

DESIGN VERIFICATION AUDIT  
OF THE  
SAFETY PARAMETER DISPLAY SYSTEM  
FOR THE  
BYRON GENERATING STATION UNITS 1 AND 2

September 26, 1985

Prepared for  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

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## TABLE OF CONTENTS

<u>Section</u>	<u>Page</u>
1.0 INTRODUCTION . . . . .	1
2.0 BACKGROUND . . . . .	2
3.0 SYSTEM DESCRIPTION . . . . .	3
4.0 AUDIT FINDINGS . . . . .	3
4.1 System Specifications and Standards Used in the Design. . . . .	3
4.2 Design of Display Formats . . . . .	4
4.3 Incorporation of Human Factors Requirements into Software Specifications . . . . .	4
4.4 Design, Code, Test, Test Software, and Data Base Instructions. . . . .	5
4.5 Integration Tests and Test Results for Displays and Scenarios . . . . .	5
4.6 Display Format Human Engineering Standard and Guidelines. . . . .	6
4.7 Display and Control Hardware Evaluation . . . . .	7
4.8 Design Validation Test Methods and Test Plans . . . . .	7
5.0 SUPPLEMENT 1 TO NUREG-0737 REQUIREMENTS. . . . .	9
5.1 Concise, Continuous Display . . . . .	9
5.2 Convenient Location . . . . .	11
5.3 Incorporation of Human Factors Principles . . . . .	11
5.4 Procedures. . . . .	11
5.5 Training For Accident Response With and Without SPDS. . . . .	12
5.6 Parameter Selection . . . . .	12
5.7 Electrical/Electronic Isolation . . . . .	13
CONCLUSIONS. . . . .	13

## TABLE OF CONTENTS (Continued)

<u>Section</u>	<u>Page</u>
REFERENCES . . . . .	15
ATTACHMENT 1     Audit Plan for Byron 1 and 2 Safety Parameter Display System	
ATTACHMENT 2     List of Audit Meeting Attendees	
ATTACHMENT 3     Human Factors SPDS Review Schedules	
ATTACHMENT 4     Commonwealth Edison Verification and Validation Schedule	

DESIGN VERIFICATION AUDIT REPORT  
FOR  
COMMONWEALTH EDISON COMPANY'S  
BYRON UNIT 1 AND 2  
SAFETY PARAMETER DISPLAY SYSTEM

1.0 INTRODUCTION

This report documents the findings of the Nuclear Regulatory Commission (NRC) design verification audit of Commonwealth Edison Company's Byron Generating Station Units 1 and 2 Safety Parameter Display System (SPDS). The purpose of the audit, as described in NUREG-0800, Section 18.2 (Reference 1), was to obtain additional information required to resolve any outstanding questions with regard to the verification and validation (V&V) program, to confirm that the V&V program is correctly implemented, and to audit any results available to date. Because of the advanced state of the Byron program, the audit included a review of the installed SPDS to ensure that the results of Commonwealth Edison Company's testing demonstrate that the SPDS meets the functional requirements and exhibit good human engineering practice. The requirements set forth in NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability" (Reference 2), served as the basis for the audit.

The history of the Byron SPDS dates back to December 29, 1983, when Commonwealth Edison Company submitted an SPDS Safety Analysis Report to the NRC (Reference 3). The SPDS Safety Analysis Report contained a verification and validation plan for all Commonwealth Edison Company stations along with the criteria for parameter selection for Byron and Braidwood Stations. On November 9, 1984, the NRC sent a Request for Additional Information Concerning the Byron/Braidwood Safety Parameter Display System to the licensee (Reference 4). Concerns regarding isolation devices, human factors engineering information, unreviewed safety questions, implementation plan and procedures, and systems review information were included in the NRC request for additional information. On February 6, 1985, the NRC sent an Audit Plan for the Byron 1 and 2 Safety Parameter Display System to the licensee (Reference 5). The audit agenda is included as Attachment 1 to this report.

