U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-309/89-20

Docket No. 50-309

License No. DPR-36

Licensee: Maine Yankee Atomic Power Company
83 Edison Drive
Augusta, Maine 04336

Facility Name: Maine Yankee

Inspection At: Wiscasset, Maine

Inspection Conducted: October 23 - 27, 1989

Inspectors: Roy K Mathew — 12/13/89
R. K. Mathew, Reactor Engineer, Plant Systems Section, EB, DRS — date

for Roy K Mathew — 12/13/89
J. M. Trapp, Senior Reactor Engineer, Special Test Programs Section, EB, DRS — date

Approved by: C J Anderson — 12/13/89
C. J. Anderson, Chief, Plant Systems Section, EB, DRS — date

Inspection Summary: Routine, announced inspection on October 23 - 27, 1989 (Report No. 50-309/89-20)

Areas Inspected: The focus of this inspection was the licensee's control of design, design changes, modifications and temporary modifications. Also included in the scope of this inspection were organization, staffing, communications, quality assurance, training and management support.

Results: One violation and one deviation were identified during the review of engineering modification packages. They are discussed in Sections 8.1 and 8.2.

DETAILS

1.0 Persons Contacted

1.1 Maine Yankee Atomic Power Company (MYAPCO)

S. A. Bailey, QC Supervisor
* R. Blackmore, Plant Manager
B. Bickford, Supervisor, Training
* E. T. Boulette, Vice President, Operations
R. Branscomb, Engineer
* R. Crosby, Senior Licensing Engineer
* J. Frothingham, Manager, Quality Programs
* J. H. Garrity, Vice President, Licensing and Engineering
* T. M. Gifford, Project Engineering Section Head
H. Gilpatrick, Principal Engineer, Design
J. Herbert, Manager, Engineering
C. James, Principal Engineer, Mechanical
* S. Leclerc, Quality Program Section Head
* S. McAallister, Acting Lead, Nuclear Safety Engineering
L. McCabe, Senior Engineer, Mechanical
* S. Nichols, Licensing Section Head
* C. Shaw, Plant Engineering Section Head
J. Speed, Engineer

1.2 State of Maine Representative

* P. Dostie, Nuclear Safety Inspector

1.3 U.S. Nuclear Regulatory Commission

*R. Freudenberger, Resident Inspector

* Denotes those present at the exit meeting

2.0 Organization/Staffing/Management Support

Maine Yankee receives engineering/technical support from the corporate engineering staff of the Yankee Nuclear Services Division (YNSD) at Bolton, Mass., the corporate engineering staff at Augusta, Maine and on-site plant engineering. The manager of the Plant Engineering Department (PED) oversees the engineering program. The managers of the Nuclear Engineering and Safety Engineering Departments are responsible for licensing, safety review and operational support for the plant.

Thirty two engineers from the plant engineering department and approximately twenty engineers from Yankee Atomic Services provide engineering and technical support for the plant. The licensee stated that the plant engineering staffing is currently maintained at an acceptable level. However, the PED has made a proposal to increase the engineering staff by 10 - 15% to reduce the existing backlog of engineering projects and to support additional review and analysis of selected operational events. They are involved in approximately 100 projects at any given time. Contractor personnel are hired to provide services on an as needed basis. Each contract engineer is supervised by the cognizant Maine Yankee engineering supervisor.

The organization for the 1990 refueling outage has been defined. The goals and objectives have been outlined to meet the corporate goals. The PED corporate goal is to complete the approval of Conceptual Package Assessment's (CPA's) and EDCR's for tasks assigned for the 1990 refueling outage by June 30 and December 31, 1989 respectively. The licensee's successful completion of the 1988 refueling outage is a prime example of the effectiveness of their planning and scheduling of outage activities. The PED met its goal by generating and performing the final PORC review of Engineering Design Change Requests (EDCRS) sufficiently in advance to support the refueling outage. The PED and their contractors completed 113% of their scheduled work during the last outage.

