U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NO.: 50-293/94-23

LICENSEE: Boston Edison Company

FACILITY: Pilgrim Nuclear Power Station
Plymouth, Massachusetts

DATES: September 25-30, 1994

INSPECTORS: A. Lohmeier 10-21-94
Alfred Lohmeier, Sr. Reactor Engineer Date
Materials Section
Division of Reactor Safety

A. Lohmeier for John Fair 10-21-94
John Fair, Sr. Mechanical Engineer Date
Mechanical Engineering Branch
Nuclear Reactor Regulation

APPROVED BY: Michael C. Modes 10/21/94
Michael C. Modes, Chief Date
Materials Section
Division of Reactor Safety

Areas Inspected: This inspection covers the review of the BECo fatigue reevaluation of the reactor components to determine whether their anticipated fatigue life will be expended prior to the end of the licensed 40-year life. The changes in the estimated numbers of transient cycles, analytic methods used in stress analysis, and resulting revised lifetime cumulative fatigue life usage factors (CUFs) from the original design fatigue evaluation were reviewed. Changes in fatigue life monitoring and evaluation procedures were reviewed together with proposed changes in the updated final safety evaluation report.

Results: The estimated numbers of lifetime transient cycles, determined from the reevaluation study, varied from the original values used in the vessel design. Reevaluation of the CUFs using the currently estimated cycles resulted in reduction of lifetime CUF reductions for all reactor components to below 0.1, with the exception of the feedwater nozzle, for which the lifetime CUF is less than 0.8.

The inspection team finds no reason to disagree with the analysis performed by BECo, or the conclusions drawn from this analysis. The conclusion of this reevaluation is that the end-of-life CUFs for the various sections and components of the vessel analyzed will remain less than the Code limit of unity. URI 50-293/
94-01-01 is closed.

PNPS has proactively pursued a comprehensive fatigue-monitoring system that alerts management to the status of accumulated fatigue cycles through the remaining plant life. After reconciliation of the ASME Code Rules used in the reevaluation, BECo will amend their UFSAR to reflect the results of the new fatigue analysis.

DETAILS

1.0 SCOPE OF INSPECTION (Inspection Procedure 37700)

The scope of this inspection covers the review of the BECo fatigue evaluation of the reactor vessel components to determine whether the anticipated fatigue life will be expended prior to the end of the licensed 40-year life. The inspectors reviewed changes in the estimated numbers of the transient cycles to which the reactor will be subjected and compared them with the numbers of cycles for each transient considered in the original fatigue evaluation. Analytic methods used in stress analysis were reviewed and compared with the methods used in the original reactor calculation method. Revised lifetime CUFs were reviewed and compared with those calculated in the original evaluation of lifetime CUFs. Changes in fatigue life monitoring and evaluation were reviewed together with changes in the updated final safety evaluation report (UFSAR).

2.0 FINDINGS

2.1 Introduction

Inspection 50-293/94-01, performed in early January of this year, found that Boston Edison (BECo) had implemented a reevaluation of the original fatigue analysis of the Pilgrim reactor vessel. An unresolved item (URI) 50-293/94-01-01 was identified in the report.

Inspection Report 50-293/94-01 noted, for some transient operating cycles, the reported numbers of cycles, after 21 years of operation, had exceeded the numbers of cycles assumed in the original fatigue analysis. Recognizing the conservatism used in the original analysis and that BECo had recognized the fatigue evaluation problem earlier and embarked on a reevaluation of the reactor component fatigue life, the inspector issued URI 50-293/94-01-01 only for the purpose of tracking the issue.

Inspection Report 50-293/94-01 also noted that the numbers of startup, power increase, loss of feedwater pumps, and safety-relief valve blowdown transients experienced after 21 years of operation, had exceeded the numbers of transients used originally to evaluate the fatigue adequacy of the vessel in CE Design Report CENC 1139, dated February 15, 1971. Furthermore, linear projection of numbers of transients anticipated for the 40-year license lifetime exceeded the numbers of transients assumed in the original design of the reactor vessel. These transients are startup, power increase, loss of feedwater pumps, safety relief valve blowdown, and other scrams.

The inspection report reviewed the resulting CUFs for some critical reactor vessel components as reported in the original design report and the Pilgrim Nuclear Power Station (PNPS) UFSAR. The cumulative usage factors for critical reactor vessel components reported in the UFSAR were the vessel shell in the core region (.301), closure studs (.786), closure flanges (.650), bottom head/support skirt juncture (.306), core shroud support (.374), feedwater nozzle (.713), recirculation inlet nozzle sleeve (.360), and control rod drive housing stub tube junction (.105).