The audit was conducted on July 24, 25 and 26, 1985, at Commonwealth Edison Company's Byron Units 1 and 2 facilities. The NRC audit team consisted of a representative from the NRC Division of Human Factors Safety, Human Factors Engineering Branch, and consultants from Science Applications International Corporation and Comex Corporation. This report was prepared by Science Applications International Corporation, but is intended to reflect the consolidated observations, conclusions, and recommendations of the NRC audit team members. The audit followed an agenda prepared by the NRC and forwarded to Commonwealth Edison Company on February 13, 1985. A list of audit meeting attendees is presented in Attachment 2 to this report.

## 2.0 BACKGROUND

Licensees and applicants for operating licenses are required to provide a Safety Parameter Display System (SPDS). The objective is to improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them (NUREG-0660, Item I.D.1, Reference 6). The need for an SPDS was confirmed in NUREG-0737 (Reference 7) and in Supplement 1 to NUREG-0737. SPDS requirements in Supplement 1 to NUREG-0737 replaced those in earlier documents. Supplement 1 to NUREG-0737 requires each licensee or applicant to implement an SPDS on a schedule negotiated with the NRC. Human factors guidelines for SPDS design are currently provided in NUREG-0800 and NUREG-0700 (Reference 8).

An SPDS is to be established according to the applicant's own safety analysis and implementation plan which must be submitted to the NRC. According to Supplement 1 to NUREG-0737, the written safety analysis shall include a description of 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. This safety analysis and the specific implementation plan for the SPDS are to be reviewed by the NRC. On-site audits are to be scheduled as necessary to confirm that the applicant is implementing an adequate design program.

### 3.0 SYSTEM DESCRIPTION

The same SPDS system will be used at both the Byron and Braidwood Stations. The system chosen is a derivative of the Westinghouse Iconic display design, modified by in-house engineering and human factors studies, and motivated by the objective of limiting the number of "display pages" comprising the SPDS.

Two display pages are used: a narrow-range and a wide-range. The narrow-range is designed for use at full operating power, and the wide-range is principally for accident conditions where parameters may vary over a wide range. Each Iconic has eight spokes. Each spoke displays one parameter or a combination of parameters. Each page is considered by the licensee to represent all the critical safety functions. Normal operating conditions are intended to display a symmetrical octagon on the screen. The Iconic distorts to an asymmetrical octagon when the parameters depart from a normal reference state. The licensee stated that the octagonal display can be used for all conditions of intended use, cold shutdown through full power as well as accident conditions.

There is an SPDS screen on the vertical display console in each of the two Byron plants. Reliance on administrative procedures ensures that the SPDS is continuously displayed to the operators.

### 4.0 AUDIT FINDINGS

#### 4.1 System Specifications and Standards Used in the Design.

The system specifications and standards used in the design of the Byron SPDS are presented in two basic documents: the "Westinghouse Iconic Display Design," and the Commonwealth Edison Company "SPDS Plant Specific Supplement." A third document, entitled "SPDS Functional Description," was produced by ARD Corporation, but was not used for specification purposes. These documents were available for our audit at the plant site, but they are not docketed with the NRC.

The Westinghouse Iconic Display Design document (a proprietary document) contains the design for both wide-range and narrow-range Iconic

displays along with the generic functional descriptions, equations, and algorithms used to drive the individual parameters on the Iconic display spokes.

The Commonwealth Edison Company's "SPDS Plant Specific Supplement" describes the plant-specific changes to the Westinghouse specifications. Changes made to the plant-specific SPDS include the addition of a parameter spoke which displays water level in the upper head of the reactor vessel (RVLIS), the display of the subcooling margin, a change in pressurizer RC pressure units from PSIA to PSIG, and the addition of specific identification points for the radiation spoke.

The NRC audit team review of the Westinghouse and plant-specific specification documentation indicated that Commonwealth Edison Company followed a process which adequately defined the general detail and plant-specific properties of the SPDS.

#### 4.2 Design of Display Formats

During the audit of the process used by Commonwealth Edison Company to design the display formats, the audit team selected the spoke which displays pressurizer pressure on the narrow-range Iconic and system pressure on the wide-range Iconic. This was done in order to make the most effective use of the limited audit time by tracing the details of one display component from the initiation phase to the implementation phase.

