U.S. NUCLEAR REGULATORY COMMISSION DCS No. 50213-850516
REGION I

Report No. 50-213/85-11

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: May 10 - June 14, 1985

Inspector: Ele McCabe, for — 7/2/85
Paul D. Swetland, Senior Resident Inspector — Date Signed

Approved by: Ele McCabe — 7/2/85
E. C. McCabe, Chief, Reactor Projects Section 3B, Division of Reactor Projects — Date Signed

Summary: Routine resident inspection (66 hours) of plant operations, radiation protection, physical security, fire protection, Emergency Plan exercise, previous inspection findings, events occurring during the inspection, Systematic Evaluation Program topics, and a special inspection of potential inter-systems loss of coolant accident configurations.

Seven open items were closed. No violations or unacceptable conditions were identified.

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.

On June 7, 1985, the licensee conducted a limited site emergency drill to exercise the station evacuation and personnel accountability procedures. The drill, simulating loss of cooling and shielding of used fuel elements in the spent fuel pool, was initiated at 9:00 a.m. Site evacuation was ordered at 9:02. The emergency response force completed protected area accountability at 9:35 and site accountability was finished at 9:42 a.m. Upon completion of these drill objectives, the exercise was terminated. The inspector verified that observations related to the improper assignment of technical support center accountability in procedure 1.5-26, Manager of Control Room Operations, were corrected by the licensee. No further discrepancies were identified.

2. Followup on Previous Inspection Findings

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

2.1 (Closed) Followup Item (213/82-22-05) The licensee was to formalize restoration procedures for emergency operations facility (EOF) distribution loads which do not re-energize following a loss of power event.

The licensee determined that only EOF ventilation fans and dampers must be realigned after the EOF emergency diesel generator restores power. The inspector reviewed the implementation of restoration responsibility and instructions in revision 3 to procedure 1.5-19, EOF Actuation. No discrepancies were identified.

2.2 (Closed) Unresolved Item (213/84-07-05) The NRC was to review repair work on the containment high range radiation monitors and licensee corrective actions for the identified design change control program deficiencies. The radiation monitor cable-to-detector connections were reworked during the 1984 refueling outage using environmentally qualified connectors and specified heat shrink insulation. The inspector identified no further discrepancies between this installation and the approved design change. Subsequent to the identification of design change control deficiencies in this modification, further design control problems were identified in other modifications, and NRC enforcement action was taken as detailed in NRC Region I Inspection Report 50-213/84-23. Licensee corrective actions include a comprehensive revision of the design change process and an ongoing independent review of that process and the implementation of previous modifications. Implementation of those actions is being separately followed by NRC. This item is closed.

2.3 (Closed) Unresolved Item (213/84-08-01) Subsequent to the identification of design deficiencies related to post-accident sample system modifications, the licensee committed to review the design change control/modification program to identify areas for improvement to prevent recurrence of similar events. Other problems identified in the design change control areas reinforced the need for a comprehensive upgrade of this area. The licensee approved revised quality assurance procedures related to design control in November 1984. Implementation of these new procedures was completed in February 1985. An independent review of the effectiveness of these measures is under way, and NRC is monitoring the progress of licensee activities as a followup to escalated enforcement action in this area. This item is closed.

2.4 (Closed) Followup Item (213/84-14-02) The licensee was to provide formal guidance to plant personnel regarding control of fire doors. In May 1985, the licensee implemented administrative control procedure 1.0-29, Control of Fire Doors. This procedure defines the proper operation of fire door hardware and prescribes appropriate compensatory actions for inoperable fire doors. The inspector verified that the control measures specified were consistent with Plant Technical Specification requirements. The inspector had no further questions in this area.

2.5 (Closed) Violation (213/84-22-01) The licensee failed to incorporate safety-related voltage regulator settings in several procedures used to shut down the emergency diesel generators (EDGs). Mis-setting the voltage regulator could prevent the EDG from assuming safeguards loads on demand. The licensee revised procedures 3.1-9, Total Loss of AC Power; 3.1-10, Partial Loss of AC Power; and 5.1-18 & 19, Loss of AC Testing

with Core Cooling Actuation to include appropriate voltage regulator settings prior to EDG shut down. Licensee and NRC review of other voltage sensitive components identified no similar discrepancies. To prevent recurrence of similar problems, the licensee has initiated a safety-related setpoint control program to define and control safety significant settings which are incorporated in the plant design. This program was implemented in October 1984 by administrative control Procedure 1.2-3.3, Setpoint Change Request. The inspector had no further questions in this area.