The licensee has established a viable system for controlling the engineering workload and for establishing and revising priorities. A five core cycle (cycles 11-15) schedule establishes project priority and tracks engineering activities. The licensee has established a prioritization system where the most safety significant projects are given a priority 1. Priority 1 projects are scheduled to be completed before the end of the next (cycle 11) refueling cycle. Lower priority projects are scheduled for completion during Cycle 12-15. The plant engineering manager is responsible for this program and a weekly schedule update. The scheduling process has proven effective as evidenced by all priority safety significant projects being on schedule.

The licensee's effort to improve engineering support to the plant was evidenced by the licensee management's decision to relocate the plant engineering staff at Augusta to the site at the end of year 1989.

The engineering staff is well staffed with experienced and degreed engineers. Their approach to resolving technical issues is to do the job right the first time. The successful completion of breaker trip device modifications for 480V circuit breakers, capacitor bank installation at the switchyard and main transformer replacements reflected technically sound engineering work.

3.C Quality Assurance (QA) Involvement in Technical Support

The quality assurance area was reviewed by the inspectors to evaluate QA involvement in assessing the quality of engineering services. QA (YNSD) and on-site QA support staff are responsible for conducting engineering audits. One QA audit of EDCR's is performed each year. A review of QA Audit Report 89-07 revealed several significant findings including weaknesses in the EQ area. Quality Assurance involvement in monitoring engineering effectiveness is adequate.

4.0 Technical Staff Training

The licensee's training program was inspected to evaluate the adequacy of training given to the engineering support personnel. The inspector discussed the program with the training supervisor and reviewed the procedures and records. The technical staff training program is designed to provide Maine Yankee professional personnel and contractor personnel with supplemental training to perform engineering activities properly, and in a safe, effective and efficient manner. The program includes initial and continuing training classes and offsite industry offered courses. The licensee recognizes the need for an effective training program and has addressed this need by the establishment of a well-staffed training department with the resources and the authority to conduct an effective program.

5.0 Communications

The PED actively participates in the daily morning meetings to discuss daily work activities. The morning meeting is the primary communication channel between the plant staff organizations. Following the morning meeting, engineering and technical support scheduled activities are reviewed and updated to support station requirements. The corporate staff engineers visit the plant at least once every two weeks. The design engineers are responsible for projects until the installation is completed and turned over to operations. This continuous interaction between engineering and the plant site is important for effective communications. A minor weakness in the interface between PED, YNSD and Instruments and Controls (I&C) was identified during the recent QA audits. The licensee is planning to improve communications by requiring design input from the cognizant plant system superintendent and the plant manager.

6.0 Management Initiatives and Responsiveness to NRC Initiatives

The licensee management's initiatives to improve plant safety and performance was evidenced by the following examples:

1) The licensee developed a Comprehensive Equipment Performance Program (CEP) to monitor and improve the reliability and performance of the key plant systems components and equipment at Maine Yankee;

2) The licensee initiatives to develop a comprehensive design basis document and functional verification of the safety system operation to design criteria continues to progress and is scheduled to be completed by Cycle 13;

3) A commitment to develop safeguard instrument loop drawings, loop accuracy calculations, and setpoints within 4 years; and

4) A commitment to develop and implement a configuration management procedure to control and maintain the management control systems and documents to properly reflect plant configuration by 1990.

These licensee initiatives are indicative of a commitment to the long term strong performance in engineering and technical support.

The licensee's management continues to demonstrates responsiveness to NRC initiatives as evidenced by the following examples:

1) NRC SSFI audit follow-up issues are planned and scheduled to be completed by December 31, 1991;

2) NRC Information Notices 88-86, 88-75 and 88-98 are assigned a higher priority to complete the work before the end of 1990 refueling outage; and

3) A commitment to generate a report in response to IE Bulletin 88-04, "Potential Safety Related Pump Loss" by December 31, 1989.