The basic display formats for the wide-range Iconic and narrow-range Iconic were developed by Westinghouse Corporation during the development of the displays. Plant-specific changes such as the addition of subcooling, reactor vessel level indication system, and radiation monitoring points were made by Commonwealth Edison Company and validated on prototype displays. The operations department was formally included in the design process.

The audit team concluded that the display formats were appropriately designed by the Westinghouse Corporation and Commonwealth Edison Company.



#### 4.3 Incorporation of Human Factors Requirements into Software Specifications

ARD Corporation was employed by Commonwealth Edison Company to provide human factors guidance in software specifications and hardware selection. ARD ensured that human factors considerations such as functional grouping, color-coding consistency of display elements, and abbreviation consistency of text were included in the graphics generation package.

It is our evaluation that human factors requirements were appropriately integrated into the software specifications.

#### 4.4 Design, Code, Test, Test Software, and Data Base Instructions

The audit team performed a limited inspection of the code and concluded that it appeared to be reasonably developed. Our inspection of the list of program instructions concluded it contained several comments made by the systems analysts indicating an appropriate checking and cross-checking process in the development of the code.

#### 4.5 Integration Tests and Test Results for Displays and Scenarios

The computer systems group performed software testing in two phases. The first phase performed laboratory testing of the algorithm software and graphic generation package prior to the installation phase. The algorithm software was tested with a formal test package. The graphic generation package was tested on a Prime computer.

During the second phase, on-site testing of the installed system from sensor to data base was performed by using two Commonwealth Edison Company test procedures, SPP 85-10 and 11. One test procedure was for the wide-range display software and one was for the narrow-range display software. The audit team review of this documentation revealed that the systems analysts performing the tests on the wide-range loop pressure spoke on the Iconic did discover a pressure spoke problem created by the operating characteristics of reactor coolant pumps. In this case, a change is being made to modify the SPDS to accommodate the actual operating characteristics of the plant. The test results and problems were thoroughly documented in the test procedures.



It is our conclusion that the integrated testing process and test results reflect an appropriate test methodology which was thoroughly implemented.

#### 4.6 Display Format Human Engineering Standard and Guidelines

The audit team performed an on-site evaluation of the display format and concluded that the format generally followed good human engineering practices and guidelines. However, the audit team did note several human engineering discrepancies. Those discrepancies are listed below.

First, the SPDS consists of a narrow-range display and a wide-range display. There is no display title to identify which display is on the screen. This does not conform to NUREG-0700 guideline 6.6.1.1 which states that displays should be clearly labeled to permit accurate human performance. The audit team recommended that WIDE RANGE and NARROW RANGE be considered as titles by the licensee.

Second, the red alarm bars at the end of each Iconic spoke are difficult to detect as they have a low color contrast with the grey background. This does not conform with NUREG-0700 guideline 6.5.1.6.e(2) which states that colors should contrast well with the background on which they appear.

Third, the wide-range steam generator level spoke does not cover the full range of plant operations as indicated in Safety Parameter Display System Documentation. The problem is that the reference level is set at 86% which is the correct reference level for operations below 2000°F in the primary coolant system. During power operations when core exit temperature is referenced at 6170°F, the reference level in the steam generators is 66% on the narrow-range which corresponds to about 60% on the wide-range scale. Therefore, the Iconic will indicate a misleading low level indication when the operator switches from narrow-range display to wide-range display. This is an indication that the SPDS development team did not perform a thorough analysis of operator tasks in relation to system engineering and system

functional objectives to establish operator information requirements as recommended in NUREG-0700 guideline 6.5.1.1.a. Commonwealth Edison Company agreed to investigate and correct this problem.

A human factors evaluation of the completed SPDS will be conducted by ARD Corporation in mid-1986. This review will include a checklist survey based on NUREG-0800, Section 18.2 criteria. It will also include operator interviews and review of the detailed control room design review task analysis data. Human engineering discrepancies documented during this review will be assessed in a process similar to the detailed control room design review assessment. The process for this evaluation is included as Attachment 3. This is an appropriate method for verifying and validating the suitability of the SPDS.

