

A HISTORY AND OVERVIEW OF THE SAFETY PARAMETER DISPLAY SYSTEM CONCEPT

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Summary

Inquiries into the accident at the Three Mile Island Nuclear Power Plant Unit 2, on March 28, 1979 brought to public attention the need to improve operators' capabilities to interact with the systems under their control. Recommendations ran the full gamut of human/machine interaction, from improvements in training and procedures to improvements in control and display hardware in the control room. This presentation briefly traces the history and development of a display concept that evolved in the post-TMI era, the Safety Parameter Display System or SPDS. The SPDS is intended to function as a detection aid for control room operators, providing an integrated overview of significant plant parameters. The purpose of this report is to describe the general concept of SPDS, its history, and its current regulatory status. A review of NRC guidance documents is included, as well as a discussion of NRC requirements placed on the SPDS. The presentation concludes with an outline of the NRC staff review process for safety parameter display systems and a synopsis of the results of generic SPDS reviews performed thus far.

Introduction

In the months following the accident at Three Mile Island, various inquiry groups investigated the problems that caused or exacerbated the sequence of events that resulted in the partial core meltdown of Unit 2. A major problem discovered by most investigators was that no one in the TMI control room appeared to be able to assemble and integrate the correct combination of symptoms that would allow an early recognition of the fact that the critical safety functions of the plant had been compromised, that is, that the core was being inadequately cooled. It was recognized that an overview of critical safety functions was necessary. In response to this need, the NRC staff proposed what is now called the "Safety Parameter Display System" or SPDS.

The Accident At Three Mile Island

To return to the events of March 28, 1979 at Three Mile Island, at about 4:00 in the morning a problem with the condensate polishers caused a feedwater pump to trip. This was followed by a turbine trip/reactor trip. The partial loss of heat sink caused the after-heat to raise primary temperature and pressure. The increased pressure caused the power-operated relief valve (the PORV) on the pressurizer to open. Thus far everything had gone as it should. Control rods were in, pressure was coming down; however, the PORV failed to close when pressure returned below the setpoint. The system was depressurizing, the heat sink had been partially lost. Primary inventory continued to relieve from the open PORV. This event went unrecognized for two hours. The operators finally discovered that they had been losing coolant through the PORV but were led to believe that the primary side was "solid" and the core had remained covered throughout the incident. This, of course, was not true. The primary side had reached saturation, much of the coolant covering the core had boiled away, and steam voids and non-condensable gases were keeping the pressurizer "solid"--giving operators the impression that the reactor vessel was

also full of coolant. The conditions in the vessel and the core went unrecognized for another fourteen hours. Finally, when the possibility of fuel failure was recognized, high pressure injection was initiated and repressurized operations began. Non-condensable gases were bled out of the system during the next few days and preparations for depressurization were begun.

The Lessons Learned Task Force Review of TMI

Investigations were initiated soon after the accident occurred. An NRC review was initiated in May, 1979. This review was performed by a team of NRC engineers and scientists called the "Lessons Learned Task Force." Their goal was to identify safety issues involved in the TMI accident and to make short-term and long-term recommendations.

During their extensive review of the accident, the members of the task force became more and more convinced that, regardless of many of the other shortcomings, there was one overriding and very disturbing issue that needed resolution: for 16 hours no one in the control room--operators, shift supervisors, station manager, engineering and operations personnel, and NRC resident inspector--recognized that the critical safety functions of the plant were not being served, that the flow through the core was insufficient, and that fuel failure had occurred. No one had a good overview of the behavior of the plant. Other investigative groups later reached the same conclusion.

It seemed that although the necessary information was, in general, physically available, it was not operationally effective. No one could assemble the separate bits of information to make the correct deductions. Since this failure applied to everyone in the control room, it appeared that there had to be a common causal factor. Looking more extensively into the operators' information-processing strategies, the staff reasoned that the assessment of plant conditions necessitated (1) a mental model of the plant processes, which could provide the basis for identifying the information that should be gathered in order to assess plant health; (2) gathering information from dispersed areas of the control room; (3) remembering that gathered information so that comparisons can be made and interrelationships determined; and (4) integrating all this information into the original mental model of the plant. The most important point of this rationale was the need for a mental model. A good model provides both a guide for collecting important data and a framework into which the data can be integrated to give the operator an overview of system behavior. At TMI, as elsewhere, no explicit models or other pattern-recognition aids were formally used. Then, as now, each operator used his own unique model of plant processes to derive his specific information-gathering and processing strategy. Normally, this causes no problem, but under conditions of stress, such as at TMI, operator models of plant behavior may turn out to be overly complex or incomplete and, therefore, useless and inappropriate.

When preconceived notions about plant behavior do not correspond to actual plant conditions, several things may happen. First, operators may tend to repeat their original, inappropriate information-gathering strategy. Second, in order to try to make actual conditions fit their preconceived notions, people often selectively disbelieve or disregard anomalous information. Third, when it becomes obvious that the situation does not fit their mental model, they regress to less effective forms of information gathering, for example, attending to all information regardless of its importance--looking for any clue at all that may be helpful. Information overload usually results, further degrading the reasoning process. Suboptimal strategies such as information queuing, the dropping out of information, and cognitive fixation are the common under such conditions of stress and overload. Of course, this description is vastly oversimplified.

