U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-309/90-15

Docket No. 50-309

License No. DPR-36

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

Facility Name: Maine Yankee Atomic Power Station

Inspection At: Wiscasset, Maine

Inspection Conducted: July 9-13, 1990

Inspectors: R. Paolino, Senior Reactor Engineer — 8-21-90 date

G. Rangarao, Reactor Engineer — 8-21-90 date

L. Kay, Reactor Engineer — 8-21-90 date

Approved by: C. J. Anderson, Chief, Plant Systems Section
Engineering Branch, DRS — 8/27/90 date

Inspection Summary: Inspection of July 9-13, 1990 (Inspection Report No. 50-309/90-15)

Areas Inspected: Special, announced inspection to review the licensee's implementation of the post accident monitoring instrumentation in accordance with Regulatory Guide (RG) 1.97, Revision 3.

Results: Based upon the results of review conducted, the inspectors determined that the licensee had adequately implemented a program to meet the recommendations of RG. 1.97, Revision 3.

No violations were identified.

DETAILS

1.0 Persons Contacted

1.1 Maine Yankee Atomic Power Company

*A. J. Cayia, Manager - Operations
*J. M. DeBartolo, Process Control Engineer
J. Frothingham, Manager - Quality Programs
*T. Gifford, Project Engineer Section Head
*R. Hayward, QA Supervisor
J. He bert, Manager - Plant Engineering
*D. K p, Lead I & C Engineer
*S. Nichols, Licensing Section Head
*R. Prouty, Assistant Plant Manager/Maintenance Manager

1.2 Yankee Atomic Electric Coorporation

E. Gingham, Electrical Engineer
J. Bonner, Electrical Engineer
R. Jones, Electrical Engineer

1.3 State of Maine

P. Dostic, State Nuclear Safety Inspector

1.4 U.S. Nuclear Regulatory Commission

C. S. Marschall, Senior Resident Inspector

*Denotes personnel present at exit meeting of July 13, 1990.

2.0 Introduction

2.1 Background

The purpose of this inspection was to verify the Licensee's implementation systems for assessing plant conditions during and following the course of an accident based upon the criteria specified in Regulatory Guide (RG) 1.97, Revision 3. The instrumentation systems were also inspected to determine if they were installed in accordance with Generic Letter No. 82-33, "Requirements for Emergency Response Capabilities" (Supplement 1 to NUREG-0737). This letter, issued on December 17, 1982, specifies those requirements regarding emergency response capabilities that have been approved by the NRC for implementation. The supplement also discusses the application of RG 1.97 to the emergency response facilities. This includes the control room (CR), the technical support center (TSC) and the emergency response facility (EOF) at nuclear power facilities. Regulatory Guide 1.97 identifies the plant variables to be measured and the instrumentation criteria for ensuring acceptable emergency response capabilities during and following the course of an accident.

Regulatory Guide 1.97 divides the Post-Accident instrumentation into three (3) categories and five (5) types. The 3 categories are noted as 1, 2, and 3. Category 1 has the most stringent requirements, whereas Category 3 the least stringent. The 5 types of instrumentation identified in the Regulatory Guide are types A, B, C, D, and E. Type A variables are plant specific and classified by the licensee; type B variables provide information to indicate that the plant safety functions are being accomplished; type C variables provide information on the breach of barriers for fission product release; type D variables indicate the operation of individual safety systems; and type E are those that indicate and determine the magnitude of the release of radioactive materials. Each variable type can be any category, except for type A which can only be category 1.

2.2 Correspondence

Maine Yankee Atomic Power Company, the licensee for the Maine Yankee Atomic Power Station provided a response to Section 6.2 of the generic letter 82-33 on February 28, 1985. This submittal addresses the recommendations of Regulatory Guide 1.97 revision 3. Additional information was provided on June 17, 1986, September 5, 1986, April 8, 1988 and April 29, 1988 describing the licensee's position on post-accident monitoring instrumentation. Specific references used to assess the licensee's response to Regulatory Guide 1.97 revision 3 include:

- Regulatory Guide 1.97 revision 3, "Instrumentation for Light Water-Cooled Nuclear Power Plants to assess Plant and Environs Conditions During and Following an Accident."
- Safety Evaluation Report - Emergency Response Capability; Conformance to Regulatory Guide 1.97 Revision 3.
- Maine Yankee Final Safety Analysis Report.
- Applicable Licensee Procedures and Electrical/Instrumentation Reference Drawings.