The inspection report discussed, in cases where the actual numbers of transients had exceeded the numbers assumed in the original fatigue calculation, the necessity of demonstrating through analysis that the CUF limitation of 1.0 in the reactor vessel components had not been exceeded. Prior to the January inspection, BECo had already contracted this study to an outside engineering firm. In anticipation of the results of the contracted study, and on the basis of the conservatism recognized by the inspector in the original fatigue evaluation, the inspector issued URI 50-293/94-01-01 for tracking purposes only.

2.2 Reevaluation of Operating Transients

In the BECo reevaluation study, the inspectors found that some of the estimates for operating cycles during the original reactor vessel design process were low. The estimates were made with only limited operating experience. The revised numbers of lifetime cycles for each transient, resulting from the reevaluation, were compared with the numbers of cycles used in the original fatigue evaluation. The results were indicated in the reevaluation report as follows:

	Cycle Counts	
Event	Original Analysis	Revised Analysis
Boltup	N/A	22
Hydro	130	22
Cold Startup	120	212
Hot Standby Startup	120	337
50% Power Reduction	14,600	379
Loss of Feedwater Heaters	80	10
Loss of Feedwater Pumps	10	26
Turbine Generator Trips	40	27
Other Scrams	147	132
Full Power Recirc Startup	5	16
Power Reduction to Hot Standby	118	176
Shutdowns	118	145
Safety Valve Blowdown Shutdown	2	47
Refueling Floodup Shutdown	118	20
Unbolting	N/A	22

The originally assumed cycles were supplied in the original equipment specification for the reactor pressure vessel. The revised analysis cycles were based on the cycles to-date, plus an extrapolation from the cycles to date, using regression analysis based on historical experience data points. Review of these data indicates that, while some transient cycles have been reduced, those estimated for cold start, hot standby startup, loss of feedwater heaters, full power recirculation startup, power reduction to hot standby, shutdown, and safety valve blowdown shutdown have increased.

The BECo report also noted that the original analysis assumed that each design cycle produced the maximum range of stress, whereas the revised cycle evaluation more accurately accounts for the actual stress range for each event in order to determine its incremental contribution to fatigue life usage.

The inspectors found the estimated numbers of lifetime transient cycles, determined from the reevaluation study, varied from the original design values. For seven transients, the original cycles were underestimated. For eight transients, the cycles were overestimated. The inspectors found that the revised lifetime cycles were based on a more realistic historical background than before.

2.3 Reevaluation of Lifetime Cumulative Usage Factors

The inspectors reviewed the reevaluation of CUFs. The results of the reevaluation, as reflected in the reevaluation report, are as follows:

Fatigue Usage Factors

Component	Original Analysis	Revised Analysis
Closure Region	.77	.049
Closure Studs	.79	.07
Bottom Head and Support Skirt	.309	.044
Feedwater Nozzle		
System Transients	.545(1)	.637
Combined (Rapid and System)	.678(1)(3)	<.8(4)
Steam Outlet Nozzle	(meets ASME exclusion rules)	
Recirculation Inlet Nozzle	.97	.037
Recirculation Outlet Nozzle	.751	excluded
Core Spray Nozzle	.437	.01
Vessel Shell	.435	.012
Vent Nozzle	(meets ASME exclusion rules)	
Instrument nozzle	(meets ASME exclusion rules)	

(1) Based on Finite Element Analysis

(3) Requires refurbishment of the sparger six times during remaining plant life.

(4) No refurbishment of the nozzle is necessary.

The reevaluation report shows that there is substantial reduction in CUFs for most reactor components due to reductions in peak stress and a more accurate representation of the actual stress range of each event. Increased cumulative usage factors due to increased cold startups and loss of feedwater pump transients were offset by reductions in stress range resulting from improved analysis methods.

The inspectors noted that new analysis was performed of the feedwater nozzle under existing system cycling and rapid thermal cycling. As a result of this reevaluation, it was determined that the end-of-life cumulative fatigue usage factor is expected to be less than 0.8 in the absence of any seal refurbishment. A finite element analysis of the nozzle performed in 1979 for rapid thermal cycling showed the cumulative fatigue usage factor was less than 1.0 assuming 6 seal refurbishments over the plant life. The current evaluation of the feedwater nozzle analysis found errors in friction factors assumed, an overconservative model for predicting secondary seal leakage flow, and an overconservative model for corrosion rates at the Inconel-Inconel piston ring gaps.

