U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 90-14

Docket No. 50-213/90-14

License No. DPR-61

Licensee: Connecticut Yankee Atomic Power Company
107 Selden Street
Berlin, Connecticut 06037-0218

Facility Name: Haddam Neck

Inspection At: Berlin, Connecticut

Inspection Conducted: July 31-August 3, 1990

Inspectors: A. Lohmeier
A. Lohmeier, Reactor Engineer, MPS, EB, DRS — 1-4-91 date

Approved by: G. H. Gray
Edwin H. Gray, Senior Reactor Engineer,
Materials and Processes Section, EB, DRS — 1/4/91 date

Inspection Summary: Inspection on July 31, 1990-August 3, 1990 (Inspection Report No. 50-213/90-14)

Areas Inspected: The inspector performed a routine inspection of documentation of cyclic transients during the lifetime history of plant operation and methods used by the licensee of evaluating the fatigue life usage of primary components resulting from the cyclic transient operation.

Results: The inspector found that considerable attention had been directed in the past toward fatigue life usage and evaluation of critical primary system components. Data is available for assessment of the effects of cyclic operation. A procedure for reporting deviation of primary system heatup and cooldown rates from Technical Specification is available. The inspector found that no procedures exist for continuing assessment of the remaining fatigue life of critical primary components.

No violations or deviations were identified as a result of this inspection.

DETAILS

1.0 Persons Contacted

*G. H. Bouchard, Unit Director, Haddam Neck Site
J. Ely, Supervisor, Component Engineering, NE Utilities Service Company (NUSCo)
C. Gladding, Engineer, Haddam Neck Site

United States Nuclear Regulatory Commission (USNRC)

*A. A. Asars, Resident Inspector
T. Shedlosky, Senior Resident Inspector

*Denotes those attending the exit meeting on August 3, 1990

The inspector also contacted other technical personnel during the inspection.

2.0 Scope of Inspection

A routine, announced inspection was conducted at the offices of Northeast Utility Service Company (NUSCo) in Berlin, Connecticut during the period July 31 through August 3, 1990. The scope of the inspection included review of documentation of plant operating transients, procedures for tracking operational transients and evaluation by the licensee of fatigue life usage of primary system components under the combined pressure and thermal cycles to which the components have been subjected at the Haddam Neck Station of the Connecticut Yankee (CY) Atomic Power Generation Company.

3.0 Background

Since the advent of commercial nuclear power generation systems, changes in the design criteria of pressure vessel components have taken place commensurate with advancement in understanding of materials behavior and computational technology. At first, nuclear vessels were designed to the rules of the ASME Boiler and Pressure Vessel Code, Section VIII, for Unfired Vessels. Recognition for the need for more stringent requirements brought about the addition of Code Cases 1270N and 1273N, augmenting the rules of Section VIII. The Code Cases provided for different levels of allowable stress, depending on the nature of the stress, e.g., primary or secondary. Finally, a stress criteria considering the effect of fatigue was provided in a new Section III of the ASME Boiler and Pressure Vessel Code. This standard provided for both classification of stress limits according to the nature of stress, and consideration of fatigue as a cyclic life failure mechanism.

Although the later nuclear power generation system components were designed to Section III standards, the reactor vessel and steam generators at Haddam Neck were designed utilizing the rules of Section VIII and Code Cases 1270N and 1273N. For the reactor vessel, the Section VIII rules were augmented by utilizing the "Tentative Structural Design Basis for Reactor Vessels and Directly Associated Components" and the Westinghouse Purchase Specification which defined an anticipated cyclic operation histogr over the plant lifetime in order to provide for fatigue evaluation. This sideration becomes important when determining the percentage of life usage of a pressure vessel expended from operation under cyclic pressure and thermal stress conditions throughout the lifetime of the vessel. With knowledge of the cyclic operation of the components up to the present time, one can estimate the remaining cyclic life of the components until fatigue failure of portions of the components occurs.

In order to ascertain the adequacy of nuclear pressure vessel components, therefore, it is necessary to maintain a careful record of the cyclic operation of the system such that the expended lifetime of the structural parts of these components can be computed. Fatigue is considered one of the major causes of pressure vessel failure. It is a safety consideration that attention be given to the present and remaining cyclic life of pressure vessel components such that the vessel (or critical parts thereof) may be replaced at the appropriate time before failure or its usage extended beyond the originally intended useful lifetime of the component.

This inspector evaluated the records kept of the cyclic history to date of the Haddam Neck primary system and the procedures utilized when the transient operation violates any such limits provided in the Technical Specification (such as heating and cooling rates of components).