7.0 Temporary Modifications

Temporary modifications are administratively controlled by procedure 0-14-2, Rev. 9, "Temporary Modification Control." The temporary modifications reviewed during this inspection are listed in Attachment B.

All temporary modifications are reviewed by the Nuclear Safety Section to determine if a "50.59 Safety Evaluation" is required. The Nuclear Safety Section also completes the safety evaluations when required. The safety evaluations reviewed were found to be adequate.

Administrative requirements for the temporary modifications, such as tagging and log reviews were found to have been properly completed in accordance with Procedure 0-14-2. A biannual review of all outstanding temporary modifications by the PORC Committee has successfully reduced the number of long term outstanding temporary modifications. The licensee maintains the configuration changes made by the temporary modifications on the control room drawings. Drawing updates were a positive means to control plant configuration changes made by temporary modifications. The control and documentation of temporary modifications reviewed were found to be thorough and in accordance with administrative procedures.

8.0 Engineering Design Change Request (EDCR) Review

Engineering Design Change Request packages reviewed are listed in Attachment B.

8.1 MOV Limit Switch and Reach Rod Modification (EDCR No. 89-802)

Modification package EDCR 89-802 provides documentation for making design changes to the Residual Heat Removal (RHR) heat exchangers component cooling outlet valves PCC-M-43 and SCC-M-165, and the RHR suction valves from the containment sump CS-M-91 and CS-M-92. This modification was partially completed for valves PCC-M-43 and SCC-M-165 in September 1989. Modification of valves CS-M-91 and 92 is scheduled for completion during the next refueling outage. The deficiency of these valves was documented during an NRC Safety System Functional Inspection (SSFI) in Inspection Report No. 50-309/89-80. The three changes made to valves PCC-M-43 and SCC-M-165 to satisfy commitments made by the licensee following the SSFI are the following:

(1) Incorporate a torque bypass feature for the first 10% travel in the open direction off the main seat, and a 10% bypass in the closed direction off the back seat.

(2) Provide a close on limit to replace the present close on torque feature.

(3) Install a mechanical support with reach rod stem guide bearings.

The purpose of the changes was to assure that these valves will open when heavily seated in the main seat. This was accomplished by bypassing the torque switch during the first 10% stem travel and providing additional support to the valve stem reach rod to prevent binding.

The design input and installation instructions for this modification were found to be adequate. The requirements of the administrative procedures controlling the design process were complete. However, the lack of a complete Component Cooling Water (CCW) system design basis document was noted as a deficiency in the design process. The design engineers were unable to use the design basis document in developing functional testing and safety reviews for this modification. The licensee stated that formulation of the design basis for the CCW system has not been developed, but will be completed. The absence of a complete design basis for the system was one of the findings of the SSFI.

A second deficiency identified in this modification package was in the area of functional testing. To assure functioning of these valves during accident conditions, calculations or testing must be performed to to verify that these valves will open under design basis conditions. This would require opening the RHR heat exchanger CCW outlet valves with accident CCW flow through the heat exchanger. In lieu of a full functional test, calculations could be performed to correlate test results to full accident flow conditions. The functional test performed consisted of several tests to verify proper mechanical and electrical design and function of these valves. A deficiency in the stroke test performed on these valves was noted. Both valves were tested in accordance with procedures 3.17.8.2, "IST/IST Valve Test for Discrepancy Reports or Repair Orders" and 5.18.3, "Limitorque Valve Operator Operation Test." These tests satisfy the ASME B&PV Code Section XI surveillance test requirements. However, these tests do not require the valves to be stroked under design flow conditions and are not adequate to verify valve function following design changes. The licensee did not perform a functional test, using design basis flow, following modification to valves PCC-M-43 and SCC-M-165. The functional tests were performed by throttling the inlet component cooling valves to the RH heat exchanger to three turns open and then stroking valves PCC-M-43 and SCC-M-165. This reduced the component cooling water flow through valves PCC-M-43 and SCC-M-165 during the functional test. The pressure differential across the valve disk when the valve is raised off it's main seat is a function of the flow through the valve. The pressure differential across this valve at the 10% open position, when the torque switch is not bypassed, was less than during design accident conditions. The reduction in differential pressure across the valve disk will reduce the torque required to open the valve. Therefore, this test did not adequately verify valve operability for it's design condition. Inadequate testing procedures for modifications is a violation of the Main Yankee Atomic Company's Technical Specification, Section 5.8.1. (50-309/89-20-01)

