UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
FORT CALHOUN STATION UNIT 1
OMAHA PUBLIC POWER DISTRICT
DOCKET NO. 50-285
SAFETY PARAMETER DISPLAY SYSTEM

I. INTRODUCTION

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

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 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. 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 (a) confirming the adequacy of the parameters selected to be displayed to detect critical safety functions, (b) confirming that means are provided to assure that the data displayed are valid, (c) 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 (d) 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 a serious safety question or seriously inadequate analysis, the Director of IE or the Director of NRR may require or direct the licensee to cease implementation.

II. SUMMARY

The staff reviewed the SPDS Safety Analysis Report for Fort Calhoun and concludes it is acceptable for the licensee to continue implementing its SPDS Program. The staff finds the process variables and system variables selected for display in the SPDS acceptable. Also, the licensee's design provides means to assure that displayed data are valid. However, continued implementation of the SPDS is conditional to a confirmatory staff review of a human factors report which the licensee has committed to submit to the NRC. In addition, the licensee did not provide sufficient information on the electrical and electronic isolation devices to allow the staff to complete its review of these devices.

III. EVALUATION

Omaha Public Power District (OPPD) submitted for staff review a Safety Analysis on the Safety Parameter Display System for the Fort Calhoun Station, Unit 1 on October 28, 1983 (Ref. 2). This submittal was in partial response to NRC Generic Letter No. 82-33 (Ref. 3). The licensee's report describes the basis on which the variables for display were selected for the SPDS. However, it did not contain information on how displayed data are validated, did not describe the human factors program used in the design and development of the display system, and did not describe how the SPDS is isolated from electrical and electronic interference with equipment and sensors that are used in safety systems. The staff requested additional information (Ref 5) and the licensee responded in Reference 6. The staff's evaluation of this material and of the Safety Analysis are presented below.

A. SPDS DESCRIPTION

The licensee's SPDS is a computer based display system. In Reference 1, the licensee states that the SPDS is part of two major modification efforts: (1) the Emergency Response Facility Computer System (ERFCS), and (2) the Inadequate Core Cooling System (ICCS).

The ERFCS is a real time data processing system which presents plant variables to the operator. The SPDS is made up of the ERFCS, 11 display terminals, and the Qualified Safety Parameter Display System (QSPDS).

The ICCS consists of three major components: (1) the Heated Junction Thermocouple (HJTC) System to monitor reactor vessel liquid inventory, (2) accident qualified Core Exit Thermocouples (CETs) for reactor temperature monitoring, and (3) a QSPDS which serves as a Class 1E data processing system for the HJTC, CETs and other 1E signals. The QSPDS provides the 1E data to the ERFCS via a fiber-optic serial data link.

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 process variables was based on Combustion Engineering Emergency Procedure Guidelines (CE EPGs). The CE EPGs have been reviewed and approved for implementation by a Staff Safety Evaluation dated July 29, 1983 (Ref. 4). A list of variables grouped by Critical Safety Functions was provided (Ref. 6) by the licensee and is reproduced as Table 1 herein. The staff reviewed the selected variables and concluded they are consistent with CE EPGs and are acceptable.

C. DISPLAY DATA VALIDATION

The staff reviewed the information provided by OPPD to determine that means are provided in the design to assure that the data displayed are valid. The licensee states that software routines stored in the ERFCS perform the data validation. Signals from instrument loops directly connected to the ERFCS are checked against pre-specified ranges. If the signal is not within the pre-specified range, an error message is displayed to the operator and also the last valid value of the displayed variable is followed by a blinking white question mark or a blinking white asterisk.

In Reference 6, the licensee states that the QSPDS checks analog input signals for an out-of-range condition as well as open thermocouple conditions. A suspect temperature reading within a set of core exit thermocouple signals is identified from the application of the Chauvenet Criterion to the data set. Signal values which fail the validity check are flagged before transmission to the ERFCS and are displayed on the ERFCS display screens with a blinking white question mark or a blinking white asterisk. The signal values which fail the validity check are also identified in the QSPDS display. In this display, suspect data is preceded by a question mark and, for out-of-range data, the numerical value of the data is replaced by a series of question marks.

Based on the information provided by the licensee, the staff confirms that means are provided in the SPDS design to assure that the data displayed are valid.

D. HUMAN FACTORS PROGRAM

The staff also evaluated the licensee's submittals for a commitment to a Human Factors Program in the development of the SPDS. The licensee states (Ref. 6) that the initial design of the SPDS did not include a formal documented review for human factors considerations. However, a human factors program is in the process of being developed in conjunction with the performance of the Detailed Control Room Design Review. Upon completion of the human factors review of the SPDS, a report will be submitted to the NRC. Further, the human factors program development and implementation will be completed on a schedule consistent with the implementation of the SPDS.

