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Report No. 50-213/89-20

Docket No. 50-213

License No. DPR-61

Licensee: Connecticut Yankee Atomic Power Company
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Facility Name: Haddam Neck Plant

Inspection at: Haddam Neck, Connecticut

Inspection Conducted: October 18, 1989 - November 21, 1989

Reporting
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Reactor Projects Section 4A
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Inspection Summary: Inspection on October 18-November 21, 1989
(Inspection Report No. 50-213/89-16)

Areas Inspected: Routine safety inspection by the resident inspectors. Areas reviewed included the reconstitution of fuel assemblies, preoperational testing of modified systems, local leak rate testing, training for the planned fitness for duty program, and the review of written reports made to the NRC.

Results: This was the second routine resident inspection during the 1989 Refueling Outage. The licensee has decided to remove the reactor vessel thermal shield from the core support barrel after concluding that the thermal shield supports had reached the end of their serviceable lifetime. In regard to this, the licensee is encouraged to communicate their position to the Office of Nuclear Reactor Regulation on the type of licensing action they will follow for this modification (i.e. 10 CFR Part 50.59 or 50.90) (section 4.1.2). Methods for maintaining nuclear fuel accountability on a rod by rod basis were verified in place during the fuel reconstitution process (section 4.1.1). The subtle design deficiency found during preoperational testing of the newly modified reactor protection system indicates that the licensee has implemented a through test program (section 4.2.1). No new Unresolved Items were identified.

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* The NRC Inspection Manual inspection procedure or temporary instruction that was used as inspection guidance is listed for each applicable report section.

DETAILS

1. Summary of Facility Activities

During this inspection period, the Fifteenth Refueling Outage was in progress. Major work activities included fuel reconstitution, and inspection of the reactor core support barrel thermal shield support attachments. Because of this, all reactor fuel remained within the the fuel storage pool during the entire inspection period. The reactor outage, which began on September 3, has been extended into April, 1990 to correct the damage to the fuel and thermal shield supports.

2. Plant Operations

2.1 Operational Safety Verification

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

- -- control room
- -- primary auxiliary building
- -- vital switchgear room
- -- radiological control point
- -- Appendix R switchgear building
- -- auxiliary feedwater pump room
- -- primary access point
- -- protected area fence
- -- yard areas
- -- intake structure
- -- diesel generator rooms
- -- turbine building

Control room instruments and plant computer indications were observed for correlation between channels and for conformance with technical specification (TS) requirements. Operability of engineered safety features, other safety related systems and onsite and offsite power sources were verified. The inspector observed various alarm conditions and confirmed that operator response was in accordance with plant operating procedures. Routine operations surveillance testing was also observed. Compliance with TS and implementation of appropriate action statements for equipment out of service was inspected. Plant radiation monitoring system indications and plant stack traces were reviewed for unexpected changes. Logs and records were reviewed to determine if entries were accurate and identified equipment status or deficiencies. These records included operating logs, turnover sheets, system safety tags, and the jumper and lifted lead book. Plant housekeeping controls were monitored, including control and storage of flammable material and other potential safety hazards. The inspector also examined the condition of various fire protection, meteorological, and seismic monitoring systems. Control room and shift manning were compared to regulatory requirements and portions of shift turnovers were observed. Control room access was properly controlled and a professional atmosphere maintained.

In addition to 171 hours of inspection during normal utility working hours, the review of plant operations was routinely conducted during portions of backshifts (evening shifts) and deep backshifts (weekend and midnight shifts). Inspection coverage was provided for 46 hours during backshifts and 4 hours during deep backshifts. Operators were alert and displayed no signs of inattention to duty or fatigue.

3. Radiological Controls

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

4. Maintenance and Surveillance

4.1 Maintenance Observation

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

-- core support barrel thermal shield inspection,

-- reactor vessel clad inspection,

-- "A" auxiliary feedwater pump disassembly and troubleshooting, and

-- fuel inspection, cleaning and reconstitution.