It was noted in the original design analysis report that the fatigue evaluation calculations were, for the most part, performed manually. With the advent of the electronic computer, reevaluation calculations were performed more rapidly with greater accuracy. Assumptions necessary to expedite rapid manual calculation of these complex structural systems were not required when using the electronic computer. As a result, more precise results are obtained and many more cases of the problem could be solved.

As a result of the reevaluation of the fatigue evaluation analysis, the inspectors found the original design fatigue evaluation analyses was very conservative. The reevaluation of the cumulative fatigue life usage resulted in reduction of lifetime cumulative fatigue life reductions for all reactor components below 0.1, with the exception of the feedwater nozzle. However, a new fatigue evaluation of the feedwater nozzle indicated a lifetime cumulative fatigue usage factor less than 0.8.

2.4 Required ASME Code Reconciliation

The new evaluation was performed to the requirements of the 1989 Edition of ASME, Section III; an edition endorsed by the NRC. Licensees are allowed to use newer editions of the Code, provided they perform a code reconciliation. The edition used by BECo for the new evaluation is not the edition used for the original design of the vessel and will require a reconciliation. BECo's initial review of the differences has not revealed any problems. After the reconciliation is performed, BECo intends to amend their UFSAR to reflect the results of the new fatigue analysis.

2.5 Changes in Fatigue Monitoring and Evaluation Procedures

The inspectors noted in Inspection Report 50-293/94-01 that PNPS has been in compliance with Technical Specification Section 6.10.B.7 relating to retention of records. As a consequence of PNPS interest in fatigue monitoring, a fatigue-monitoring procedure is presently under development that will provide for comprehensive monitoring of plant operational transients. An outline of a draft fatigue-monitoring procedure was reviewed by the inspectors and found to be as follows:

(1) Normal procedural boilerplate containing definitions, precautions, acceptance criteria, engineering evaluation processes, and data collection procedures.

(2) Expansion of the number of original GE SIL 318 events to be monitored from 12 to 19.

(3) Data collection and counting process changed from a quarterly to a monthly basis.

(4) Data to be monitored will be monthly performance reports, operations daily log books, scram reports, data collection and station performance calculation, and vessel heatup and cooldown. The system engineer will tabulate event counts, categorize them and provide for an independent review. A computer file will be updated regularly.

(5) A cognizant mechanical design engineer will review the data and comment.

(6) After the review process, data collection will be approved by applicable departments. A followup memo will be generated and filed in the records retention facility.

It is intended that the new procedure will be implemented subsequent to MCO-10.

The inspectors noted that PNPS has proactively pursued a comprehensive fatigue- monitoring system that alerts management to the status of cumulative fatigue cycles through the remaining plant life.

2.6 Unresolved Item URI 50-293/94-01-01

Based on this inspection, the inspection team finds no reason to disagree with the analysis performed by BECo, or the conclusions drawn from this analysis. The reassessment of the CUFs of the reactor vessel at Pilgrim Station is comprehensive and thorough. The assumptions used, the computational methods applied, and the conclusions drawn are correct and complete. The conclusion of this reevaluation is that the end-of-life CUFs for the various sections and components of the vessel analyzed will remain less than the Code limit of unity. Therefore, NRC URI 50-293/94-01-01 is closed.

3.0 CONCLUSIONS

- The estimated numbers of lifetime transient cycles determined from the reevaluation study varied from the original design values. Seven transient cycles were underestimated and eight transients cycles were overestimated. The revised lifetime cycles are based on actual plant operational data.

- The original design fatigue evaluation analyses were overly conservative. Reevaluation of the cumulative fatigue life usage using currently estimated cycles resulted in values for all reactor components below 0.1, with the exception of the feedwater nozzle for which the lifetime cumulative fatigue usage factor is less than the code limit of 1.0.

- The NRC finds no reason to disagree with the analysis performed by BECo, or the conclusions drawn from this analysis. The reassessment of the fatigue endurance of the reactor vessel at Pilgrim Station is comprehensive and thorough. The assumptions used, the computational methods applied, and the conclusions drawn are correct and complete. The conclusion of this reevaluation is that the end-of-life CUFs for the various sections and components of the vessel analyzed will remain less than the Code limit of unity. Therefore, NRC URI 50-293/94-01-01 is closed.