4.0 Documents Reviewed

UFSAR Connecticut Yankee Haddam Neck Plant
Technical Specification (Section 3.4.9)
Combustion Engineering (CE) reactor pressure boundary fatigue analysis
Westinghouse (WHGS) Haddam Neck plant life extension (PLEX) feasibility study
Connecticut Yankee (CY) Haddam Neck Plant operational logs (sample)
Yearly Net Generation (Avg MWe) Charts for CY
Component Cycle Tracking Procedure (ODI No. 140)
Letter, Pittman to Mroczka, Nuclear Review Board Mtg. 12/3/86, PLEX
Letter, Ahern to Ely, 2/1/90, Steam Generator Replacement Project, PLEX
Letter Cirilli to Pittman, CY Evaluation of Fatigue Usage Factors for PLAR 85-13
Proposed License Amendment Report PLAR 85-13
Crack Growth Estimation Report for Determining Inspection Intervals (SIA) Report No. SIR 89 025, May 1989
Interpretations of Boiler and Pressure Vessel Code Section VIII code Cases 1270N-5 and 1273 N-7

5.0 Findings

The inspector reviewed the available information at NUSCo relating to monitoring the operational history of the Haddam Neck Plant and the results of converting this operational history into an assessment of the fatigue life of these components. Of importance in this review was the results of a Westinghouse life extension feasibility study which provided for an assessment of remaining life expectancy of the reactor pressure vessel and steam generator. One effect of this study was to underscore the need for comprehensive structural evaluation and monitoring of operational data.

5.1 Haddam Neck Procedures in Implementing Technical Specification Limitations

The inspector reviewed the licensee system for documenting cyclic operating transients and reporting approaching deviations from the limits in Haddam Neck procedure ODI No. 140. Transients included in the documentation are reactor system heatup and cooldown, pressurizer cooldown, loss of load, reactor coolant pump loss of flow, and reactor trip. For the RCS heatup and cooldown, 150 cycles of each are allowed for changes of 200F to 550F and 550F to 208F average coolant temperature with the rate limited to 100F per hour. The pressurizer cooldown from 650F at 200F per hour is limited to a maximum of 150 cycles. The loss of load transients (60 cycles max), loss of flow in one reactor coolant loop (60 cycles max), and 300 reactor trip cycles are also subject to reporting by the operations supervisor to the engineering department when the limiting numbers of transients are being approached.

The Operations Supervisor is responsible for maintaining a dedicated log of all normal and extraordinary transient cycles. Control room operators are responsible for documenting the applicable transients in the control room log. The operations supervisor must notify the engineering department when cyclic or transient limits are being approached. The inspector was shown a computerized daily operating log, dated September 30, 1985, which included operational data for percent power, rod position, reactor coolant system, RCS seal system, containment, pressurizer, and steam generator. The inspector was furthermore shown the yearly operating MWe net power generation charts over the period 1968 through 1989 which provide for an overview of the plant ordinary and extraordinary transient operation.

Review by the inspector indicated that the licensee is providing for the documentation of transient operation in a manner consistent with continuing evaluation of the expended fatigue life of the components. A procedure is available for reporting to the engineering department any nonconformities with respect to the Technical Specification. To date, there were no reported deviations from the Technical Specification over the operating period of the plant related to transient operation.

5.2 Applicable Codes and Standards

The CY Haddam Neck reactor pressure vessel was designed in accordance with the 1959 edition of the ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels with addenda through 1960. The ASME Section VIII was supplemented by the "Tentative Structural Design Basis for Reactor Vessels and Directly Associated Components" together with Westinghouse Equipment Specification 67E195, Rev. 0. Consistent with the Westinghouse Equipment Specification , a complete structural and fatigue analysis of the reactor vessel was performed by Combustion Engineering in Analytic Report 1090 and Addendum CENC 1090 A1. The 1963 Edition of Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Vessels" was used as a reference guide. The Reactor vessel predates the issuance of NRC Regulatory Guides.

The Westinghouse steam generators were designed and manufactured to the rules of Section VIII of the ASME Boiler and Pressure Vessel Code. "Unfired Pressure Vessels" and Code Cases 1270N and 1273N which provides for structural analysis under extended stress considerations but without evaluation of fatigue life expectancy.

5.3. Combustion Engineering Reactor Vessel Pressure Boundary Fatigue Evaluation

The inspector reviewed the Combustion Engineering analytic report No. 1090 and its addenda 1090 A-1 completed in January 1965 and November 1965, respectively, which provided for a fatigue evaluation of the reactor pressure vessel system. The report evaluated the ability of the reactor pressure vessel to sustain the fatigue effect of cyclic operation over its 30 year design lifetime. The inspector noted that although fatigue analysis was not required at the time of reactor vessel design under the rules of ASME section VIII and code cases 1270N/1273N, the rules of "Tentative Design Basis for Reactor Vessels and Directly Associated Components" were used to supplement the reactor vessel design evaluation with fatigue life usage considerations.