4.1.1 Reactor Vessel Thermal Shield Support Degradation

An inservice inspection was made of the reactor vessel thermal shield attachments to the core support barrel earlier in the refueling outage (Reference Inspection Report 50-213/89-16, paragraph 4.1.1). This inspection revealed significant deficiencies with various components of the system used to stabilize the thermal shield. Based on engineering analysis the licensee has concluded that various components of the support system had reached the end of their serviceable lifetime.

The details of this and of a related fuel clad problem were presented to the NRC during an October 25 meeting (Reference Meeting Report Docket 50-213, dated November 17, 1989). The licensee has concluded that repair of this component is not practical and is beginning to plan a strategy for the removal of the thermal shield and its support components from the core support barrel.

The inspector was informed that after preliminary evaluation of this modification the licensee believed that it was within the scope allowed by 10 CFR 50.59. The inspector requested that the licensee inform the NRC Office of Nuclear Reactor Regulation as soon as possible after reaching a conclusion. This was because certain aspects of the thermal shield and its influence on the reactor vessel metallurgy and reactor vessel flow characteristics implied a need for an evaluation by the NRC in accordance with 10 CFR 50.90.

The core support barrel assembly was placed back into position in the reactor vessel with no additional work performed during this inspection period.

4.1.2 Inspection and Reconstitution of Reactor Fuel

Inspections conducted earlier in the refueling outage found that damage to fuel rod clad had resulted from metal chips which accumulated in the region between the assembly lower nozzle and the first spacer grid. Ultrasonic examination revealed clad failures of approximately 343 fuel rods within 109 fuel assemblies intended for reuse. An unknown additional number of fuel rods had sustained damage without clad perforation (Reference Inspection Report 50-213/89-16, paragraph 4.1.2).

The licensee has visually inspected all reuse assemblies via CCTV and mapped the debris sites within each. The results from ultrasonic examinations and visual inspections have led to the second phase of the licensee's fuel recovery effort.

With the aid of the fuel manufacturer, each assembly to be reused is being "reconstituted". During this process fuel rods with clad failures are removed from an assembly and replaced with an acceptable rod of similar power history.

Since fuel rods located either adjacent to failed rods or located at debris sites may have damaged clad, they are examined visually and by eddy current testing (ECT). Rods are rejected if defects are found greater than 20 percent of wall thickness. If significant defects are found, the area of inspection is increased to include additional rod locations. This progresses until acceptable clad conditions are found.

At present, all fuel found damaged beyond use is of stainless steel clad material. It has a significantly thinner wall thickness than the four zircaloy clad lead test assemblies (0.0165 inch vs 0.027 inch) which were part of the core during the last operating cycle. There have been no unacceptable defects found in zircaloy clad fuel to date.

The scope of this inspection work includes all reload assemblies and is quite extensive. Initial estimates of damaged and defective fuel is between 700 and 1200 rods. This will require approximately 2500 rods to be inspected.

The debris, which caused the damage, is believed to be stainless steel machine chips or flakes which escaped controls to capture them during modifications to the thermal shield supports performed during the last refueling. The issue of effective cleanup of this material remains under review by the licensee.

The inspector reviewed the process for selection of donor rods being used in the reconstitution process. Rods are scavenged from designated once or twice used fuel assemblies. Each rod is inspected to be free of defects and has an accumulated exposure similar to the damaged rod being replaced. The inspector verified that a system is in place to track the location of each rod individually along with its accumulated power history. The revised data for exposure at rod locations are provided to the organizations performing the nuclear, thermal - hydraulic and safety analysis. This data will be used in revised calculations and for nuclear material accountability. There were no unacceptable conditions identified.

This work is expected to continue through April of 1990.