#### 4.7 Display and Control Hardware Evaluation.

Human factors considerations have been a part of the SPDS selection and implementation process at Byron. Commonwealth Edison Company's human factors consultants, ARD Corporation, prepared "Human Factors Considerations to Monitor Selection" in 1982, prior to procurement of the display hardware. Since 1982, ARD has played an active role in the human factors aspects of the implemented system. Further, a detailed evaluation of the human factors suitability of the implemented SPDS will be performed in mid-1986.

The audit team observed that the SPDS locations, controls, and hardware conformed to good human engineering practices.

#### 4.8 Design Validation Test Methods and Test Plans.

In order to conduct a detailed audit of the design validation test methods and plans, the audit team concentrated its efforts on one iconic spoke. The spoke selected for evaluation was the pressure spoke which displays pressurizer pressure on the narrow range display and primary system pressure on the wide range display.

The pressurizer pressure spoke displays the redundant input average of the pressurizer pressure inputs along with the constant reference value of 2235 psig. If the active value is invalid, then it is displayed as XXX and

the iconic is forced to the full deflection high alarm state. If the active value is valid, then it is compared to the reference value and the iconic is deflected from the reference octagon accordingly. The full deflection high alarm state is achieved at 2335 psig and is based on pressure operated relief valve (PORV) lifting specifications. The full deflection low alarm stated is based on safety injection which begins at 2000 psig.

The wide range pressure spoke displays reactor coolant system loop pressure and the pressure reference in units of psig. If A and C reactor coolant pumps are running or if no pumps are running the active reactor coolant system pressure is the redundant average of the wide range pressures. Both alarm limits and the reference are functions of core exit temperature and the reactor trip status. Since core exit temperature and the reactor trip status are determined by the core exit temperature spoke, this spoke utilizes these values in the reactor coolant system pressure limit calculations.

The audit team evaluation of the pressure spoke included a review of the Westinghouse system specification and Commonwealth Edison Company's validation test procedures SPP-10 and SPP-11 for testing of the narrow range and wide range iconic logic. The system specifications appropriately define the general detail and plant specific properties such as point identifications and sensors, etc. Our limited inspection of the software code concluded that it was reasonably developed and contained several comments made by the systems analysts indicating a thorough analysis of the code. Software tests procedures contained in SPP-10 and SPP-11 were conducted on the narrow range pressurizer pressure spoke and the wide range reactor coolant system spoke. As a result of the SPP-11 tests on the reactor coolant system pressure spoke, the Commonwealth Edison Company systems analysts identified pressure spoke problems created when coolant pumps are started. They are taking appropriate steps to modify the software to accomodate actual system characteristics.

Our audit of the data validation for the pressure spoke concluded that the licensee's methodology and coding of displayed data was appropriate.

Another design verification and validation process will be conducted by an independent group within the Commonwealth Edison Company. The

independent group is scheduled to perform the verification and validation in September, 1987. The schedule for all Commonwealth Edison Company verification and validation reviews conducted by the independent review group is included in Attachment 4. No procedures or details were available for this activity at the time of the audit.

## 5.0 SUPPLEMENT 1 TO NUREG-0737 REQUIREMENTS

### 5.1 Concise, Continuous Display.

Supplement 1 to NUREG-0737 states that "... the SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant."

By definition, a concise display removes superfluous or expanded detail in order to summarize the status of the five critical safety functions. The two Byron SPDS Iconics display specific plant parameters which can be used in to evaluate the critical safety functions. The following is a list of Byron parameters associated with plant function monitoring. This list was included as Table E.17-1 of the B/B-FSAR which was published in September, 1983.

