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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER STATION

DOCKET NO. 50-309

SAFETY PARAMETER DISPLAY SYSTEM

1.0 INTRODUCTION

All holders of operating licenses issued by the Nuclear Regulatory Commission (licensees) and applicants for an operating license must provide a Safety Parameter Display System (SPDS) in the control rooms of their plant. The Commission-approved requirements for the SPDS are defined in Supplement 1 to NUREG-0737 (Reference 1).

The purpose of the SPDS is to provide a concise display of critical plant variables to control room operators to aid them in rapidly and reliably determining the safety status of the plant. NUREG-0737, Supplement 1, requires licensees and applicants to prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Licensees and applicants shall also prepare an implementation plan for the SPDS, which contains schedules for design, development, installation, and full operation of the SPDS as well as a design verification and validation plan. The safety analysis and the implementation plan are to be submitted to the NRC for staff review. The results from the staff's review are to be published in a Safety Evaluation (SE).

Prompt implementation of the SPDS in operating reactors is a design goal of prime importance. The review of human factors design of the SPDS for operating reactors called for in NUREG-0737, Supplement 1, is designed to avoid delays resulting from the time required for NRC staff review. The NRC staff will not review operating reactor SPDS designs for compliance with the requirements of Supplement 1 of NUREG-0737 prior to implementation unless a pre-implementation review has been specifically requested by licensees. The licensee's safety analysis and SPDS implementation plan will be reviewed by the NRC staff only to determine if a serious safety question is posed or if the analysis is seriously inadequate. The NRC staff review to accomplish this will be directed at (1) confirming the adequacy of the parameters selected to be displayed to detect critical safety functions, (2) confirming that means are provided to assure that the data displayed are valid, (3) confirming that the licensee has committed to a human factors program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator, and (4) confirming that the SPDS will be suitably isolated from electrical and electronic interference with equipment and sensors that are used in safety systems. If, based on this review, the staff identifies serious safety questions or seriously inadequate analysis, the Director of IE or the Director of NRR may require or direct the licensee to cease implementation.

2.0 SUMMARY

The staff reviewed the Maine Yankee Atomic Power Company's SPDS Safety Analysis for Maine Yankee Atomic Power Station. Based on the results from the review, we conclude that no serious safety questions are posed by the proposed SPDS and therefore, implementation of the SPDS by the licensee may continue. However, the staff's review did identify several recommendations. The licensee has evaluated these recommendations and his actions concerning those recommendations will be reviewed during the next 10 CFR 50.59 inspection.

3.0 EVALUATION

3.1 Background

By letter dated May 31, 1985, the Maine Yankee Atomic Power Company submitted to the NRC a Safety Analysis (Reference 2) on the SPDS. The staff reviewed the analysis and because of insufficient information was unable to complete the review. A request for additional information (Reference 3) was forwarded to the licensee and a plant site visit (June 12-13, 1986) was conducted to obtain the information needed to complete the review. The licensee's responses to the staff's information request (References 4 and 5), the information gathered during the plant site visit, and the information in the safety analysis serve as the basis of the safety evaluation that follows next.

3.2 Description

The licensee's SPDS is computer-based and is designed to assist plant personnel to evaluate plant status during normal and off-normal conditions. Within the control room, two color graphic display monitors serve as the SPDS display interface to control room personnel. Each monitor has a keyboard, which serves as an interface for humans to select the desired display format. Different display formats may be used on the two monitors at the same time.

The safety analysis states that the SPDS display formats are not normally displayed on the monitors. However, upon receipt of any alarm to the SPDS, the SPDS default display format appears automatically upon the monitor assigned to the plant shift supervisor. The default display format contains overall status data on the critical safety functions and on the plant. Also, a menu for access to all other SPDS display formats is presented. Items from the menu are selectable via the keyboard associated with the monitor.

3.3 Parameter Selection

Section 4.1f of Supplement 1 to NUREG-0737 states that:

> "The minimum information to be provided shall be sufficient to provide information to plant operators about:
>
> 1. Reactivity control;
> 2. Reactor core cooling and heat removal from the primary system;
> 3. Reactor coolant system integrity;
> 4. Radioactivity control; and
> 5. Containment conditions."

For review purposes, these five items have been designated as critical safety functions (CSF's).