- The inspectors note that PNPS has proactively pursued a comprehensive fatigue-monitoring system that alerts management to the status of cumulative fatigue cycles through the remaining plant life.

- The new evaluation was performed to the requirements of the 1989 Edition of ASME, Section III; an edition endorsed by the NRC. Licensees are allowed to use newer editions of the Code, provided they reconcile the differences. The ASME edition of Section III used by BECo for the new evaluation is not the edition used for the original design of the vessel and will require a reconciliation. BECo's initial review of the differences has not revealed any problems and after the reconciliation is performed, BECo will amend their UFSAR to reflect the results of the new fatigue analysis.

4.0 MANAGEMENT MEETINGS

The inspectors met with PNPS engineering and licensing personnel at the entrance meeting on September 26, 1994, and at the exit meeting on September 30, 1994, at the PNPS in Plymouth, Massachusetts. Also in attendance at the exit meeting were representatives of the State of Massachusetts, the local community of Plymouth, and the public media. The names of personnel at the entrance and exit meetings, and those contacted during the inspection are shown on Attachment 1. The findings of the inspection were discussed with management personnel at the January 25, 1994, exit meeting. The licensee did not disagree with the findings of the inspector.

5.0 EXIT REMARKS

Attachment 2 contains a copy of a prepared statement made at the exit meeting and released for general distribution after the inspection was concluded.

Attachment: Persons Contacted

ATTACHMENT 1

Persons Contacted

The following persons were contacted at the entrance meeting on September 26, 1994, at the exit meeting on January 25, 1993, and during the course of the inspection:

Boston Edison Power Company

* E. T. Boulette	Senior Vice President, Nuclear
* D. W. Ellis	Compliance Supervisor (Acting)
* R. V. Fairbank	Manager, Regulatory Affairs & Emergency Preparedness
* A. Flanagan	Boston Edison Company, NUS Information
* J. M. Fulton	Senior Counsel
C. Garrow	Sr. QA Engineer, Quality Assurance
* J. P. Gerety	Manager, Mechanical Systems
* L. Harrington	MDPH Radiation Control Program
* J. D. Keyes	Manager (Acting), Licensing/Compliance Division
* P. L. Markson	Nuclear Information
C. Martin	Senior Engineer, Systems & Structural Analysis
P. L. Murkson	Communications Specialist
H. V. Oheim	Manager, Nuclear Engineering Systems Division
* L. J. Olivier	V. P., Nuclear Operations, Station Director
* J. S. Roberts	Principal Engineer
W. C. Rothert	General Manager, Technical
R. Schifone	Quality Assurance/Quality Control
* D. F. Tarantino	Manager, Nuclear Information

Altran Engineering and Management Consultants

M. A. Eissa	Vice President, Engineering Services
P. K. Shah	Senior Consultant

State of Massachussets

* J. B. Muckerheide	Engineer, State of Massachusetts

Plymouth Township

* G. Long	Old Colony Memorial

Media Representatives

* E. Copp	WQRC News
* J. Juliano	NUS
* R. Stone	WATD - FM

U. S. Nuclear Regulatory Commission

* J. MacDonald	Senior Resident Inspector
* R. H. Wessman	Chief, Mechanical Engineering Branch
* J. T. Wiggins	Director, Division of Reactor Safety
* C. I. Wu	Mechanical Engineer

An asterisk (*) indicates attendance at the exit meeting.

ATTACHMENT 2

The following is the prepared statement made by Mr. Michael C. Modes, Chief, Materials Section, Division of Reactor Safety, Region I at the exit meeting on September 30, 1994 and released for general distribution after the inspection was concluded:

As the consequence of an inspection performed by Mr. Lohmeier in early January of this year, the NRC noted in report 94-01 that Boston Edison (BECo) had implemented a reevaluation of the original fatigue analysis of the Pilgrim reactor vessel. An unresolved item was generated in the report so the NRC could plan for the conclusion of the reevaluation. This week the NRC performed an inspection to review the final analysis and determine the status of the unresolved item. The subject of this review is reactor vessel fatigue.

In some cases metal has a limited endurance when subjected to repeated loads. Fatigue strength is the property of a metal that measures its ability to sustain these repeated loads. In the design of a pressure vessel it is important to evaluate the ability of the vessel to sustain these loads. The vessel can then be designed to safely accommodate these loads during its useful life.

