ENCLOSURE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SAFETY PARAMETER DISPLAY SYSTEM

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

I. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 was issued by D. G. Eisenhut, Director, Division of Licensing, Office of Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses and holders of construction permits. This letter included additional clarification regarding Safety Parameter Display Systems (SPDS), relating to the requirements for emergency response capability. These requirements have been published as Supplement 1 to NUREG-0737, "TMI Action Plan Requirements."

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 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.

Prompt implementation of the SPDS in operating reactors is a design goal of prime importance. The staff's review of SPDS documentation 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.

Section 18.2 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," describes how the staff performs SPDS reviews for licensees requesting a pre-implementation review. The staff's review process will evaluate the (1) safety analysis report, (2) the implementation plan, and (3) the verification and validation (V&V) plan. In addition, three separate audit meeting/site visits, as described below, may be arranged through the Division of Licensing Project Manager. As dictated by the comprehensiveness of the applicant/licensee's documentation and the schedule for design and implementation of the SPDS, the objectives of these audits may be met in fewer site visits.

Design Verification Audit: The purpose of this audit meeting is to obtain additional information required to resolve any outstanding questions about the Verification and Validation (V&V) Program, to

confirm that the V&V Program is being correctly implemented, and to audit the results of the V&V activities to date. At this meeting, the applicant should provide a thorough description of the SPDS design process. Emphasis should be placed on how the applicant is assuring that the implemented SPDS will: provide appropriate parameters, be isolated from safety systems, provide reliable and valid data, and incorporate good human engineering practice. To the extent dictated by the completeness of the V&V program plan, the HFEB reviewer will arrange for participation of PSRB and ICSB reviewers at this meeting.

Design Validation Audit: After review of all documentation, an audit may be conducted to review the as-built prototype or installed SPDS. The purpose of this audit is to assure that the results of the applicant/licensee's testing demonstrate that the SPDS meets the functional requirements of the design and to assure that the SPDS exhibits good human engineering practice.

Installation Audit: As necessary, a final audit may be conducted at the site to ascertain that the SPDS has been installed in accordance with the applicant/licensee's plan and is functioning properly. A specific concern is that the data displayed reflect the sensor signal which measures the variable displayed. This audit will be coordinated with and may be conducted by the NRC Resident Inspector.

II. SUMMARY OF STAFF REVIEW

The staff reviewed the SPDS Safety Analysis Report for Monticello, and concludes that it is acceptable for the licensee to continue implementing its SPDS Program. The staff finds the parameter selection for the Monticello SPDS to be generally acceptable, but could not confirm that the validation testing of parameters will adequately cover beyond-design-basis conditions. The information needed by the staff to conduct a confirmatory review of this issue is defined herein.

III. EVALUATION

Northern States Power Company (NSP) submitted for staff review a Safety Analysis Report (Ref. 1) and Implementation Plan (Ref. 2) on the Safety Parameter Display System (SPDS) for the Monticello Nuclear Generating Plant. These submittals describe the display system, provide the design bases for the system, discuss parameter selection and display formats, describe the human factors considerations used in the design, and contain a Verification and Validation Plan for the design. The staff's review of the licensee's submittals is presented in the text which follows.

A. SPDS DESCRIPTION

The Monticello Safety Parameter Display System is a subsystem of a computer system known as the Plant Monitoring and Display System (PMDS). The SPDS was developed and supplied by General Electric Company (G.E.). The Monticello SPDS is identical in most respects to the generic G.E. Safety Parameter Display System that was audited by the staff as part of the GESSAR II Final Design Approval effort (Ref. 3 and 4).

The Monticello SPDS consists of three primary displays that are designed to support the information needs of the Emergency Procedure Guidelines (EPGs). These displays, RPV Control Display, Containment Control Display, and Critical Plant Variables Display, are elaborated in special function displays. The special function displays provide: 1) two-dimensional plots of the limiting conditions defined in the Emergency Operating Procedures (EOPs), e.g., Drywell Design Pressure Curve, 2) trend plots of all control parameters, showing data from the most recent 30 minutes; 3) the validation status of SPDS input data, and 4) radiation monitoring displays.

The Monticello displays differ slightly from the generic G.E. displays, but in ways that are likely to improve the useability of the displays. For instance, a radiation status box has been added. Because of the overall similarity of the Monticello design and the G.E. generic design, the staff has based some of the conclusions stated in this Safety Evaluation Report on its audit of the G.E. generic prototype (Ref. 4).

B. 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:

(i) Reactivity Control

(ii) Reactor core cooling and heat removal from the primary system

(iii) Reactor coolant system integrity

(iv) Radioactivity control

(v) Containment conditions."