#### Reactivity Control

- Power Mismatch
- T<sub>avg</sub>
- Startup Rate
- Core Exit Temperature

#### Reactor Core Cooling

- Core Exit Temperature
- NR SG Level
- WR SG Level

#### Reactor Coolant System Integrity

- NR SG Level
- WR SG Level
- WR RCS Pressure

Pressurizer Level  
Pressurizer Pressure  
Net Charging/Letdown Flow Rate

Reactor Coolant System Inventory Control

Net Charging/Letdown Flow Rate  
Pressurizer Level  
Containment Floor Drain Sump Level

Containment Activity Level

Containment Activity  
Containment Floor Drain Sump Level

Containment Integrity

Containment Temperature  
Containment Pressure

Secondary System Status

NR SG Level  
WR SG Level  
Power Mismatch  
Tavg

During the development of the SPDS, several additional parameters were added to the Iconic. The additional parameters include subcooling margin; reactor vessel level indication system; steam jet air ejector radiation; steam generator blowdown radiation; and main steam loop A, B, C and D radiation.

In conclusion, the audit team determined that the wide-range SPDS Iconic does provide a concise display of the minimum five critical safety functions required by Supplement 1 to NUREG-0737. We concluded that the narrow-range Iconic provides a concise display of key parameters during power operations while the wide-range Iconic may be rapidly accessed in order to assess the status of the five critical safety functions during accident conditions. We did not review the basis for parameter selection nor the adequacy of the parameters selected to present the states of the critical safety functions.

## 5.2 Convenient Location

The SPDS is displayed on the control board for Unit 1 and in the same location on the board for Unit 2. Byron management plans to also have it available on the main control room center desk; and the shift supervisor has requested that it also be displayed in his office. From the center desk operations personnel can see the Unit 1 and Unit 2 control board displays. The SPDS is displayed on one of the two CRT screens which are located to the right and left of the center section of the vertical display boards. With the SPDS displays integrated into existing CRT screens, the SPDS appears to be accepted as part of the normal instrumentation by the operators and to receive commensurate respect. We conclude this is a convenient location.

## 5.3 Incorporation of Human Factors Principles

Human factors experts from ARD Corporation have participated in the design of the SPDS system, and will review it as a part of the DCRDR. The following human factors discrepancies were noted by the NRC audit team: The red-on-grey background numbers are difficult to read, and the cyan "reference" octagons are also difficult to see from a distance. The yellow color used to indicate plant status is very clearly seen. The yellow Iconic violates the color-coding conventions by depicting both normal and abnormal conditions. However, given the design of the display, this color coding appeared acceptable. Otherwise, the audit team judged human factors aspects of the display acceptable.

## 5.4 Procedures

The SPDS was not developed with its integration into the Byron EOPs as a goal. It was developed as a quick-look device to assess the overall status of the plant so that action can be taken based on the control room's normal instrumentation. Byron's EOPs do not refer to the SPDS as an action instrument.

## 5.5 Training for Accident Response With and Without SPDS

The audit team reviewed the Byron training plan which includes the training on the SPDS. Since the SPDS is part of the process computer utilizing normal computer data points, instruction in its use is a part of the overall process computer instructional package. The portion devoted to the SPDS discusses its use as a quick-look device to assist in the prevention and mitigation of emergency conditions through the monitoring of the critical safety functions. It then covers the design of the SPDS including the selection of parameters and their association with each of the critical safety functions, the logic of their grouping on the spokes of the Iconic display, and the algorithms used.

Because the SPDS is considered an aid to accident prevention and mitigation and not an "action" instrument, there is little chance that the operators will rely too heavily on it rather than on their class 1E instrumentation; moreover, it does not appear likely that its loss will have a significantly adverse effect on an operator's performance in preventing or mitigating an accident.

The SPDS will be provided in the Byron and Braidwood simulator; training programs covering the operators' reactions to emergency conditions with the SPDS functional and inoperable should be developed.

## 5.6 Parameter Selection

Final approval of the parameter selection of the SPDS will be made by the Procedures and Systems Review Branch (PSRB) at the NRC. The information which follows is supplied to aid PSRB in their review.