4.2 Surveillance Observation

The inspector witnessed selected surveillance tests to determine whether properly approved procedures were in use; technical specification frequency and action statement requirements were satisfied; necessary equipment tagging was performed; test instrumentation was in calibration and properly used; testing was performed by qualified personnel; test results satisfied acceptance criteria and unacceptable results were properly dispositioned. Portions of the following activity was reviewed:

-- SUR 5.1-17B, Emergency Diesel Generator EG-2B Manual Starting and Loading Test

4.2.1 <u>Reactor Protection System Preoperational Testing</u>

During a preoperational test, the licensee determined that in a certain configuration, instrumentation used within a new reactor protection system (RPS) failed to achieve a desired design protective feature trip with a single component failure. A extensive modification has been installed to replace the original plant equipment which performed the RPS function with two trip systems each of two subchannels in a modern configuration using a microprocessor based Foxboro Spec 200 system.

Specifically, loss of power to portions of the trip logic did not always result in all output devices failing to the safe trip condition. This was found to occur only when two or more "dry contact" output relays (Foxboro N-2AO-L2C-R, logic to contact converter cards) were driven by a single logic module; and, only if the logic to contact converter cards (L2C-R) received power from separate power distribution modules (Foxboro N-2AX-DP11).

The logic to contact converter cards are simple electromechanical relays; the logic module is of microprocessor technology.

The Spec 200 equipment consists of rack mounted modules. Each sub-channel is provided with its own power supply; distribution to each rack level or "nest" is through fused outputs of a power distribution module (DP-11). The module provides undervoltage protection by deenergizing its output.

Circuit analysis revealed that interaction had occurred between two logic to contact converter cards (L2C-R) when each was powered through separate power distribution modules (DP-11). Current from the energized power distribution module passes through the relay coils of the L2C-R cards into a "sneak" circuit provided by an LED in the deenergized DP-11. This path provides sufficient current to hold the relays in an energized position although the logic module is deenergized.

The problem appears to be easily corrected and any design changes will be supported through the vender.

Although the effect of this single failure design deficiency is minimized because of channel redundancy within the Haddam Neck Plant's new reactor protective system, the licensee and the equipment vendor are evaluating 10 CFR Part 21 reportability because of the generic implications of this condition.

4.2.2 Local Leak Rate Testing

During this inspection period, the licensee continued performance of local leak rate tests (LLRT's) in accordance with 10 CFR 50, Appendix J. The inspector reviewed the test results for the following LLRT's:

-- SUR 5.7-100, Electrical Penetrations P-B
-- SUR 5.7-149, Inservice Testing of Containment Heating Steam Isolation Valves, HS-TV-380 and 381

The tests satisfy the surveillance requirements of technical specifications 4.4 and 4.10. The test results were satisfactory and no items of concern were identified.

The inspector observed portions of preparation and performance of procedure ST11.7-29, RCP seal water supply check valve shop LLRT. This ST covered performance of the LLRT for new reactor coolant pump (RCP) seal water supply check valves being installed by plant design change record (PDCR) 962, RCP seal water injection replacement check valves.

During review of the test configuration, the inspector identified that the check valve identification number inscribed on the valve by the manufacturer did not agree with the piping and instrument diagram valve identification number; the valves were marked CH-CV-305A, B, C and D rather than CH-CV-405A, B, C and D. This difference apparently had not been noted by those handling the new valves or making the modification. When informed of the issue, the licensee interrupted testing until this discrepancy was resolved.

Quality services division (QSD) personnel conducted a field verification of all four piping assemblies to verify materials traceability and that the assemblies were constructed in accordance with the design. No deficiencies were identified. The documentation of this verification has been added to the associated work order packages. The valves are identified as CH-CV-405A, B, C and D as described by PDCR 962.

The licensee intends to inspect plant components for obsolete or conflicting identification tags. A program is active to install highly visible identification tags. This review will confirm that other forms of marking do not conflict with these new tags.

The inspectors agree that this action appears to be appropriate and will enhance the present program. No further concerns were identified.