For review purposes, these five items have been designated as Critical Safety Functions.

The selection of the SPDS process variables for display was made by G.E. based on the BWR generic Emergency Procedure Guidelines (EPGs). The staff confirmed that the variables selected are consistent with the presently approved BWR EPGs (Revision 3). The staff finds the SPDS variable selection acceptable. In addition, the staff noted that the design includes sufficient capability for expanding the system so that additional variables (such as hydrogen concentration) may be added as a result of future revisions to the generic EPGs.

The staff identified one potential shortcoming in the variable selection process. The system performance validation for Monticello includes simulator evaluations using transients selected to exercise the EOPs in coordination with the SPDS displays. The staff finds this basic approach acceptable; however, since the simulator is generally limited to design-basis events, the staff requests that the licensee provide additional discussion of provisions for SPDS/EOP walkthroughs to cover beyond design basis conditions (examples for consideration include primary containment venting at elevated pressures, liquid level measurements below the top of the active fuel, and conditions of high radiation in the reactor building and the plant vent stack).

C. DISPLAY DATA VALIDATION

The staff reviewed the NSP submittal to determine that means are provided in the design to assure that the data displayed are valid. Generally, the data validation includes range and consistency checks as well as data conditioning algorithms for compensating and weighting individual signal values.

Based on the above, the staff confirms that means are provided in the Monticello SPDS design to assure that the data displayed are validated.

D. HUMAN FACTORS PROGRAM

The staff also evaluated the NSP submittal for a commitment to a Human Factors Program in the development of the SPDS. As evidence of

licensee's commitment to a human factors program, the staff noted several aspects of the design process. First, the licensee has based the Monticello SPDS design on the generic G.E. design which was developed by a multi-disciplinary team and included a comprehensive human factors review (Ref. 3). Secondly, a contract is in effect between NSP and G.E. for review of the plant-specific SPDS with regard to display content, information accessibility, and data useability. Finally, a "Plant Interface and Human Factors Review Group" will review any plant-specific modifications, identify simulator and training requirements, and perform final static and dynamic system performance tests.

The displays appear to be well designed and are generally consistent with acceptable human factors principles. On the basis of the staff's knowledge about the G.E. design process (Ref. 3 and 4) and the information provided in the NSP submittals, the staff concludes that an appropriate commitment was made to incorporate human factors considerations into the design of the Monticello SPDS.

E. ELECTRICAL AND ELECTRONIC ISOLATION

NUREG-0737, Supplement 1, requires that the SPDS be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems.

The fiber-optic cable system used for isolation supplies the necessary electrical isolation to meet all requirements of maximum credible faults and electrical interference considerations. In addition, General Electric has performed environmental (IEEE-323-1974) and seismic (IEEE-384-1975) qualification tests as part of the qualification program for the isolators.

Based on its audit of the licensee's information on the isolation devices used in the Monticello design, the staff concludes that the design methodology and the hardware used (fiber-optic cable) for interfacing the SPDS with safety-related systems are acceptable, and that this equipment meets the Commission's requirements of NUREG-0737, Supplement No. 1.

IV. CONCLUSIONS

The NRC staff reviewed the Monticello Safety Analysis to confirm the adequacy of the parameters selected to be displayed to monitor 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, the staff concludes that no serious safety questions are posed by the proposed SPDS and, therefore, implementation may continue.

This conclusion is based on the following:

1. The variables selected for display are generally adequate to assess critical safety functions.

2. The licensee has stated that the SPDS will be suitably isolated from plant safety systems through the use of fiber optics cables.

3. The licensee's design provides means to assure that displayed data are validated.

4. The design and implementation of the licensee's design includes appropriate commitment to a human factors engineering program that should allow reasonable assurance that the information provided will be readily perceived and comprehended by its users.

V. REFERENCES

1. Letter from D. Musolf (NSP) to Director, Office of Nuclear Reactor Regulation (NRC) dated December 26, 1984 with attachment, "General Electric Topical Report, NEDC 30806P" (Proprietary).

2. Letter from D. Musolf (NSP) to Director, Office of Nuclear Reactor Regulation (NRC) dated May 31, 1984, with attachment, "General Electric Topical Report NEDC-30627" (Proprietary).

3. Safety Evaluation Report for the General Electric Company's GESSAR II Safety Parameter Display System, Appendix G, Supplement 4 of NUREG-0979, dated July 1985.

4. Memorandum for C. Thomas (NRC) from S. Weiss (NRC), Subject: Design Verification Audit Report for the General Electric Safety Parameter Display System; dated November 2, 1984.

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Dated: