U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/90-12

Docket No. 50-423

License No. NPF-49

Licensee: Northeast Nuclear Energy Company
P.O. Box 270
Hartford, CT 06141-0270

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection at: Waterford, Connecticut

Inspection Conducted: July 23-27, 1990

Inspectors: R. Paolino 8-7-90
R. Paolino, Senior Reactor Engineer, Plant Systems Section, EB, DRS — date

A. Della Greca 8/9/90
A. Della Greca, Senior Reactor Engineer, Plant Systems Section, EB, DRS — date

L. Kay 8/9/90
L. Kay, Reactor Engineer, Plant Systems Section, EB, DRS — date

Approved by: C J Anderson 8/13/90
C. J. Anderson, Chief, Plant Systems Section, Engineering Branch, DRS — date

Inspection Summary: Inspection of July 23-27, 1990 (Inspection Report No. 50-423/90-12)

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 2.

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 2.

No violations were identified.

DETAILS

1.0 Persons Contacted

1.1 Northeast Nuclear Energy Company

*	P. A. Blasioli	Supervisor Nuclear Licensing
	S. Caligiuri	Electrical Specialist
*	M. A. Ciccone	Senior Engineer
*	C. Clement	Director, MP3
	R. G Joshi	Senior Licensing Engineer
	F. L kascek	Mechanical Engineer
*	S. M. Oates	Nuclear Licensing
*	G. Olsen	Senior Engineer
*	R. Peterson	Senior Engineer
	R. V. Richter	Engineering Supervisor
*	M. F. Samek	Supervisor I&C Engineering
*	T. A. Shaffer	Manager I&C Engineering
	S. Stricker	Senior Electrical Engineer
	S. Wainio	Senior Engineer
	R. Young	Engineering Supervisor
*	K. Zita	Engineering Technician

* Denotes personnel present at the exit meeting of July 27, 1990.

2.0 Introduction

2.1 Background

The purpose of this inspection was to verify the Licensee's implementation of instrumentation 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 2. 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

The licensee's response to the NRC Generic Letter 82-33 for Millstone Nuclear Power Station, Unit 3, was provided in a submittal dated April 15, 1983. This response referred to an earlier submittal, dated February 2, 1983, which describes the licensee's position on post accident monitoring instrumentation. Additional information was provided on December 16, 1983 and January 13, 1984. At that time the licensee identified all variables required for post accident monitoring and whether or not the instrumentation complied with the recommendations of Regulatory Guide 1.97.

2.3 References

The specific references used to assess the licensee's response to Regulatory Guide 1.97 are as identified below:

- Regulatory Guide 1.97, Revision 2, "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 2.
- Millstone, Unit 3, Final Safety Analysis Report, Section 7.5.
- Applicable Licensee Procedures and 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 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 system (RCS) pressure (wide range and extended range), RCS temperature (hot leg and cold leg), steam generator level (wide range and narrow range), pressurizer level, containment pressure (narrow range and wide range), auxiliary feedwater flow, core exit temperature, containment hydrogen concentration, and neutron flux. Except for neutron flux which was identified as type B, all of the variables were listed in the FSAR as type A. For each variable, the characteristics examined by the inspectors included physical location of instrument components, function, physical and electrical separation, power sources, environmental and seismic qualification status, type and identification of display instruments, ranges and calibration.

An evaluation of applicable documents revealed that the instruments located in a harsh environment were qualified for that environment and were included in the EQ master list. No list existed for safety related instrumentation located in a mild environment and no list existed for seismic Category I equipment. However, all but four of the instruments within the scope of review were found in the plant's computerized master equipment data base. For each component the list clearly specified that the device was environmentally and seismically qualified. The Quality assurance procurement of these instruments was also reviewed. For the four devices (two indicators and two recorder) which had been inadvertently left off the list, the licensee provided procurement and qualification documents to adequately show that they had been procured and qualified to Class 1E requirements.

Instrument loops were found to be in calibration and calibrated within specified periods. Display instruments were found to be uniquely marked with purple and yellow dots, according to the electrical division from which they derived their power. Scales and instrument ranges were found to be in accordance with the recommendations of Regulatory Guide 1.97 and adequate overlapping of scales existed where more than one instrument was used to cover the range. Recording was provided for at least one of the redundant instrument loops and, in those cases where the recorder was also used as indicator, a Class 1E recorder was provided. Except as described below, adequate redundancy of post accident instrumentation was provided.

Power supplies were found to be located in different areas of the plant segregated by fire walls. Separation between wires from redundant divisions was generally observed by means of barriers and distance in accordance with the recommendations of RG 1.75. In one isolated case, within a local panel for the hydrogen concentration monitoring system, two redundant cables were observed by the NRC to be closer than the minimum six inch recommended distance. The licensee initiated immediate corrective action by issuing a work request to [illegible] the cables. This work was completed prior to the end of the inspection. This violation is not being cited because the criteria specified in 10 CFR 2, Appendix C, V.A of the Enforcement Policy were satisfied. Within the control room, where redundant display instruments were adjacent to one another, metal enclosures around the instruments were used to achieve the required separation. In all cases, redundant instrument used redundant power supplies with adequate isolation between Class 1E and non Class 1E components.

4.1 Steam Generator Level/Auxiliary Feedwater Flow

The licensee identified the steam generator level (wide and narrow range) and the auxiliary feedwater flow as type A variables. For type A variables RG 1.97 recommends that redundant instruments be provided. However, an evaluation of the licensee's documents revealed that, in the case of these variables, redundant instruments had been provided only for the narrow range level channels. Subsequent discussions with the licensee indicated that they considered steam generator level and auxiliary feedwater flow to be redundant to each other and the redundancy to be achieved by means of diverse instrumentation. Although it was also determined that the loss of one division of power supply would result in the loss of indication of both flow and wide range level for two of the four steam generators, the design was determined to be acceptable and in accordance with the intent of section 1.3.1.b of Regulatory Guide 1.97. Acceptability was based upon the fact that only one steam generator is required for the safe shutdown of the plant and that narrow range level instruments provide adequate backup information.

4.2 RCS Hot Leg Water Temperature

The licensee determined the RCS hot leg water temperature to be a type A variable. For type A variables RG 1.97 recommends that redundant instrumentation be provided. However, a review of the plant drawings revealed that no redundant instruments had been provided. In response to the inspector's observation, the licensee indicated that they considered the core exit temperature instruments to provide adequate redundant backup information for the subject variable. Although the deviation was not specifically identified in the licensee's submittal to the NRC, the inspector determined the design to be acceptable and in conformance with the intent of RG 1.97. An interview of a plant operator showed that they are aware that core exit temperature instruments provide adequate redundant indication for the steam generator hot leg temperature. The licensee, nonetheless, committed to revise the FSAR and related emergency procedures to clearly identify the availability of the alternate instrumentation to the RCS hot leg water temperature.

4.3 Isolation Devices

Where a Category 1 signal is used as input to a non-Category 1 system, RG 1.97 specifies the use of isolation devices which are fully qualified for use in Category 1 circuits. The inspector examined the circuits involved and determined that the isolation as well as the separation criteria had been adequately implemented by the licensee. In particular, the inspection revealed that analog and digital signals to recorders, annunciators and to the plant computer, as applicable, were primarily transmitted through Class 1E Westinghouse, series 7300, isolator cards. Westinghouse protection system demultiplexers were also found to be used in one application. The Westinghouse WCAP test reports provided by the licensee adequately demonstrated acceptability of the devices as effective isolators between the safety related and the non-safety related portions of the circuits.

5.0 Exit Meeting

The inspectors met with the licensee representatives denoted in Section 1.0 of the report at the conclusion of the inspection, on July 27, 1990. At that time, the scope of the inspection and the inspection results were summarized. At no time,during the inspection, was written material given to the licensee.