During a shutdown approximately one month following this modification, the licensee attempted to open valves PCC-M-43 and SCC-M-165, while placing the RHR system in shutdown cooling. At this time valve SCC-M-165 opened 10% and then tripped on over torque. The reason for this valve failure was attributed to increase in CCW flow (non-safety CCW loads are reduced during plant shutdown), and mechanical problems with the valve. Following the failure of SCC-M-165 to open, the inlet valve for CCW to the RHR heat exchanger was closed, valve SCC-M-165 was then successfully opened. The torque bypass limit switch setting for valve SCC-M-165 was reset to 15%, and the valve was successfully tested with shutdown flow conditions. The flow during shutdown cooling is less than that expected during design basis accidents since some non-safety CCW loads are not isolated when in the shutdown cooling configuration. Valve PCC-M-43 performed satisfactorily when opened with shutdown flow conditions. The licensee has committed to verify that these valves will function during a design basis accident condition by writing a justification for continuing operation and developing and conducting a functional test of PCC-M-43 and SCC-M-165 during the next shutdown of sufficient duration.

8.2 Steam Generator Wide Range Pressure Instrumentation Upgrade (EDCR Number 88-51)

Modification package EDCR 88-51 provides documentation for upgrading the existing steam generator pressure indication in accordance with Regulatory Guide (RG) 1.97 requirement for Type A, Category 1 instruments.

The modification package review assessed the licensee's conformance to requirements specified in the NRC Order, dated June 12, 1984, the commitments made per Generic Letter 82-33 and Supplement 1 to NUREG-0737. The licensee's letter, dated February 28, 1985, identified the steam generator pressure as a Type A variable, Category 1 to accomplish manually controlled action for post accident monitoring. The letter also stated that it met the Regulatory Guide 1.97, Revision 3, guideline with the exception of the instrument range. The NRC previously reviewed this deviation and recommended that the range needs be updated to meet the R.G. 1.97, Rev. 3 criteria.

For Category 1 instrumentation, Regulatory Guide 1.97, Revision 3, Table 1, "Design and Qualification Criteria", Section 2 states that, "No single failure ..... should prevent the operators from being presented the information necessary for them to determine the safety status of the plant ..... This may be accomplished by providing additional independent channels of information of the same variable ......" Section 6 states that "Continuous real time display should be provided. The indications may be dial, digital display, CRT or strip chart recorder. Recording of instrumentation readout information should be provided for at least one redundant channel ..." and Section 8 states that, "Types A, B and C instruments designated as Categories 1 and 2 should be specifically identified with a common designation on the control panels so that the operators can easily discern that they are intended for use under accident condition."

During the review and field verification of modification package EDCR 88-51, the inspector noted that the steam generator pressure (wide range) did not have redundant channel recording devices or equipment to meet the RG 1.97, Revision 3 criteria. This is a deviation from the original licensee commitment to conform to Regulatory Guide 1.97, Revision 3. (50-309/89-20-02)

9.0 Exit Meeting

At the conclusion of the site inspection, on October 27, 1989, an exit interview was conducted with the licensee's senior site representatives (denoted in Section 1) to discuss the results and conclusions of this inspection.

At no time during this inspection was written material provided to the licensee by the inspector. Based on the NRC Region I review of this report and discussions held with licensee representatives during this inspection, it was determined that this report does not contain information subject to 10 CFR 2.790 restrictions.