2.6 (Closed) Followup Item (213/80-BU-05) In order to assure that adequate vent capacity was available to prevent collapsing the refueling water storage tank (RWST) during maximum pump-down conditions, a 14 inch temporary vent was installed on the RWST manway. The licensee was to complete permanent modifications to the RWST vent and restore the manway to its original configuration. The licensee completed the installation of plant Design Change Request 532, implementing these modifications, in April 1985. The inspector reviewed the completed modification package and verified the satisfactory implementation of this change. This action completes NRC review of licensee commitments to IE Bulletin 80-05.

3. Followup on Systematic Evaluation Program Findings

3.1 Topic VI-1, Organic Materials and Post-Accident Chemistry (Action Item 4.21)

To control the pH of containment sump water used for post-accident recirculation cooling, the licensee committed to install baskets of trisodium phosphate (TSP) in the containment lower level. The licensee installed these baskets during the 1984 refueling outage. The inspector reviewed the implementation of PDCR 618, TSP Baskets. Two 25 cubic foot TSP baskets were installed close to the containment sump suction. Periodic surveillance of the chemical effectiveness was implemented on June 7, 1985, by procedure 5.4-36, TSP Surveillance Requirements. The inspector verified that operability and surveillance requirements for the TSP baskets were included in draft Standard Technical Specifications for the Haddam Neck Plant, which are to be submitted for approval in 1985. The inspector had no further questions in this area.

4. Review of Interfacing Systems Loss of Coolant Accidents (LOCAs)

Recently, events have occurred at several reactor facilities in which low pressure piping connected to the reactor coolant system has been overpressurized. The potential for loss of coolant accidents in these "interfacing" low pressure systems had been previously reviewed by NRC. Generic licensing action was implemented to reduce the risk of this event for pressurized water reactors during the period 1980-1981 because of a dominance of this accident sequence for PWRs in the Reactor Safety Study (WASH 1400). These recent overpressurization events have occurred at boiling water reactors and have resulted, in part, because of multiple component/personnel failures. During

this inspection, the arrangement, maintenance, and testing of such interfacing systems were reviewed to verify the data used in previous evaluations of the interfacing LOCA concern and to assess the potential for bypassing of system components or interlocks designed to protect against this accident sequence. The inspector also interviewed selected operations and maintenance personnel to evaluate their understanding of the interfacing LOCA event and the effect of certain maintenance or test activities on the potential occurrence of this event.

NRC has compiled summary descriptions of interfacing systems at many reactor facilities. This computer data base is used for NRC evaluations of certain accident sequences and was published in NRC NUREG/CR 2069, Summary Report on a Survey of Light Water Reactor Safety Systems. As part of this inspection, the accuracy of this data base was verified by review of plant drawings and walk down of accessible systems. Inspection findings were as follows:

-- The descriptions of low pressure systems connected to the RCS in NUREG/CR-2069 were correct. Two interfacing systems not listed in NUREG/CR-2069 (the reactor coolant drain and letdown systems) would be difficult to overpressurize because of their large capacity relief protection which returns directly to the volume control tank with no downstream isolation valves.

-- NUREG/CR-2069 lists the design pressure of the low pressure safety injection system (LPSI) and the residual heat removal (RHR) system as 875 psig. This number is based on the piping design specification for the installed Class 601 piping at 200 degrees F. The licensee classes LPSI at 600 psi based on Class 601 pipe at 300-500 degrees F. For the RHR system, the design pressure is limited by the RHR heat exchanger design of 500 psig.