The licensee's safety analysis states that the main role of the SPDS is to aid operators in determining the status of the CSF's identified within the emergency operating procedures (EOP's). The licensee's EOP's are derived from the Westinghouse Owners Group's emergency response guidelines (ERGs), the human factors task analysis, and USNRC RG 1.97 data. The licensee identifies six CSF's, with an order of priority as follows:

1. Subcriticality;
2. Core cooling;
3. Heat sink;
4. Integrity;
5. Containment; and
6. Inventory.

Our review of the licensee's CSF's noted that with the exception of Radiation Control, the CSF's are responsive to the requirements of NUREG-0737, Supplement 1. The staff queried the licensee's personnel on Radiation Control data within the SPDS during the plant site visit. The licensee's response described a display format within the containment CSF that contained radiation data. This display format is within the second level of detail of the display hierarchy and a copy of the display page is contained in Appendix A. Our review of the data within this display page concluded that the data was adequate to evaluate the radiation control CSF. We conclude that the design of the licensee's SPDS responds to all of the CSF's required by NUREG-0737, Supplement 1.

Our review of the licensee's Safety Analysis evaluated the parameters selected to evaluate each CSF. Table I describes the SPDS displays and the parameters within each display format as stated by the licensee's safety analysis. Our review of the parameters selected to evaluate each CSF concluded they were adequate, but noted several parameters important for assessing some of the CSF's were missing from the SPDS. The missing parameters consisted of a reactor water level, low pressure safety injection flow, and the status of containment isolation.

During the plant site visit, the staff identified the parameters missing from the SPDS and discussed the importance of these parameters in evaluating the Core Cooling Function, the Heat Sink Function, and the status of Containment Integrity. The licensee responded by restating a previous commitment to add the reactor water level to the SPDS upon installation of Inadequate Core Cooling Instrumentation (Reference 6). Furthermore, the licensee illustrated that low pressure safety injection flow and high pressure safety injection flow were displayed on a second level display format within the Inventory CSF and that status of containment isolation was displayed within the top level display format for the Containment CSF. A copy of these display pages is also presented in Appendix A. Our review of containment status noted that only the demand signal for isolation was present and we recommend that the actual measured status also be presented. In conclusion, the staff finds the parameters selected by the licensee for use in the SPDS acceptable, but we recommend that the measured status of containment isolation also be displayed.

3.4 Display Data Validation

The staff evaluated the licensee's design to determine that means are provided in the display system to assure that the data displayed are valid. The licensee's Safety Analysis addresses data integrity by evaluating the instrument loops that provide signals to the SPDS. Any instrument loop with a zero volt output is considered as a failed loop. Such failures will be indicated on the displays with the word "FAIL" in place of numerical values for the parameter.

Instrument loops may also fail to a non-zero output of volts. To detect these type of failures, a continuous comparison of the instrument loop's signal with other Class 1E instrument channels is done. For three instrument channels providing the same parameter, the comparison is designed to detect any large differences among the signals. The licensee states that acceptable differences will be defined for each parameter. During our plant site visit, the staff learned that the most important SPDS parameters were derived from Class 1E instruments, and with one exception, these parameters used a minimum of two Class 1E sensors as data sources. In the one exception, a containment high range radiation monitor, the licensee stated that a redundant monitor was scheduled to be added during the next refueling outage.

Non-Class 1E data is also used within the SPDS. During our plant site visit, the staff learned that other than a failure to zero test, no additional data validation is performed upon this data. Based on the information contained in the safety analysis and the information obtained at the plant site, the staff confirms that the licensee's design provides a means to assure that the data displayed are valid.

3.5 Human Factors Program

The staff evaluated the licensee's safety analysis and conducted a short assessment of the SPDS during the plant site visit. The object of this effort was to evaluate the licensee's design process for a commitment to a Human Factors Program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator.

SPDS display formats are not displayed continuously. Upon an SPDS alarm, the default SPDS display format is automatically presented on the monitor. The default display format contains the general status of each CSF and presents specific data on plant primary side equipment. As the SPDS alarms, most parameters within the SPDS have two or more setpoints. One setpoint, for normal operating conditions, is typically a pre-alarm setpoint to alert the operator to non-ordinary changes in the operating characteristics of the plant. A second setpoint is typically for post-trip situations; these setpoints are largely based upon the EOP's and Engineered Safety Features post-trip requirements. Post-trip setpoints are designed to alert the operator to conditions outside of EOP requirements or Technical Specification Limitations. This type of display automation relieves the operator from searching for the most critical data. The staff's review did not evaluate the setpoints nor their implementation into the display system.