3.0 Scope

The scope of the NRC inspection included: Identification of measured variables; method for measuring the parameter of interest (direct or indirect); display and recording methods used redundancy of power supplies; independence and physical/electrical separation of electrical circuits; range and overlapping features of multiple instrument indicators; equipment qualification (environmental and seismic); equipment identification for RG 1.97 Revision 3 instruments; service, test and surveillance frequency.

4.0 Inspection Details

The inspectors held discussions with various members of the licensee's staff, reviewed drawings and procedures and selected variables for physical inspection. To assess the licensee's implementation of RG 1.97, walkdowns were performed for selected sensing instruments and power distribution equipment at various locations of the auxiliary building and for display instruments in the control room. The instrument variables which were reviewed included reactor coolant systems (RCS) pressure , RCS temperatures (hot leg and cold leg), steam generator level (wide range and narrow range) pressurizer level, containment pressure, steam generator pressure and containment sump level.

All of the variables were listed as type A, category 1. For each variable, the characteristics examined by the inspectors included physical location of instrument components, function, physical/electrical separation, power source, environmental and seismic qualification status, type and identification of display/recording instruments, ranges and calibration.

An evaluation of applicable documents revealed that the instruments located in a harsh environment were qualified for that environment. Maine Yankee does not have a Q list, however, seismic qualifications of components are controlled by:

- Maine Yankee's Guidelines for Seismic Systems, Structures and Components. (Page 14 specifies that all class 1E components are seismic.)

- The Maine Yankee Instrument List identifies which instruments are class 1E/seismic.

- Seismic qualification of new components is addressed as part of EDCR development procedure and FSAR Section 2.5.5.

Maine Yankee conducts instrumentation testing and calibration in accordance with the Quality Assurance Program and the Technical Specifications. RG 1.97 class 1E instrumentation is calibrated and tested at each refueling outage. Each system is tested for its operability before each startup and following refueling outage. Additional monthly surveillances are performed on equipment as required by the Technical Specification. Set point adjustments are controlled by key-locked switches. Module calibrations are controlled by administrative procedures that require access by trained authorized technicians.

Instrument loops have been designed to facilitate the identification of a malfunction of module or channel. Annunciators are provided to indicate a loss of power for each vital bus. To meet the single failure criteria of RG 1.97 the instruments are powered from different buses with an isolator provided for the channel that is not on the same bus. Maine Yankee does not separate instrumentation trains in accordance with the guidance provided by RG 1.97. The separation criteria employed by Main Yankee is as described in Section 8.3.7 of the FSAR. Indicator/Recorder scales and instrument ranges were found to be generally in accordance with the RG 1.97 revision 3 guidelines. Where exceptions were made Maine Yankee provided supporting documentation to justify the deviation. These deviations are addressed in the NRC Safety Evaluation Report (SER), Supplement 4, Appendix L and found to be acceptable in meeting the guidelines of the RG 1.97 revision 3.

Display instrumentation in the control room is not specifically identified as RG 1.97 instrumentation. This item, in addition to two other deviations, was identified in a previous NRC inspection (Item no. 89-20-002, Report No. 309/89-20) conducted in October 23-27, 1989. The licensee responded (MN-90-11) to the deviations in a letter to the NRC dated January 22, 1990. In its response, the licensee indicated that equipment identification is done in accordance with Main Yankee's Detailed Control Room Design Review (DCRDR) Program. However, the DCRDR Program changes are not complete and a method of identification has not been chosen yet.

For the short term the licensee is providing colored labels to identify the RG 1.97 instrumentation. The labels were applied to RG 1.97 instrumentation prior to the completion of this inspection. For the long term fix the licensee plans to have all of DCRDR changes completed by the end of the refueling outage following the end of core cycle 13 which is approximately June 1993. Instrument identification will be addressed concurrent with completion of the DCRDR changes. This item is being tracked by Maine Yankee's Commitment Management System (CMS control No. 14-14-06) and scheduled for completion by June 30, 1993.

5.0 Exit Meeting

The inspectors met with licensee representatives denoted in Details, paragraph 1 of the report at the conclusion of the inspection on July 13, 1990. At that time, the scope of the inspection results were summarized. At no time during the inspection was written material given to the licensee.