U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-443/90-20

Docket No. 50-443

License No. NPF-56

Licensee: Public Service Company of New Hampshire
Post Office Box 300
Seabrook, NH 03874

Facility Name: Seabrook Nuclear Power Station

Inspection at: Seabrook, New Hampshire

Inspection Conducted: September 24-28, 1990

Inspectors: G. Rangarao For 10/24/90
A. Della Greca, Senior Reactor Engineer, Plant Systems Section, EB — date

G. Rangarao 10/24/90
G. Rangarao, Reactor Engineer, Plant Systems Section, EB, — date

Approved by: C. J. Anderson 10/24/90
C. J. Anderson, Chief, Plant Systems Section, Engineering Branch, DRS — date

Inspection Summary: Inspection of September 24-28, 1990 (Inspection Report No. 50-443/90-20)

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 Northeast Nuclear Energy Company

* R. Bergeron	Engineering program Manager
B. Beuchel	I&C Engineering Supervisor
B. Brown	Mechanical Engineering Supervisor
* J. P. Cady, Jr.	ISEG Supervisor
T. Carr	Plant Technical Support Engineer
R. M. Cooney	Maintenance Manager
D. Covill	NQG Surveillance Supervisor
* W. A. Di Profio	Assistant Station Manager
* B. L. Drawbridge	Executive Director, Nuclear Production
R. Godbout	I&C Working Foreman
* T. Harper	Director licensing Services
J. Kotkowski	Electrical Engineering Supervisor
* R. L. Krohn	NRC Coordinator
* J. Malone	Operations Administration Supervisor
* D. Moody	Station Manager
T. Murphy	I&C Department Manager
* J. M. Peschel	Regulatory Compliance Manager
J Peterson	Assistant Operations Manager
* N. A. Pillsbury	Director Quality Programs
* E. J. Sovetsky	Technical Projects Supervisor
* M. F. Toole	I&C Supervisor
* P. J. Tutinas	Engineering Programs Supervisor
* J. M. Vargas	Manager of Engineering
* C. J. Vincent	Q.C. Department Supervisor
* J. Warnock	Nuclear Quality Manager

1.2 Yankee Atomic Energy Company

* W. G. Alcusky	I&C Engineer
* W. H. Reed	Lead I&C Engineer

1.3 Consultants

* R. Faix	Engineer, Westinghouse
* R. P. Neustadter	I&C Engineer, UE&C

1.4 U.S. Nuclear Regulatory Commission

N. Dudley	Senior Resident Inspector
* R. Fuhrmeister	Resident Inspector

* Denotes personnel present at the exit meeting of September 28, 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 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 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 3, "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."
- Supplemental Safety Evaluation Report - Conformance to Regulatory Guide 1.97.
- Seabrook Final Safety Analysis Report, Section 7.5 and Appendix 7A, Deviations of AMI Variables from Reg. Guide 1.97/FSAR Subsection 7.5.4.4 Design Criteria.
- 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 reviewed included: (1) all of the type A variables - reactor coolant system (RCS) pressure, core exit temperature, degree of subcooling, steam generator pressure, steam generator level (wide range and narrow range), pressurizer level, refueling water storage tank level, and containment hydrogen concentration; (2) three type B variables - RCS temperature (hot leg and cold leg) and containment pressure; and (3) four type D variables - accumulator tank level, accumulator tank pressure, pressurizer relief tank level and containment sump water temperature. Except for those which are identified as type D, all of the above variables are listed in the FSAR as being Category I.

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 seismically and environmentally qualified for their respective mounting locations and were included in the EQ master list. Similarly, all safety related instruments located in a mild environment were identified in the Class 1E list and were seismically qualified either through the licensee's environmental qualification program or as part of the panels and racks with which they are associated. The Quality Assurance procurement of these instruments was also reviewed. Except as described in section 4.1, below, all devices were found to be covered by the licensee's QA program.

Instrument loops were found to be calibrated within specified periods and to be currently in calibration. A specific concern regarding the calibration status of a steam generator level transmitter is discussed in details in section 4.2, below.

Display instruments required for Post-Accident monitoring were found to be uniquely marked with orange tags and easily distinguishable from other panel instrumentation, in accordance with the recommendations of Regulatory Guide 1.97. Where more than one instrument was used to cover the range of a variable, adequate overlapping of scales existed. Trending recorders were provided for each type A variable and redundancy of Post-Accident instrumentation was adequate. Sections 4.3 and 4.6 discuss the licensee's resolution of potentially confusing instrument scales in the control room.

Except as described in section 4.4, below, separation between instruments, cables and wires from redundant divisions was generally in accordance with the recommendations of RG 1.75. 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 instruments used redundant power supplies with adequate isolation between Class 1E and non-Class 1E components.

The inspectors also reviewed the licensee's actions to satisfy the commitments made to the NRC regarding accumulator tank level and pressure, containment sump water temperature, and pressurizer relief tank temperature. These commitments are discussed in the supplemental safety evaluation report. The results of this review are summarized in Section 4.5, below.

4.1 Qualification of Display Instruments

Footnotes 1 and 2 on sheet 37 of FSAR Table 7.5-1 appeared to imply inadequate qualification for two indicators (RC-TI-433A & B) and eight recorders (FW-LR-519, 529, 539, and 549; RC-PR-405; RC-TR-433A & B; CBS-LR-2385). Discussion with the licensee revealed that these instruments, although listed under Type A and B variables in the FSAR Table 7.5-1 Sheet 1, 3 and 5, were not used for Post-Accident monitoring and they are not specified in RG 1.97. However, they provide additional back up to RG 1.97 instrumentation. Therefore, these instruments do not require qualification as per RG 1.97 except for seismic mountings. Evaluation of the design showed that these instruments were treated similar to the safety related equipment for panel mounting, signal isolation and wiring. The licensee agreed that the notes needed clarification. This will be addressed in the next FSAR update. The inspectors had no further questions.

4.2 Instrument Calibration

While reviewing the licensee's calibration records, the NRC inspector observed that a Rosemount transmitter, FW-LT-548, which is used for measurement of a steam generator level, had been found out of tolerance, on several previous occasions. The inspector discussed this observation with I&C Maintenance and determined that the original transmitter had been recently changed to a Rosemount transmitter and that some of the earlier notations pertained to the old transmitter. The inspector was also informed that, as a result of Information Notice 90-01, the transmitter in question was being trended on a weekly basis, together with 15 more transmitters associated with the measurement of the steam generators level. In support of the above, the licensee furnished a report of the data recorded on

August 30, 1990. A summary review of the data showed that the loop in question displayed an output which was consistently lower than that of the other three transmitters associated with the same steam generator and that the error appeared to be on the order of 0.208V, on average, i.e., 4.472V vs. 4.950V. In contrast, the tolerance specified for that transmitter, in the applicable procedure, was +/- 0.020V. The licensee explained that the measurements were taken during plant operation and at the computer input. Therefore, the measurements would be affected by the dynamics of the process and by the added error from the interposing card (+/- 0.050V). The inspector agreed that part of the error could have been the result of the specific conditions identified by the licensee. However, the consistency of the measurements, pointed to a different cause. The inspector pointed out that other steam generator level instrument loops appeared to be similarly out of tolerance, e.g. 4.887V average for FW-LT-538 vs. 5.106V for FW-LT-553.

Following the inspection, on October 9, 1990, the licensee reported that they had discussed the finding with Westinghouse and that they had concluded that the apparent error was process related. The licensee suggested that engineering had been aware of the trending results, but had concluded that the error would not affect the overall accuracy of the loop. Since all sources of error need be addressed in loop accuracy and setpoint calculations, regardless of their source, this issue is unresolved pending the NRC's review of the licensee's corrective actions. (50-443/90-20-01)

4.3 Hydrogen Concentration

Table 7.5-1 of the FSAR indicates that the actual range provided for containment hydrogen concentration is 0-20%. However, during the plant walkdown, the inspectors observed that the control room instruments were equipped with a 0-10% scale. The inspectors also determined that corresponding meters on a control room back panel were equipped with a switch to change the scale from 0-10% to 0-20%. Since the potential for erroneous information to the operator existed, the inspector asked the licensee to identify the effect of the switch on the control room instruments. The licensee was not able to provide the requested information by the end of the inspection. However, in a telephone conversation, on October 9, 1990, the licensee reported that their investigation had concluded that the switches associated with the back panel meters also affected the ones in the control room. Therefore, if the switch were left in the 0-20% position, the operator would receive a false, non-conservative indication of containment hydrogen concentration. The licensee also reported that they already had initiated necessary actions to disable the switches. However, the licensee did not consider the condition a major concern since they had administrative controls to ensure correct information to the operator. In support of this position, the licensee provided portions of a calibration procedure (IX1614.912D, Rev. 4) which requires verification that the dual range selector switch is in the 0-10% range. The licensee provided parts of operations procedure OS1023.71, Rev. 5, which requires placing the switch in the 0-10% range prior to placing the analyzer in standby mode. Neither procedure could prevent the inadvertent positioning of the switch in the higher range. However, since adequate corrective actions were already initiated by the licensee, the issue is closed.

4.4 Electrical Isolation and Separation

Where a Category I signal is used as input to a non-Category I 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 transmitted through Class 1E Westinghouse, series 7300, isolator cards. The Westinghouse WCAP test reports adequately demonstrate acceptability of the devices as effective isolators between the safety related and the non-safety related portions of the circuits.