Across the top of each SPDS display format are six CSF status indication boxes. A change of color or flashing of a particular box indicates that a parameter supporting that CSF has reached a setpoint and a possible challenge to that CSF exists. These visual cues are useful in communicating the safety status of the plant to the operator. Furthermore, dedicated labeled keys, located beneath the display screen, are colored coded and hierarchy structured to support the two levels of detail for each CSF. With this type of interface design, rapid access to data for a CSF in an alarmed state is easily achieved. Finally, most display formats within the library were uncluttered and easily read.

Our plant site visit did note two problem areas. The first problem noted was the data sample frequency for parameters monitored by the SPDS. The licensee stated that process parameters used within the SPDS were monitored every ten seconds. Furthermore, the SPDS program within the computer was executed once every ten seconds. Most SPDS systems reviewed by the staff use a data sample rate of one second. The staff stated that a ten second data sample period was large and could result in the loss of process data, specifically for such parameters as neutron flux and coolant pressure. The staff recommended that the licensee re-evaluate the data sample rate used within the SPDS and select and use a data sample rate based upon process dynamics and operator information needs. By phone conversation April 8, 1987, the licensee indicated that action had been taken concerning this recommendation.

The second problem the staff noted during its plant site visit was a lack of consistency among display formats. Several human engineering discrepancies, such as inconsistent use of units, labels, abbreviations, scales, numerical display of temperature to the nearest degree, tenth of degree, and hundredth of a degree, and hard to read red text on a black background were noted. The staff recommends that the licensee adopt a human factors standard/guideline for CRT displays, review the display formats, and correct the discrepancies identified from the review. In addition, the display standard/guideline used should be consistent with the guidelines used by the licensee in the Detailed Control Room Design Review.

Based on our review of the information contained in the Safety Analysis and our observations of the display system at the plant site, the staff confirms that the licensee has some elements of human factors within the design.

In the licensee's Safety Analysis, a description of the Design Verification and Validation Program used to develop the SPDS is presented. The safety analysis states that the SPDS design was independently reviewed and validated by the Nuclear Services Division of Yankee Atomic Electric Company. This effort also included reviews by the EOP Working Group, operations personnel, plant management, and the Human Factors Working Group. Other activities consisted of algorithm reviews, formal work-throughs of code, and unit testing of software modules.

The Safety Analysis also describes a Validation Program. Each display format is to be validated as being accurate in the methods used to present information. A series of event simulations on the Maine Yankee simulator and a series of event walk-throughs on the Main Control Board with the EOP's will test the effectiveness of each display format. During the staff's visit to the plant site, we did not evaluate the results from the Verification and Validation Program. Based on our review of the licensee's Verification and Validation Program as presented in the Safety Analysis, the staff concludes that it is adequate.

3.6 Electrical and Electronic Isolation

The SPDS must be isolated from equipment and sensors than are used in safety systems (NUREG-0737, Supplement 1).

In order to satisfy the NRC requirements concerning the SPDS, Maine Yankee Atomic Power Company submitted a Safety Analysis Report by letter dated May 31, 1985 (Ref. 2). The report provided a description and a safety analysis of the SPDS at the Maine Yankee Atomic Power Plant. This report did not address the requirement that the SPDS must be isolated from equipment and sensors that are used in safety systems to prevent electrical and electronic interference. A request for additional information, which included specific questions on these isolators, was sent to the licensee on February 11, 1985 (Reference 3). The requested information was received in letters dated April 18, 1986 (Reference 4), June 3, 1986 (Reference 5), July 1, 1986 (Reference 7) and July 16, 1986 (Reference. 8). Several telephone conferences were held with the licensee to clarify the information submitted on the Energy Incorporated's isolation amplifiers and to discuss the analysis used in arriving at the values of the maximum credible fault.

The Class 1E inputs to the SPDS are isolated from the non-Class 1E SPDS by analog isolation devices provided by Energy Incorporated (EI). These devices are EI's Model 1622. EI isolators use a Burr-Brown optical isolator for the Class 1E (input) to non-Class 1E (output) insolation. The isolators were subjected to a surge withstand capability test, a functional test, a hi-pot test, and a design basis fault test. The isolators successfully passed the surge withstand capability test, the functional test, and the hi-pot test. The design basis fault test applied the maximum credible fault (MCF) voltage/current to the output terminals of the isolator in the transverse mode.

The values of the MCF voltage/current used in the design basis fault test were 480VAC at 10 amps, 120VAC at 20 amps, and 140VDC at 10 amps. These MCF values bound the values of 120VAC at a potential current of 10 amps, which were analyzed for the Maine Yankee plant.