The staff plans to conduct a confirmatory review on the forthcoming human factors report. To facilitate this review, the staff recommends that the licensee include photographs/sketches of the top-level display formats for the control room SPDS. Similar data for the operator's interface to the display system should also be included in the report. The staff will report on the results of this review in a Safety Evaluation.

E. ELECTRICAL AND ELECTRONIC ISOLATION

NUREG-0737, Supplement No. 1, requires that the SPDS be suitably isolated from equipment and sensors that are used in safety systems to prevent electrical and electronic interference. The staff audited the licensee's information (Ref. 6) for the adequacy of the isolators (fiber-optic) between the safety systems and the SPDS. Fiber-optic cables serve as an interface between the Class 1E inputs and the ERFC. This fiber-optic data link extends between two modems, (data translation devices that change analog signals to digital and digital signals to analog). This unique isolator possesses inherent characteristics that eliminate ground loops and common ground shifts in electronic circuits and provides complete electrical ground isolation between transmitter and receiver. Fiber-optic cables present no fire hazards when their fibers are damaged. In addition, no local secondary damages can occur because fiber cables neither produce sparks nor dissipate heat. The construction of the fiber-optic cable is such that the cable contains no electrically conductive material. The relative permitivity (dielectric constant) of a material is a measure of the material's isolation capability. The dielectric constant of a material is referenced relative to free space (a vacuum) and is a dimensionless number. Dry air possesses a dielectric constant of 1.00059. Glass possesses a dielectric constant in the range of 4.0 to 7.0 depending upon the

specific type. The higher the dielectric constant, the greater the isolation that is provided. Thus, fiber-optic cables have an isolation capability that is 4 to 7 times greater than dry air. The voltage breakdown rating of a typical fiber-optic cable is on the order of 250 KV per meter.

A fault at either end of the data link might destroy the modem but will not propagate over the fiber-optic cable. For example, one of the tests that must be performed to qualify an isolator is the application of the maximum credible fault (voltage, current) to the output of the device to verify that the fault does not propagate or degrade the input (Class 1E side). This postulated failure does not affect the fiber-optic cable. As stated above, the optical fibers are totally dielectric (i.e., the electrical energy resulting from the fault will not propagate through the optical fiber). Another characteristic of the fiber-optic cable is its nonsusceptibility to the coupling of cross-talk and electromagnetic interference (EMI).

These devices are installed in the control room which is a mild environment. The requirement for environmental qualification (10 CFR 50.49) is not applicable to a mild environment. The licensee states that there is documentation provided by Combustion Engineering (primary QSPDS vendor) certifying seismic and Class 1E qualification of these devices.

Insufficient information was provided for nine types of isolators which directly interface with the ERFC. The information supplied in Omaha Public Power District's letter dated December 7, 1984 (Ref. 6) did not address the qualification of the following devices: (a) Technology for Energy (TEC) model 156 isolators, (b) Scientific Columbus model 7005-SC-BA Isolating Transmitters, (c) Gems/Deval model RE-36562 receivers, (d) Comsip, Inc. AGM series 4000 transmitter, (e) Foxboro model N-240-V21 isolators, (f) Scientific Columbus model UT110A2 transducer, (g) Relay Contacts, (h) Coil to Contact Isolation, and (i) Reactor Protection System buffered outputs.

The licensee shall provide the following information to the NRC for confirmatory review of these devices:

a. For each type of device used to accomplish electrical isolation at Fort Calhoun, describe the specific testing performed to demonstrate that the device is acceptable for its application(s). This description should include elementary diagrams where necessary to indicate the test configuration and how the maximum credible faults were applied to the devices.

b. Data to verify that the maximum credible faults applied during the test were the maximum voltage/current to which the device could be exposed, and define how the maximum voltage/current was determined.

c. Data to verify that the maximum credible fault was applied to the output of the device in the transverse mode (between signal and return) and other faults were considered (i.e., open and short circuits).

d. Define the pass/fail acceptance criteria for each type of device.

e. Provide a commitment that the isolation devices comply with the environmental qualifications (10 CFR 50.49) and with the seismic qualifications which were the basis for plant licensing.

f. Provide a description of the measures taken to protect the safety systems from electrical interfence (i.e., Electrostatic Coupling, EMI, Common Mode and Crosstalk) that may be generated by the SPDS.

IV. CONCLUSIONS

The NRC staff reviewed the Fort Calhoun 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 from safety systems. Based on its review to date, the staff concludes that no serious safety questions are posed by the proposed SPDS and, therefore, implementation may continue.

These conclusions are based on the following:

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

2. The licensee's design provides means to assure that displayed data are valid.

3. The fiber-optic cables used within the licensee's design are qualified isolators and are acceptable for interfacing the SPDS with safety systems.