As stated above, the basis for the design was the Westinghouse Iconic display, utilizing two top-level pages to cover all operating modes. The narrow-range Iconic covers the normal operating range while the wide-range Iconic covers the remaining modes, including refueling. In order to achieve this broad coverage, two pages were developed for the SPDS. The actual parameters selected are included in Section 5.1 of this report.



In order to examine the criteria for parameter selection, we walked through several hypothetical emergency scenarios with a licensed plant operator who also contributed to the SPDS design. The emergency conditions which were examined were cold water accident--both secondary and primary systems, at power, start-up, and hot stand-by; primary system leak, large and small, during all modes of operation including refueling; loss of off-site power, loss of both off-site and on-site power during all modes of operation; primary containment leak; and, dropping and rupturing a fuel assembly during refueling.

A suggested improvement is the inclusion of containment sump level on the wide-range display. It is present on the narrow-range. If included on the wide-range as well, it would be an important indicator of primary leakage during modes of operation other than full power, such as accident conditions.

#### 5.7 Electrical/Electronic Isolation

We reviewed the documentation on isolation to be provided to the Instrumentation and Control Systems Branch. If the isolation used in the computer is found to be satisfactory, SPDS electrical and electronic isolation should also be satisfactory, since all SPDS signals are obtained from the normal process computer data base.

We reviewed several electrical schematics to determine that SPDS should continue to operate during a total loss of on-site and off-site power. Both the process computer and the SPDS display in the control room are powered from supplies with battery backup.

#### CONCLUSIONS

The NRC audit team verified that the design of the Byron Unit 1 SPDS should meet the requirements of Supplement 1 to NUREG-0737. However, there were several problems noted by the audit team. The corporate verification and validation project for the Byron SPDS had not been fully developed at the time of the audit. Therefore, the corporate verification and validation report should be forwarded to the NRC in order to complete the evaluation of this issue. In terms of human engineering problems, the audit team noted

that there was no Iconic title (wide range/narrow range), the wide-range steam generator level does not function correctly during all modes of operation, and the red alarm bars on grey background have low color contrast.

The above findings should not diminish the fact that the audit team concluded that the displays and the systematic process used to develop the displays should result in an SPDS which meets the NRC requirements. The human factors aspects and user acceptance are both positive aspects of the Byron system. In addition, the audit team has noted the positive support provided by the station management during the development and implementation of the SPDS.

## REFERENCES

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 18.2, Rev. 0, "Safety Parameter Display System (SPDS)," and Appendix A to SRP Section 18.2, "Human Factors Review Guidelines for the Safety Parameter Display System," November 1984.
2. NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," USNRC, Washington, D.C., December 1982, transmitted to reactor licensees via Generic Letter 82-33, December 17, 1982.
3. Letter from E. Douglas Swartz, Nuclear Licensing Administrator, Commonwealth Edison Company to Harold R. Denton, Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Subject: Dresden Station Units 2 and 3, Quad Cities Units 1 and 2, Zion Station Units 1 and 2, Byron Station Units 1 and 2, Braidwood Station Units 1 and 2, NUREG-0737 Supplement 1 SPDS Safety Analysis, Commonwealth Edison Company, Chicago, Illinois, December 29, 1983.
4. Letter from B.J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, U.S. Nuclear Regulatory Commission, to Dennis L. Farrar, Director of Nuclear Licensing Commonwealth Edison Company, Chicago, Illinois, Subject: Request for Additional Information - Byron Braidwood SPDS, U.S. Nuclear Regulatory Commission, November 9, 1984.
5. Memorandum for B.J. Youngblood, Chief, Licensing Branch No. 1, Division of Licensing, from D. Tondi, Acting Chief, Human Factors Engineering Branch, Division of Human Factors Safety, Subject: Staff Audit of Byron 1 and 2 Safety Parameter Display System, U.S. Nuclear Regulatory Commission, February 6, 1985.
6. NUREG-0660, Vol. 1, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC, Washington, D.C., May 1980; Rev. 1, August 1980.
7. NUREG-0737, "Requirements for Emergency Response Capability," USNRC, Washington, D.C., November 1980.

8. NUREG-0700, "Guidelines for Control Room Design Reviews," USNRC, Washington, D.C., September 1981.

ATTACHMENT 1

AUDIT PLAN FOR BYRON 1 AND 2  
SAFETY PARAMETER DISPLAY SYSTEM

AUDIT PLAN  
FOR THE  
BYRON 1 AND 2  
SAFETY PARAMETER DISPLAY SYSTEM

Background

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 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 of the staff's review are to be published in a Safety Evaluation Report (SER).

Commonwealth Edison Company submitted a safety analysis (Ref. 1) for the Byron Units 1 and 2 SPDS. The staff reviewed the safety analysis and concluded that insufficient information was provided to complete our evaluation. Reference 2, a request for additional information was forwarded to the Commonwealth Edison Company. To facilitate the completion of the review, the staff will audit the Byron Units 1 and 2 SPDS.

Audit Schedule

The staff proposes this be scheduled for April 23-25, 1985. We anticipate that the audit will require two full days (April 23-24) of effort. We plan an exit briefing for the morning of April 25, 1985.

NRC Audit Team

The NRC Audit Team will consist of representatives from the Human Factors Engineering Branch, Procedures and Systems Review Branch, and from the Instrumentation and Control Systems Branch. In addition, the staff will be assisted in the audit by Science Applications International Corporation (SAIC).

### Audit Tasks

The audit consists of four sets of tasks that are defined as:

- I. General Issues
- II. Human Factors Engineering Audit

Details on each of these sets of tasks are provided next.

Review Basis: NUREG-0737, Supplement 1, "Clarification of TMI Action Plan, Requirements for Emergency Response Capability."

#### I. General Issues

<u>Topics</u>	<u>Audit Needs:</u>	<u>Estimated Time (Hours)</u>
1. An entry briefing by the NRC audit team to discuss schedule and audit plan.	A conference room or equivalent to hold briefing	0.25
2. Staff caucus to discuss results of audit.	A conference room or equivalent.	2
3. An exit briefing by the NRC audit team to report on the findings of the audit.	A conference room or equivalent to hold briefing.	0.5
4. Commonwealth Edison is to define the scope of the SPDS within the computer system in which it is implemented and in terms of the SPDS as stated in NUREG-0737, Supplement 1.	Have available all elements of the design as it currently exists consisting of hardware, software and display formats.	0.5
5. Staff audit of the Design Verification and Validation Program used in the development of the SPDS.	Have available the Design Verification and Validation Program. Also, on a part-time basis, have available a qualified person capable of answering staff questions on the program.	1.5



<u>Topics</u>	<u>Audit Needs:</u>	<u>Estimated Time (Hours)</u>
II. Human Factors Engineering Audit		
1. Staff audit of the System Specifications and standards used in the design, such as human factors engineering standards.	System Specifications, generic application data and standards used in the design. Also, on a part-time basis, have available personnel capable of answering questions on the specifications and standards. The licensee should also be prepared to discuss details of the Human Factors Program used in the design along with data validation techniques and the means used to inform the operator of invalid data.	2
2. Staff audit of the validation of the display formats utilizing man-in-the-loop tests of a prototype display, if applicable.	The validation program and the results from the program. Also have available personnel capable of answering staff questions on the validation program and the results from the program.	2
3. Staff audit of the software specifications for incorporation of human factors requirements.	The generic software requirements, the generic spec., and human factors standards used in the design. Also have available personnel capable of answering questions on these documents.	1.5
4. Staff audit of the design, code, test software and data base instructions (if applicable).	Design documentation, listing of code, and description of data base. Also have available personnel capable of answering questions on these documents.	1.5
5. Staff audit of integration tests and test results for displays and scenarios, applicable.	Documents and test plans for integration tests along with test results. Also have available personnel to answer questions on these documents.	2