While performing the plant walkdown, the NRC inspectors observed that cables belonging to different channels and different trains could be found within the same instrument rack. The instrument racks contain the reactor protection system logic cards. Since, in two cases, two cables associated with different channels and trains passed and nearly touched each others, the inspectors questioned the acceptability of the installation. In response, the licensee provided a report of tests performed by Westinghouse, WCAP-8892-A. This report, previously accepted by the NRC, indicates that no damage occurred to cables and cards when subjected to maximum credible fault currents and voltages. The inspectors had no further questions.

4.5 Previously Identified Deviations

During the safety review of Seabrook's R.G. 1.97 submittal, the NRC found three deviations taken by the licensee regarding three type D variables to be unacceptable. The variables questioned by the NRC were: (1) accumulator tank pressure or level; (2) containment sump water temperature; and (3) pressurizer relief tank temperature. To resolve the issues involved the licensee committed to provide environmentally qualified instrument loops for variables (1) and (2) and to increase the range of monitoring for variable (3).

During the current review, the inspector evaluated the status of the commitments made by the licensee and determined that design change request (DCR) Nos. 87-316, 87-0318, and 87-384 had been prepared and issued to: (1) replace the existing accumulator tanks pressure transmitters with qualified ones; (2) install qualified thermocouples in lieu of the existing ones, at the inlet side of the containment building spray heat exchangers; and (3) extend the monitoring range of the pressurizer relief tank temperature by replacing the meter scale and recalibrating the instrument loop. Review of the above DCR package confirmed resolution of the issues. Therefore, the items are closed.

4.6 Recorder Chart Paper

While reviewing the records of calibration for steam generator level instrument loop L-502, the inspector observed a note written by a technician, on January 30, 1990. According to the note, another I&C technician had previously recommended that the chart paper for recorder 1-FW-XR-502 be changed to a new design (CID No. 55527507). The reason for the request was to reduce additional extrapolation for reading (each small division equaled 1.538461538%). The note also complained that "the problem had not been resolved" by the date of the note.

Apparently, the problem arose from the fact that the same recorder is used for both level (0-100%) and pressure (0-1300 psig). The paper used was calibrated for pressure with 13 major and 5 minor division. Therefore, the chart could be easily calibrated for pressure, but not for level. The inspector showed the note to the licensee and asked the licensee to determine if the situation had been corrected. During the field inspection, the NRC noted that the chart paper had been changed to one with a 0-100% scale. However, the inspector did not consider the change to be adequate since the new scale, while resolving the level situation, now required a calculation to determine the steam generator's pressure. In addition, the pressure portion of the scale deviated from the FSAR table, which states 0-1300 psig. Further discussions with the licensee revealed that the correct chart paper had been purchased, but the records had not been revised to identify its storage location.

By the end of the inspection, the correct paper was loaded in the affected recorders and the record had been revised. Therefore, the item is closed.

5.0 Unresolved Item

Unresolved items are matters about which additional information is necessary in order to determine whether they are acceptable or the constitute a violation. One unresolved item is discussed in details under Section 4.2.

6.0 Exit Meeting

The inspectors met with the licensee representatives denoted in Section 1.0 of the report at the conclusion of the inspection, on September 28, 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.

OUTSTANDING ITEMS FILE SINGLE DOCKET ENTRY FORM

REPORT HOURS

1. Operations ____
2. Rad-Con ____
3. Maintenance ____
4. Surveillance ____
5. Emerg. Prep. ____
6. Sec/Safegrds. ____
7. Outages ____
8. Training ____
9. Licensing ____
10. QA ____
11. Other ✓
12. Fire Protection/ Housekeeping ____

Docket No. 443

Originator A. Della Greca

Reviewing Supervisor C/9

Item Number	Type	SALP Area	Area	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt/	Date O/M/Clsd (MM-DD-YY)
90-20-01	UNR	ENG'G	SMG	01-31-91	- -	09-28-90

Originator/Modifier	Resp Sec
DELLAGRECA	E/E

Descriptive Title

LOOP ACCURACY/SETPOINT CALCULATIONS FOR STEAM GENERATORS' LEVEL

Item Number	Type	SALP Area	Area	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt/	Date O/M/Clsd (MM-DD-YY)
- -				- -	- -	- -

Originator/Modifier	Resp Sec

Descriptive Title

Item Number	Type	SALP Area	Area	Action Due Date (MM-DD-YY)	Updt/Clsout Rpt/	Date O/M/Clsd (MM-DD-YY)
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Originator/Modifier	Resp Sec

Descriptive Title

Region I Form 6
(April 1987)