However, this conclusion is subject to a confirmatory staff review of additional information needed from the licensee. The continued implementation of the SPDS by the licensee is conditional to a satisfactory confirmatory review by the staff on the licensee's human factors report. Also, adequate information was not provided by the licensee for the staff to confirm acceptability of all of the isolation devices used to interface the SPDS with the plant. We request the licensee provide information on these devices for the staff to conduct a confirmatory review.

The conclusion that 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 or when sufficient information is available for the staff to make such a determination.

V. REFERENCES

1. Letter from W. C. Jones, Omaha Public Power District, to R. A. Clark, NRC, subject: "Generic Letter 82-33, Supplement 1 to NUREG-0737, Emergency Response Capabilities," dated April 15, 1983.

2. Letter from W. C. Jones, Omaha Public Power District, to James R. Miller, NRC, subject: "Safety Parameter Display System Analysis," dated October 28, 1983, with Attachment, "SPDS Parameter Selection, Safety Analysis."

3. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," USNRC Report NUREG-0737, Supplement No. 1, January 1983.

4. Safety Evaluation of "Emergency Procedure Guidelines," for CEN-152, Revision 1 (SER dated July 29, 1983).

5. Letter from J. R. Miller, NRC, to W. C. Jones, Omaha Public Power District, subject: "Request for Additional Information," dated July 24, 1984.

6. Letter from R. L. Andrews, Omaha Public Power District, to J. R. Miller, NRC, Subject: "Safety Parameter Display System (SPDS)," dated December 7, 1984.

TABLE I

Relationship to Critical Safety Functions
and SPDS Parameter Selection Safety Analysis

1. NUREG-0737, Supplement 1, Critical Safety Function: Reactivity Control

Corresponding EPG Safety Function(s): Reactivity Control

EPG Parameters	SPDS Variables
CEA Bottom Lights	CEA Full In Positions
Reactor Power	Wide Range Log Power
Startup Rate	Startup Rate
Boron Concentration	Boronometer Boron Concentration

2. NUREG-0737, Supplement 1, Critical Safety Function: Reactor Core Cooling and Heat Removal From the Primary System

Corresponding EPGs Safety Function(s): RCS and Core Heat Removal

EPG Parameters	SPDS Variables
Steam Generator Level	Wide Range Steam Generator Level (Both)
Steam Generator Pressure	Wide Range Steam Generator Pressure (Both)
Feedwater Flow	Feedwater Flow to Both Steam Generators
T_H Temperature	RTD's on Both Hot Legs
T_C Temperature	RTD's on All Four Cold Legs
ΔT Temperature	ΔT Between Hot and Cold Legs
T_{AVE} Temperature	T_{AVE} Temperature
RCS Subcooling	Saturation Margin Saturation Margin Upper Head Saturation Margin
ECCS Delivery	HPSI Flow LPSI Flow
CET Temperature	Maximum CET Temperature All CET Temperatures Representative CET Temperatures
RCS Pressure	Pressurizer Pressure

3. NUREG-0737, Supplement 1, Critical Safety Function: Reactor Coolant System Integrity

Corresponding EPG Safety Function(s): RCS Inventory and Pressure Control

EPG Parameters	SPDS Variables
Pressurized Level	Pressurizer Level
RCS Pressure	Pressurizer Pressure
RCS Subcooling	Saturation Margin Saturation Margin Upper Head Saturation Margin
ECCS Delivery	HPSI Flow LPSI Flow
	Reactor Vessel Level

4. NUREG-0737, Supplement 1, Critical Safety Function: Radioactivity Control

Corresponding EPG Safety Function(s): Containment Isolation

EPG Parameters	SPDS Variables
Containment Radiation Monitors	Containment Radiation Monitors
Containment Pressure	Containment Pressure
Containment Isolation Valve Status	Status of Cont. Iso. Valves
Steam Plant Radiation Monitors	Secondary System Activity Monitors - Main Steam Line Monitor - Gaseous Effluent Monitors - Condenser Off-Gas Monitor - Liquid Effluent Monitors

5. NUREG-0737, Supplement 1, Critical Safety Function: Containment Conditions

Corresponding EPG Safety Function(s): Containment Temperature, Pressure, and Combustible Gas Control

EPG Parameters	SPDS Variables
Containment Temperature	Containment Temperature
Containment Pressure	Containment Pressure
Containment H_2 Concentration	Containment H_2 Concentration
Containment Spray Flow	Containment Spray Flow
Containment Sump Level	Containment Sump Level

Adequacy of SPDS Variables: As detailed in the above safety analysis, all EPG parameters are monitored by the Fort Calhoun Station SPDS.

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