The pass/fail criteria invoked in the design basis fault test stated that, during and following the application of the MCF to the non-Class 1E output of the isolator, the isolation barrier is not breached and the MCF does not propagate to the Class 1E input.

The test data on the EI Model 1622 have been reviewed and accepted by the staff in Reference 9 and are applicable to the Maine Yankee plant.

In addition to the tests previously mentioned, the isolators were qualified to IEEE-381-1977, "Standard Criteria for Type Tests for Class 1E Modules Used in Nuclear Power Generating Stations." The isolators are located in a mild environment; therefore, they do not come under the requirements of 10 CFR 50.49.

Based on the staff's review of the licensee's submittals with respect to EI's analog isolation devices and on the prior review and acceptance of identical isolators at other plants, the staff concludes that these devices are qualified isolators and are acceptable for interfacing the SPDS with Class 1E systems. The staff also concludes that this equipment meets the Commission's requirements in NUREG-0737, Supplement 1.

4.0 CONCLUSIONS

The NRC staff reviewed Maine Yankee Atomic Power Company's Safety Analysis to confirm the adequacy of the variables selected to be displayed to monitor the critical safety functions, to confirm that means are provided to assure that the data displayed are valid, to confirm that the licensee has committed to a Human Factors Program, to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator, and to confirm that the SPDS is suitably isolated. Based on its review to date, the staff concludes that no serious safety questions are posed by the proposed SPDS, and therefore, implementation of the SPDS by the licensee may continue.

Based upon the results from our review, we conclude:

- the parameters selected for display and evaluation of the Critical Safety Functions are generally acceptable, however, we recommend that the measured status of containment isolation also be displayed;
- a means to assure that the displayed data are valid is provided in the design;
- A Human Factors Program was used in the design of the display;
- the Design Verification and Validation Program is adequate; and
- the SPDS is suitably isolated.

A human factors review of the display formats will be conducted during Cycle 10 refueling and corrections made as necessary. This item will be verified at the next 10 CFR 50.59 inspection.

The data sample rate within the SPDS has been re-evaluated. A data sample rate based upon process dynamics and operator's information needs should be identified and used. This item will be verified at the next 10 CFR 50.59 inspection.

The conclusion that the SPDS implementation may continue does not imply that the SPDS meets or will meet the requirements of Supplement 1 to NUREG-0737. Such confirmation can be made only after a post-implementation audit, and after the staff would evaluate the licensee's response to the above recommendations.

An appropriate implementation schedule will be developed by the Project Manager via discussions with the licensee. Licensees are required to inform the Commission, in writing, of any significant changes in the estimated completion schedule identified in the staff's safety evaluation and when the action has actually been completed.

Date:

Principal Contributor:
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5.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements, Requirements For Emergency Response Capability," U.S. NRC Report NUREG-0737, Supplement 1, January 1983.

2. Letter from G.D. Whittier, Maine Yankee Atomic Power Company, to J.R. Miller, NRC, Subject: Maine Yankee Safety Parameter Display System - Safety Evaluation, dated May 31, 1985.

3. Letter from A.C. Thadani, NRC, to J. Randazza, Maine Yankee Atomic Power Company, Subject: Request for Additional Information Concerning The SPDS For Maine Yankee, dated February 11, 1986.

4. Letter from G.D. Whittier, Maine Yankee Atomic Power Company, to A.C. Thadani, NRC, Subject: Additional Information on SPDS, dated April 18, 1986.

5. Letter from S.M. Stilwell, Maine Yankee Project, to: P. Sears, NRC, Subject: Energy Incorporated Isolation Devices, dated June 3, 1986.

6. Letter from G.D. Whittier, Maine Yankee Atomic Power Company, to E.J. Butcher, Jr., NRC, Subject: Answers to Questions Concerning Inadequate Core Cooling Instrumentation, dated September 6, 1985.

7. Letter from G.D. Whittier, Maine Yankee Atomic Power Company, to A.C. Thadani, NRC, Subject: SPDS Isolation Devices, dated July 1, 1986.

8. Letter from G.D. Whittier, Maine Yankee Atomic Power Company, to A.C. Thadani, NRC, Subject: SPDS Isolation Devices, dated July 16, 1986.

9. Memorandum from D.M. Crutchfield, NRC, to D.H. Jaffe, NRC, Subject: Safety Evaluation Input for the Millstone 2 Safety Parameter Display System, dated February 2, 1987.