5.4 Westinghouse Study of Life Extension Feasibility

The inspector reviewed a life extension feasibility study by Westinghouse Electric Corporation which was suggested by the licensee as an indication of the attention given the cyclic history evaluation of the Haddam Neck plant.

In 1985, Westinghouse Electric Corporation was retained by Connecticut Yankee to perform a study of the feasibility of extending the operating lifetime of the Haddam plant beyond its design lifetime. In order to provide for the assessment of remaining life of the plant components, it was necessary for Westinghouse to perform a review of the historically significant transients to determine the cumulative fatigue damage to critical plant components resulting from the operation to date. The inspector reviewed the results of this study at the Berlin office and subsequent to the visit there.

Westinghouse considered both normal plant operating transients and extraordinary plant operating transients. The Westinghouse approach compiled a composite record of Haddam operating experience from a wide range of sources both internal and external to Haddam. Much judgement was required to extract the transient operational data from the total data base. Sources used for transient operational data were reliability data bases including the NRC Graybook, LERs, NUREG 0826, BEARDS listing of plant outages and derating events, and graphs of plant MW output. Plant operating logs were found difficult to use for the purpose of this study.

Future operating transient event predictions by Westinghouse were based on conservative extrapolations of typical Westinghouse plant operating history. In these predictions, conservative interpretations of the severity of past transient operating data was used.

Westinghouse determined that four critical components would be considered in the study, namely, reactor vessel shell/supports/internals, steam generators, reactor coolant piping, and containment. The design life of the Haddam plant was originally 30 years. On the basis of CE reactor fatigue analysis, NUSCo reevaluated the reactor design life to be upgraded to 40 years. The target for life extension was an additional 20 years, to a total life expectancy of 60 years. To date (1990), 23 years of life have been expended and there remain 17 years of design lifetime of the reactor pressure vessel.

Westinghouse determined that for the extension of reactor pressure vessel system consideration must be given two major concerns: fatigue life usage due to additional cycles and pressurized thermal shock (PTS) as the beltline material experiences higher radiation induced embrittlement. Based on revised fatigue considerations, therefore, it was concluded by Westinghouse that the extension of the 40 year operating life to 60 years was feasible. Analysis by NUSCo of the radiation effects on reactor vessel belt material indicate that the original 30 year life can be achieved and with changes in fuel management a life expectancy of up to 55 years is feasible.

In evaluating the steam generators' life expectancy, Westinghouse determined that a design life of 30 years and beyond could be achieved with some derating (at least one percent). Fatigue evaluations indicated concern for three component features at an extended life of 60 years. These include the feedwater nozzle, divider plate and the girth weld. However, the features are not limiting during the 30 year design lifetime and are of such a nature as to be repairable. There have been replacement studies for the steam generators should replacement be necessary. Economic studies play an important role in the timing of such replacement.

5.5 Summary and Conclusions

As a result of review, of the licensee documentation related to the monitoring of plant cyclic history and evaluating the effect of the cyclic history on remaining life expectancy, the following was observed:

(1) The licensee maintains a procedure (OD No. 140) for monitoring plant operation by using a system for reporting deviations of operating conditions from Technical Specification requirements to the engineering department. However, this procedure does not include comparison of the actual numbers and characteristics of the system transient operating cycles with those anticipated in the original design calculations such that estimation of remaining design fatigue life may be evaluated.

(2) The licensee has supported studies of component operating life usage toward evaluating remaining life expectancy for proposed component life extension. The inspector has been shown results of original reactor vessel fatigue evaluation and studies by Westinghouse of remaining life expectancy of primary system components. Many of these studies have been based on comparisons with analyses of similar components under typical operating conditions at other plants. No procedures exist for continuing accountings or analyses based on updated actual cyclic experience beyond the foregoing studies.

6.0 Unresolved Items

Unresolved items are matters about which information is required in order to ascertain whether they are acceptable, violations, or deviations. There are no unresolved items.

7.0 Management Meetings

The licensee's management was informed of the scope and purpose of the inspection at the entrance interview on July 31, 1990. The findings of the inspection were discussed with the licensee's representatives during the course of the inspection and presented to the licensee's management at the exit interview on August 3, 1990. (See section 1.0 for those in attendance.) Although the interviews and inspection took place at the Berlin, Connecticut offices of NUSCo, both entrance and exit interviews were held at the Haddam site.

No written material was provided to the licensee by the inspector. The licensee did not indicate that proprietary information was involved within the scope of this inspection.