-- As a result of the 1980-81 NRC and licensee activities related to interfacing LOCAs, several corrective measures were implemented to prevent potential LOCA events. High/low pressure boundary valves are leak tested while the plant is shutdown for refueling and subsequent inservice testing of high/low pressure boundary valves is exempted. The boundary valves remain closed during plant operation, thereby minimizing the potential for a LOCA. The safety injection boundary check valve leak tests have been incorporated into Technical Specifications 3.14 and 4.3. Implementation of these tests was reviewed by NRC in inspection 50-213/84-28.

-- The inspector determined that routine operations and surveillance did not create increased potential for interfacing LOCAs because of the multiple boundary valves which are routinely cycled only during cold shutdown conditions. The refueling interval leak testing of these boundary valves increases confidence in the integrity of the high/low pressure boundary. This maintenance of double isolation for low pressure systems reduced the checking of operability of the safety injection motor-operated valves (MOVs). These valves, which open to provide emergency core

cooling, are no longer routinely tested monthly during plant operation. Subsequently, refueling outage checks have been performed with no inoperability identified.

One surveillance which may increase the risk of an interfacing LOCA is the six-week interval protection system channel check of the safety injection actuation (SIA) logic. During these tests a partial trip signal develops in the SIA logic. Another, spurious trip input could actuate safety injection. If such were to occur, the safety injection MOVs would open leaving the check valve as a single boundary. As stated above, refueling interval check valve leak testing has established confidence in this boundary. The licensee's surveillance procedures provide adequate precaution against inadvertent SIAs and operators and technicians were familiar with the high risk nature of these tests. The licensee has also committed to install pressure sensing interlocks in the safety injection MOV opening circuits to prevent the isolation valves from opening until RCS pressure has dropped below the design pressure of the safety injection system.

-- Maintenance of interfacing systems is conducted in accordance with the licensee's routine quality assurance program, including appropriate documentation, administrative and quality control, and post-maintenance testing. No special requirements are established for interfacing systems, however, the nature of these systems' required operating conditions and the frequency of valve operation and testing necessitates that major maintenance be conducted only during cold shutdown. Boundary leak testing is required prior to the following plant startup. As such, the potential for an interfacing LOCA caused by maintenance is low. The inspector identified that safety injection MOV stroking could occur after tightening of the valve packing when necessary. Operators were confident of the safety of this evolution based on the refueling interval testing of the boundary check valves and the need to reestablish operability of the MOV after this maintenance.

The inspector also reviewed the licensee's evaluation of IE Information Notice 84-74, Isolation of the Reactor Coolant System from Low-Pressure Systems Outside Containment, which reevaluated the interfacing LOCA event and found no further potential interfacing system pathways and no surveillance or preventive maintenance activities which compromise protective measures for this event. The inspector had no further questions in this area.

5. Followup on Events Occurring During the Inspection

5.1 Licensee Event Reports (LERs)

The following LER was reviewed for clarity, accuracy of the description of the cause, and adequacy of the 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 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.

-- 85-11 Multiple Dropped Control Rods

On May 16, 1985 two control rods (Nos. 30 and 33) dropped into the reactor core when their rod control group was withdrawn during a routine plant makeup (dilution) operation. Normal dropped rod indications including two rod bottom lights and rod position deviation and dropped rod annunciators were observed. The required automatic turbine runback was in progress. Operators manually tripped the plant by procedure, upon verification of multiple dropped rods. All plant systems functioned properly in response to the trip, and the plant was stablized in the hot standby mode. Subsequent troubleshooting of the rod drive mechanisms for rods 30 and 33 identified no abnormal operation. The licensee conducted limited rod exercise testing to verify satisfactory operation of the two rods. No problems were found. The reactor was restarted on May 16 with all control rods functioning properly. The plant returned to full power operation on May 18, after completing secondary chemistry cleanup holds at 5 and 25 percent power.

During licensee review of rod control maintenance history, it was determined that previous rod failures of these and other rods had occurred in 1969 and 1980. The licensee has conferred with utility and vendor representitives to determine whether any common failure mechanism could be identified. No immediate corrective action was identified, however, a plan of further rod testing has been developed and will be implemented during the next available plant maintenance outage. The inspector will follow the results of this test program during a subsequent inspection (IFI 213/85-11-01).

7. <u>Exit Interview</u>

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