March 24, 1986

Docket Nos. 50-317
50-318

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Mr. J. A. Tiernan
Vice President-Nuclear Energy
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, MD 21203

Dear Mr. Tiernan:

The staff reviewed the Baltimore Gas and Electric Company's Safety Parameter Display System Analysis (SPDS) for Calvert Cliffs Unit 1 and 2 in connection with TMI Action Item I.D.2. Based on the results of the review, it is concluded that no serious safety questions are posed by the proposed SPDS and therefore, implementation of the SPDS should continue.

The NRC staff reviewed the SPDS analysis to confirm the adequacy of the variables 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. The details of this review are contained in the enclosed Safety Evaluation.

With regard to the suitability of isolation, based on the information obtained from Baltimore Gas and Electric Company, we note that a change is being contemplated in isolation method. In the event that the isolation method is modified, additional test information on maximum credible fault for the data acquisition system input modules will be required for confirmatory review by the staff.

The conclusion that SPDS implementation should 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.

Sincerely,

/S/ Original signed by:

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
BALTIMORE GAS & ELECTRIC COMPANY
CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2
DOCKET NOS. 50-317 AND 50-318
SAFETY PARAMETER DISPLAY SYSTEM

I. 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 room of their plant. 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 review of the 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 (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 serious safety questions or seriously inadequate analyses, the Director of IE or the Director of NRR may require or direct the licensee to cease implementation.

## II. SUMMARY

The staff reviewed the Baltimore Gas and Electric Company's SPDS Safety Analysis for Calvert Cliffs Units 1 and 2. Based on the results of the review, 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.

## III. EVALUATION

### A. Background

The Baltimore Gas and Electric Company submitted to the NRC a Safety Analysis (Reference 1) on the Safety Parameter Display System (SPDS). The staff reviewed the analysis and, because of insufficient information, was unable to complete the review. A request for additional information (Reference 2) was forwarded to the licensee and the licensee's response (Reference 3) was evaluated by the staff. This safety evaluation is based on the results from the staff's review of the material identified.

### B. Description

The licensee's SPDS is computer-based and is an integral part of of new plant computer. The display units consist of cathode-ray tubes with one 13-inch touch screen monitor per unit, mounted on the shift supervisor's console, and a 13-inch desk-mounted monitor in the Technical Support Center.

The library of display formats in the SPDS is organized under the following licensee identified Critical Safety Function (CSF) headings:

1. Reactivity
2. RCS Pressure and Inventory
3. Core/RCS Heat Removal

4. Containment Environment
5. Containment Isolation
6. Radioactivity Control
7. Vital Auxiliaries

The licensee states that under each CSF heading, parameters are displayed which support the CSF in a manner consistent with the new function-oriented Emergency Operation Procedures (EOPs). Furthermore, these displays make extensive use of color and data coding techniques. Displays are selected by the operator through keyboard action, the CRT cursor, or the touch screen poke points.

C. 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 (CSFs).

In the evaluation of the functions and variables monitored by the licensee's SPDS, the staff considered the generic emergency procedure guidelines, which are required by I.C.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," as a principal technical source of variables important to operational safety. The licensee states (Reference 3) that the variables displayed on the SPDS were selected to provide the indications required to verify that the safety functions described in CEN-152, "Combustion Engineering Emergency Procedure Guidelines," are being fulfilled. In addition to the safety functions identified in CEN-152, the licensee added two additional safety functions to the SPDS: 1) radiation control, and 2) maintenance of vital auxiliaries.

The licensee's safety functions consist of seven functions. The seven critical safety function headings were identified earlier in the report under Section III.B, Description. The staff's review of these functions concludes that they encompass the five Critical Safety Functions identified in NUREG-0737, Supplement 1.

The staff reviewed the variables selected by the licensee for each of the Critical Safety Functions and found that the variables selected did provide the minimum information needed by an operator to evaluate each of the critical safety functions. Moreover, on February 11, 1986, the staff spoke with the licensee's personnel to determine the range of displayed neutron flux. It was informed that the range was from $10^{-8}$ to 200% of design power, which it finds adequate in monitoring reactivity control.

The staff review noted that the licensee selected many system variables for display in the SPDS. Many of these system variables are pertinent to the Engineered Safeguard Systems of the plant. The staff recognizes the addition of these variables as an enhancement to the SPDS, which should aid operators in the execution of the Emegency Operation Procedures.

Based on the review of the variables selected by the licensee for display in the SPDS, the staff confirms that the variables are adequate to evaluate the status of the critical safety functions required by the NRC.