The Pilgrim reactor vessel was originally designed and fabricated by Combustion Engineering (CE). The calculations used for the fatigue evaluation, and its assumptions, are reported in CE report CENC 1139, dated February 15, 1971. This report is a thermal and structural analyses in conformance with General Electric (GE) Specification 21S1110AB, Rev 3, dated August 19, 1970 and in compliance with the rules of American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, 1965 Edition with 1966 Addenda. It also utilizes the Code interpretations expressed in references 1332-4, 1335-2, 1336, and 1339-2.

In order to design the vessel, limiting parameters were initially established by GE. These design parameters consisted of things such as design pressure (1250 psig) and temperature (575° F), and design life objective (40 years). In addition, GE Specification 21A1110AB and GE drawing 730E941 originally identified anticipated operational transients judged to be important enough to be considered in the design. For the purpose of calculating the 40 year fatigue endurance of the vessel, the number of occurrences assumed, for these transients, also was established. For example, the original designers assumed the vessel would be subjected to 120 startups at a heat up rate of 100 °F per hour. In other examples, the original designers assumed the vessel would be subjected to five sudden pump starts of a cold recirculation loop or 14,600 occurrences of 50% power operation. These are three examples of the thirteen transients deemed important to the original design.

Utilizing these assumptions, CE calculated the fatigue usage of various sections of the vessel utilizing methods accepted by ASME and embraced by the NRC. With the exception of the limited application of some computer codes in design (Seal-Shell-2), the calculations basically were performed manually. Because the calculations were performed by hand, conservative methods were used that would reduce the total number of computations required. For example, in some cases CE determined the highest strain range in the thirteen transients, for a particular section of the vessel, and applied it to all thirteen transients; a calculation method called "lumping". This method effectively reduced the calculations, for a particular section, from thirteen to one. The final fatigue life of the vessel was then reported, for a section, in the form of a cumulative usage factor (CUF). The CUF is a measure of the fatigue experienced by the components that are subject to repeated cycles of stress caused by transients. As long as the number remains below unity, the vessel section has not approached the end of its fatigue life.

The current, extensive application of computers has obviated the need for these conservative calculation methods. BECo has taken advantage of this evolution to redo the original calculations and refine the results. For example, BECo was able to use all thirteen transient conditions and compute the fatigue calculation more quickly than it took to originally perform the one, lumped, hand calculation. The computer also has led to refinements in determining thermal and pressure stress distributions in complicated geometries. For example BECo used computer tools such as finite element analysis (FEA) to determine a more precise location and value for stress in a nozzle.

In addition, the validity of the original transient assumptions now can be checked against accumulated operating experience. One of the original transients the designers assumed would happen was the previously mentioned sudden start of a cold recirculation loop. This transient is administratively controlled in Pilgrim technical specification Paragraph 3.6.A.4. This transient is not administratively allowed and, as a consequence, has not occurred since the plant commenced operations. Therefore, BECo was able to eliminate it from consideration for some fatigue calculations.

Reassessment of the conservatism, refinement of the analysis tools and the application of the computer to the computationally intensive parts of the fatigue predictions have contributed to the fatigue computations redone by BECo. In addition, a better operational understanding of the transients, based on experience, also contribute to changes in the calculated usage factors (and therefore the end-of-life cycle count) for the eight critical areas of the vessel identified in the BECo Updated Final Safety Analysis Report (UFSAR), Appendix C, Table C.3-8.

The analysis was performed to the requirements of the 1989 Edition of ASME, Section III; an edition endorsed by the NRC. Licensees are allowed to use newer editions of the Code provided they reconcile the differences. The edition used by BECo is not the basis used for the original design of the vessel and will require a reconciliation with the previous edition. BECo's initial review of the differences has not revealed any problems. After the reconciliation is performed BECo will amend their UFSAR to reflect the results of the new fatigue analysis.

Based on this inspection the NRC finds no reason to disagree with the analysis performed by BECo or the conclusions drawn from this analysis. The reassessment of the fatigue endurance of the reactor vessel at Pilgrim Station is comprehensive and thorough. The assumptions used, the computational methods applied and the conclusions drawn appear to be correct and complete. The conclusion of this reevaluation is that the end-of-life CUFs for the various sections and components of the vessel analyzed will remain less than the Code limit of unity. Therefore, NRC unresolved issue 50-293/94 01 01 is closed.