5. Security

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

5.1 Fitness For Duty Program Training

During this inspection period the licensee conducted employee training in the fitness for duty (FFD) program. The training was given to all employees with unescorted access. Course objectives included an overview of the licensee's fitness for duty policy and the employee assistance program as required by 10 CFR 26.21. This course also addressed the information required for persons assigned to station security escort duties addressed within 10 CFR 26.22.

The inspector attended the training classes and verified that the information required for escort training was presented. The licensee is providing additional training to supervisors; this was not reviewed by the inspectors.

The licensee will begin the FFD Program on December 3, 1989, one month prior to the date specified by 10 CFR Part 26.

6. Engineering and Technical Support

The inspector reviewed selected design changes and modifications made to the facility which the licensee determined were not unreviewed safety questions and did not require prior NRC approval as described by 10 CFR 50.59. Particular attention was given to safety evaluations, Plant Operations Review Committee approval, procedural controls, post-modification testing, procedure changes resulting from this modification, operator training, and UFSAR and drawing revisions. The following activities were reviewed:

6.1 Reactor Coolant Pump Seal Water Supply Seismic Qualification

The licensee discovered the lack of documentation for seismic design evaluation for the reactor coolant pump seal water injection piping, a reactor coolant system (RCS) auxiliary subsystem. This finding was made during the engineering work for a plant modification which will upgrade the containment isolation valves of this system. The modification was designed with current seismic methodology.

During this process a review was to be made of the original plant design and inspection records. A search of design records failed to disclose previous seismic consideration to this piping.

This two-inch line is potentially significant to reactor safety in that a postulated pipe break with the original configuration could have compromised both RCS and containment integrity. The referenced modification corrected that potential hazard to the seal water injection piping for each of the four reactor coolant pumps.

Because the seismic classification for piping of this type is defined in the Updated Final Safety Analysis Report, Table 3.2-1, and that original plant design and inspection records failed to disclose previous seismic consideration of this piping, a notification required by 10 CFR 50.72(b)(2)(iii)(C) was made on November 20. The issue of the lack of seismic documentation for this system remains under review by the licensee.

Pending the completion of this review, there were no unacceptable conditions identified.

7. Safety Assessment and Quality Verification

7.1 Plant Operations Review Committee

The inspector attended several Plant Operations Review Committee (PORC) meetings. Technical specification 6.5 requirements for required member attendance were verified. The meeting agendas included procedural changes, proposed changes to the Technical Specifications, Plant Design Change Records, and minutes from previous meetings. The PORC meetings were characterized by frank discussions and questioning of the proposed changes. In particular, consideration was given to assure clarity and consistency among procedures. Items for which adequate review time was not available were postponed to allow committee members time for further review and comment. Dissenting opinions were encouraged and resolved to the satisfaction of the committee prior to approval. The inspectors observed that PORC adequately monitors and evaluates plant performance and conducts a thorough self-assessment of plant activities and programs.

7.2 Review of Written Reports

Periodic and special reports, licensee event reports (LERs), and safeguards event reports (SERs) were reviewed for clarity, validity, accuracy of the root cause and safety significance description, and adequacy of corrective action. The inspector determined whether further information was required. The inspector also verified that the reporting requirements of 10 CFR 50.73, 10 CFR 73.71, station administrative and operating procedures, and Technical Specification 6.9 had been met. The following reports were reviewed:

LER 89-16	Potential Rating Deficiency Identified in Molded Case Circuit Breakers
LER 89-17	Steam Generator Eddy Current Testing Results Classified as Category C-3
LER 89-18	Containment Penetration Fails Type C Local Leak Rate Test
LER 89-19	Inoperable Fire Barrier Identified in Screenwell Building
SER 89-S06	Safeguards Event Report

Haddam Neck Plant Monthly Operating Report 89-10, covering the period October 1, 1989 to October 31, 1989

No unacceptable conditions were identified.

8. Exit Interview

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