D. 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. It found that data validation techniques are used in the SPDS alarm algorithms and in the data algorithms, which prepare sensor signals for display as process variables. Inverse video and reverse video are used to code invalid data on the screen.

In data validation, the licensee indicates that, for a process variable which has two or more instrument channels, a comparison of the signals from each channel is made. After the comparison of the signals, the two signals in closest agreement are averaged and then used for display purposes. Also, an instrument loop uncertainty is used to evaluate the validity of each signal. The loop uncertainty is based upon worst case accuracy of components within the loop. If the signals from the two closest channels deviate from each other by more than the calculated loop uncertainty, then the data will be flagged as invalid.

Based on the review of the data validation techniques and the data coding techniques used in the licensee's design, the staff confirms that means are provided to assure that the data displayed are valid.

E. Human Factors Program

The staff evaluated 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. The review found the display hierarchy structured and based upon the Critical Safety Functions. Furthermore, access to a specific display may be achieved by one of three methods: keyboard action, CRT cursor, or touch screen poke points. Also, each display page includes a matrix of CSF alarm windows to alert the operator to the status of all CSFs, no matter which display is being viewed at a given time. These features of the design aid the operator when rapid access to specific data is needed.

The licensee also described how color is used to code the status of data displayed in the SPDS. Staff review of this use of color concluded that it conformed to accepted human factor practices; green for normal, yellow for caution, and red for danger.

The principal users of the SPDS will be the Shift Supervisor and the Shift Technical Advisor. The Shift Supervisor will be seated directly in front of the 13-inch monitors at a viewing distance of approximately 20 inches. This location of the SPDS appears to be consistent with the Shift Supervisor's duties.

Based on this evidence, the staff confirms that human factors engineering was an integral part of the licensee's SPDS design process.

In the Safety Analysis Report, the licensee states that a design verification and validation program will be implemented. The licensee identified the following as design review checkpoints:

a. Identification and definition of critical safety functions
b. Determination of supporting I/O
c. Formatting of displays
d. Development of alarm algorithms
e. Definition and man machine interface protocol
f. Review of program flowcharts
g. Review of coding
h. Witness of validation testing.

The staff was unable to evaluate the depth and scope of this program; however, it does appear elements of a design verification and validation program do exist.

F. Electrical and Electronic Isolation

The licensee's SAR did not address the requirement that SPDS must be suitably isolated from equipment sensors that are used in safety systems to prevent electrical and electronic interference. The licensee's response to the staff's request for additional information on isolation devices is presented in Reference 3. The staff's review of this information follows.

All information collected by the SPDS from Class 1E safety-related instrument systems is received by a Data Acquisition (DAS). The DAS includes remote Input/Output (I/O) cabinets, which receive analog and digital information from plant instrumentation loops. The output of the DAS is a high speed serial data link (fiber optic cable). The inputs are isolated with various modules selected for the specific signal type. These isolator modules have undergone some qualification testing as specified in Reference 3 although sufficient information has not been submitted for the staff to determine their qualification as isolators. The licensee states that these I/O cabinets are presently supplied with qualified Class 1E power and are themselves qualified as Class 1E equipment; therefore, the fiber optic cable becomes the isolator between Class 1E safety-related equipment and the non-1E SPDS. The licensee informed the staff during a June 17, 1985 telephone conference that at some time in the future, they plan to remove the Class 1E power from the I/O cabinets and use the input modules mentioned above as the required isolation between Class 1E instrument loops and the SPDS. If and when this modification takes place, the licensee will be required to qualify these devices to withstand the maximum credible fault and provide additional information to the staff for confirmatory review.

The fiber optic cable transmits digital information using light instead of electric current and is a unique isolator, which 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 optics neither produce sparks nor dissipate heat. The construction of the fiber optic cable is such that the cable contains no electrically conductive material. 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 these 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 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 optical fiber cable is its nonsusceptibility to the coupling of cross-talk and electromagnetic interference (EMI).

Based on the staff's audit of the Baltimore Gas and Electric information on the isolation devices (fiber optic cables) in the Calvert Cliffs design, it concludes that the design methodology and the hardware used 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. In the event the isolation method (fiber optic cables to input modules) is modified as discussed in the June 17, 1985 telephone conference, additional testing information on the maximum credible fault for the DAS input modules will be required for confirmatory review.

## IV. CONCLUSIONS

The NRC staff reviewed Baltimore Gas and Electric Company's Calvert Cliffs Analysis to confirm the adequacy of the variables 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 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.