<u>Topics</u>	<u>Audit Needs:</u>	<u>Estimated Time (Hours)</u>
6. Staff audit of selected display formats for conformance to human engineering standards and guidelines. Evaluate if display flicker exists; also determine adequacy of time lag for display of data.	Selected display formats on prototype display system, if available. As a minimum, a hard copy, in color, of selected display formats will suffice.	2.5
7. Staff audit of display devices, display controls, and keyboards, etc. for conformance to human engineering standards and guidelines.	Have available display devices, display controls and keyboards, etc. Also have available personnel to answer staff questions on these devices.	2
8. Staff audit of design validation test methods, and test plans.	Documents on test methods and test plans, if available. If documents are not available, provide a discussion on validation testing.	1

#### REFERENCES

1. Letter from E. Swartz, Commonwealth Edison Company, to H. R. Denton, NRC, subject: "Byron Station Units 1 and 2 SPDS Safety Analysis," dated December 29, 1983.
2. Letter from B. J. Youngblood, NRC, to D. L. Farrar, Commonwealth Edison Company, subject: "Request for Additional Information - Byron/Braidwood SPDS," dated November 9, 1984.

ATTACHMENT 2

LIST OF AUDIT MEETING ATTENDEES

Byron - SPDS Audit  
Entrance 1330

Ken Ainger  
Leo Beltracchi  
Don Brindle  
Jeffrey Colborn  
Joseph DeBor  
Whit Hansen  
Julian Hinds  
Laurence Huetteman  
Bob Kershner  
Tim Melloch  
Christopher Olmsted  
Bob Querio  
Richard Stark  
Dale St. Clair  
Tom Weis

CECo - Nuclear Licensing  
HFEB/NRC  
Operating - Byron  
Tech. Staff/CECo  
SAIC/NRC  
Comex/NRC  
Sr. Resident Inspector, USNRC  
Comp Sys/CECo  
HFE/ARD Corp  
CECo - Byron Station  
Comp Sys/CECo  
Sta. Supt. - Byron  
SAIC/NRC  
Tech Staff Superv. Byron  
PED

ATTACHMENT 3

HUMAN FACTORS SPDS REVIEW SCHEDULES

DRESDEN/QUAD CITIES  
HUMAN FACTORS SPDS SUPPORT

	5/15	6/1	6/15	6/30	7/15	7/30	8/15
SPDS DESIGN REVIEW	△						
SPDS SPECIFIC OPERATOR SURVEY COMPILE HEDS		△	△				
APPLY HF CHECKLIST TO SPDS COMPILE HEDS		△	△	△			
DCRDR TASK ANALYSIS DATA ANALYSIS COMPILE HEDS			△	△	△		
ASSESS HEDS					△	△	
HED RESPONSES						△	△



ATTACHMENT 4

COMMONWEALTH EDISON VERIFICATION AND VALIDATION SCHEDULE

COMMONWEALTH EDISON COMPANY  
SPDS REQUIREMENTS REVIEW REPORT

Figure 3.4

II. SPDS V&V REVIEW SCHEDULE

Complete Review of (Responsible Department)	Control Point	Completion Date					
		Dresden	Quad Cities	LaSalle	Zion	Byron	Braidwood
a. System Requirements (SNED)	A	10/31/84	10/31/84	10/31/84	10/31/84	10/31/84	10/31/84
b. System Design (SNED)	B	11/30/84	11/30/84	11/30/84	11/30/84	11/30/84	11/30/84
c. Software Design (COMP. SYSTEMS)	C	11/1/85	3/1/85	4/1/84	5/1/84	6/1/85	6/1/86
d. Software Development & Testing (COMP. SYSTEMS)	D	1/1/85	5/1/85	6/1/85	10/1/85	8/1/85	8/1/86
e. Hardware Design (COMP. SYSTEMS)	E	12/1/85	4/1/85	5/1/85	6/1/85	7/1/85	7/1/86
f. Hardware Installation & Testing (SYSTEM OAD)	F	2/1/85	3/1/85	4/1/85	5/1/85	6/1/85	6/1/86
g. Preoperational Test, (STATION)	G	7/1/85	7/1/85	1/1/86	7/1/86	2/1/87	2/1/87
h. V&V REPORT (SNED)	H	1/1/86	1/1/86	7/1/86	1/1/87	8/1/87	8/1/87

