary information indicates that contractor supervisors were aware of the correct methods for valve installation but that work crews did not adequately follow procedures. Contractor Quality Control (QC) personnel were also used for this PDCR. The licensee has indicated that the contractor is performing a review of their QC program and coverage and personnel procedure adherence. An inspection of work done by this contractor has been performed and no further deficiencies were identified. The inspector had no additional concerns at this time; this matter will be reviewed in future inspections.

7. Periodic and Special Reports

Upon receipt, periodic and special reports submitted pursuant to Technical Specification 6.9 were reviewed. This review assessed whether: the reported information was valid and included the NRC required data; test results and supporting information were consistent with design predictions and performance specifications; and 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 87-12, Covering the Period December 1, 1987 through December 31, 1987.

-- Monthly Operating Report 88-01, Covering the Period January 1 through January 31 1988.

-- New Switchgear Building Construction Bimonthly Progress Report No. 8, dated February 1, 1988.

8. Qualification of Containment High Radiation Monitors

In February 1987, the NRC and several licensees (including Northeast Utilities) received a 10 CFR Part 21 report from General Atomic Technologies, Inc. informing them that a defect was identified in the Sorrento Electronics Containment High Range Radiation Detectors. The defect is in the inside containment portion of the ion chamber's signal coaxial cable which was manufactured by Rockbestos Co. Under post-LOCA conditions, with elevated containment temperatures, the cable insulation resistances vary. The variances are a function of the cable configuration and the containment post-LOCA temperature profile.

Haddam Neck has two of these detectors (CD-1 and 2) located on the charging floor in containment. They are used for radiation monitoring of containment under accident conditions. Technical Specification (TS) 3.23, Post Accident Instrumentation, requires that two containment high radiation monitors be operable during Modes 1-4, with the capability to measure up to 1E8 R/hr and an alarm setpoint less than or equal to 100 R/hr. Indications from these detectors are used in the Emergency Operating Procedures (EOPs) and Emergency Implementing Procedures (EPIPs). The EOPs contain a decision making step based on the indications of CD-1 and 2 in ES 1.4, Transfer To Two Path Recirculation. Whether the detectors are indicating above or below 20,000 R/hr determines if the recirculation path includes systems outside containment. At radiation levels above 20,000 R/hr, the path will not include equipment outside containment. CD-1 and 2 indications are used in EPIP 1.5-7, Radiological Dose Assessment, for offsite dose calculations. The action level for these calculations is when the detectors are reading at least 34,000 R/hr.

Between February and June 1987, the licensee had extensive communications with Sorrento Electronics to determine the effect of this defect on Haddam Neck's detectors. An analysis was conducted specific to the station's cable configuration and post-LOCA temperature profile. These calculations were finalized during this inspection period and are documented for the Millstone Units and Haddam Neck in Reportability Evaluation Form 87-21. The licensee concluded that CD-1 and 2 exhibit an error of 9.4 R/hr in the nonconservative direction. The licensee has evaluated the implications of this inaccuracy and determined that, since the detectors' ranges are so broad and negligible in the action range, they are operable.

It was also noted that the error in the low range does not meet the guidance in Regulatory Guide RG-1.97, Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident. This is because they do not meet the factor of two accuracy specified for the lower range. The licensee plans to submit an information letter to NRC describing this exception to RG-1.97. This matter will be reviewed during future inspections.

9. Engineering Expertise On Shift

On January 21, the Office of Nuclear Reactor Regulation (NRR) informed the inspectors of concerns involving the use of dual-role Shift Supervisor/Shift Technical Advisors (SS/STA) at the Millstone plants and Haddam Neck. The current Haddam Neck Technical Specifications (TS) Table 6.2-1, Minimum Shift Crew Composition, specifies a crew of seven people during modes 1, 2, and 3. Because the SS serves in a dual capacity (SS and STA), these seven positions have been filled by six individuals. This issue is discussed in detail in the Millstone Unit 1 NRC Inspection Report 50-245/88-02. No NRC field inspection action is planned until the latest licensee's submittal on this matter is evaluated by the NRC.

10. Security Badge Retrieval

The licensee has installed equipment to prevent unauthorized removal of security keycards. Each keycard is tagged with a microwave resonance device similar to those used to prevent theft from retail establishments. The detection equipment is located at an exit point. This equipment is not a required plant physical protection system but should improve the overall implementation of the program by reducing the likelihood of unauthorized removal of keycards and also assist with the monitoring by security officers.

1 . Exit Interview

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