However, based on the information obtained from the licensee, the staff noted that the licensee is contemplating a change in isolation method. In the event that the isolation method is modified, additional test information on maximum credible fault for the DAS input modules will be required for confirmatory review by the staff.

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.

Date: March 24, 1986

Principal contributor: L. Beltracchi

References

1. Letter from A. E. Lundvall, Baltimore Gas and Electric Company. to J. R. Miller, NRC, Subject: Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Safety Parameter Display System, Safety Analysis, dated June 6, 1984.

2. Letter from J. R. Miller, NRC to A. E. Lundvall, Baltimore Gas and Electric Company, Subject: Request for Additional Information, Safety Parameter Display System, dated October 23, 1984.

3. Letter from A. E. Lundvall, Baltimore Gas and Electric Company, to J. R. Miller, NRC, Subject: Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Safety Parameter Display System, dated February 4, 1985.

# TABLE 1
## SAFETY FUNCTIONS AND VARIABLES

| Safety Function | Variables |
|---|---|
| CSF #1: Reactivity Control | Nuclear Flux Power Level |
| | Start-up Rate |
| | Charging Flow |
| | ECCS Flow |
| | Boron Concentration |
| | Cold Leg Temperature |
| | CEA Position |
| | Status of Letdown Isolation Valves |
| CSF #2: RCS Pressure/Integrity | Pressurizer Pressure |
| | Saturation Margin |
| | Pressurizer Level |
| | Reactor Vessel Level |
| | Core Exit Temperature (CET) |
| | Net Charging Flow |
| | HPSI Flow |
| | LPSI Flow |
| | Containment Water Level |
| | Status of RCS Relief and Vent Valves |
| | Containment Area Radiation |
| | Secondary System Radiation |
| | Main Vent Radiation |
| | Status of SIAS (Safety Injection Actuation Signal) |
| | Status of RAS (Recirculation Actuation Signal) |
| | Component Cooling System Head Tank Level |
| | Quench Tank Level |
| | Quench Tank Temperature |
| | Quench Tank Pressure |
| | Volume Control Tank (VCT) Level |

| Safety Function | Variables |
|---|---|
| CSF #3: Core/RCS Heat Removal | Saturation Margin |
| | RCS Flow |
| | $T_{hot} - T_{cold}$ |
| | $T_{cold}$ |
| | Steam Generator Level |
| | Total Feedwater Flow |
| | Steam Flow |
| | Pressurizer Pressure |
| | Reactor Vessel Level |
| | Core Exit Temperature |
| CSF #4: Containment Environment | Containment Pressure |
| | Containment Temperature |
| | Containment Spray Flow |
| | Total Service Water to Containment Coolers |
| | Containment Radiation |
| | Hydrogen Concentration |
| | Containment Water Level |
| | Main Vent Radiation |
| | Status of CSAS (Containment Spray Actuation Signal) |
| CSF #5: Containment Isolation | Status of Containment Isolation Valves open directly to containment atmosphere |
| | Main Vent Radiation |
| | Containment Area Radiation |
| | Status of CIS (Containment Isolation Signal) |

| Safety Function | Variables |
|---|---|
| CSF #6: Radiation Control | Containment Radiation |
| | Main Steam Effluent Radiation |
| | Condenser Off-Gas Radiation |
| | Steam Gen Blowdown Tank Radiation |
| | Steam Gen Blowdown IX Radiation |
| | Main Vent Radiation |
| | Control Room Vent Radiation |
| | Spent Fuel Pool Vent Radiation |
| | ECCS Pump Room Vent Radiation |
| | Access Control Area Vent Radiation |
| | Gas Waste Discharge Radiation |
| | Liquid Waste Discharge Radiation |
| | Service Water System Radiation |
| | Letdown System Radiation |
| | Component Cooling System Radiation |
| CSF #7: Vital Auxiliaries | Saltwater System Header Pressures |
| | Services Water System Header Pressures |
| | Component Cooling Water System Header Pressure |
| | Instrument Air Pressure |
| | Indication of voltage (nominal/low) on 4 kV bus 11, 14, 21, 24 |
| | Indication of voltage (nominal/ on 480 V bus 11A, 11B, 14A, 14B, 21A, 21B, 24A, 24B |
| | Indication of voltage (nominal/low) on 125 VDC bus 11, 12, 21, 22 |
| | Indication of voltage (nominal/low) on Vital 120 VAC bus 11, 12, 13, 14 |