There are many other contributing factors. The operators at TMI did not blatantly disregard important facts. The hardware made it easy for them to disbelieve or disregard information--temperature downstream of the PORV was a traditionally unreliable indication of flow, incore temperatures were off-scale and had to be jury-rigged to get a wider range read-out. The method itself was suspect, and the results were inconsistent and very easy to disbelieve. In the face of a myriad of confusing facts, operators responded predictably, ignoring suspect information in favor of traditionally reliable information like pressurizer level.

The Concept Is Proposed--NUREG-0585

At this point, the staff felt that they had successfully focussed on the major problem: relevant information about plant status was dispersed, lacked immediacy and reliability, and could not be easily integrated into any meaningful model. The solution, first proposed in NUREG-0585, was to gather together a minimum set of plant parameters that would be descriptive of plant processes. Lacking a commonly acceptable mental model of plant processes, the staff proposed to symbolize plant processes in what was then called a "safety stat vector." The concept was based on the perceived need of the operators to have a simple, integrated, robust measure of plant health that was concisely displayed and easily understood. Such a display would, by design, gather the most important, safety-critical information for the operator. Vagaries of inappropriate modeling or losses from short-term memory would be minimized. The degree to which the critical safety functions were being satisfied would be immediately and continuously available.

SPDS--An Action Plan Item

In May, 1980, a "plant safety parameter display console" was included as part of the NRC Action Plan, NUREG-0660. Essentially, the concept remained unchanged. However, the description now more clearly implied by the word "console" that the SPDS would be serving an independent function, a function that could not be served by monitoring scattered meters and dials throughout the control room.

The Concept Is Elaborated--NUREG-0696

In February, 1981, a more extensive description of the SPDS concept was published as a section of NUREG-0696, "Functional Criteria for Emergency Response Facilities." By this time many owners' groups, industry groups, and vendors were well on the way to specifying design requirements for their own safety para-

meter display system. Much information had been exchanged prior to the publication of NUREG-0696, and this give-and-take exchange resulted in a much clearer definition of the SPDS concept. Section 5 of NUREG-0696 represents the first complete description of the SPDS function and its relationship to the control room and other emergency response facilities. The functional definition is concisely stated as follows: "The primary function of the SPDS is to aid the operator in the rapid detection of abnormal operating conditions." In this statement, the function is explicitly spelled out as aiding detection rather than diagnosis. However, the text goes on to suggest that secondary function such as diagnostic aids may be desirable and could be included in SPDS designs.

Once again, the concept of simplification and concentration of data is reiterated in the statement. "As an operator aid, the SPDS serves to concentrate a minimum set of plant parameters from which plant safety status can be assessed." This basic concept is further elaborated in another important statement, "The SPDS is to provide a continuous indication of plant parameters or derived variables representative of the safety status of the plant." Implicit in this statement is the idea that the SPDS is not to be a glorified annunciator, but rather a collection of continuously displayed leading indicators of plant process behavior. This point is further clarified in the statement that the SPDS shall be in operation during normal and abnormal operating conditions," and that the SPDS shall be capable of displaying pertinent information during steady-state and transient conditions."

Also in this section is the first formal reference to pattern recognition as a detection aid. "The design of the primary or principal display format shall be as simple as possible...and shall include pattern and coding techniques to assist the operator's memory recall for the detection and recognition of unsafe operating conditions."

NUREG-0696 also discusses the reliability issue. Since perceived instrument reliability (rather than actual reliability) was a major factor at TMI, the staff felt that high standards should be set so that operators would have confidence in the SPDS-displayed information. It was originally envisioned that the SPDS should be Class 1E, seismically qualified. Since many of the proposed designs at that time were CRT/microprocessor-based and were not easily amenable to such strict qualifications, it was suggested in NUREG-0696 that a seismically qualified backup would be sufficient, provided that the primary SPDS met an unavailability goal of .01 for nonseismic conditions. Later, this issue will be more realistically addressed in the final requirements, in that prompt implementation will take precedence over the need for seismically qualified support instrumentation.

Acceptance Criteria Are Drafted

The next, and final, significant guidance document concerning SPDS was the draft version of NUREG-0835, "Human Factors Acceptance Criteria for the Safety Parameter Display System," published in October, 1981. The basic concept is reiterated in that document, but with special emphasis on CRT applications. Choice of parameters and proper formatting are discussed. The need for trend and rate of change data is highlighted. As stated in Section 3.2, "Operator monitoring of parameter trends is a key task in evaluating the safety status of the plant. This trend information is needed to assist the operators in determining the severity of an abnormality when a transient condition develops."

NUREG-0835 suggests that parameters or derived variables and their display ranges be chosen so that, even during slowly evolving transients, the operator will be able to detect abnormalities, but will not be unduly plagued by "false alarms." The explicit statement in NUREG-0835 is that "Displays should use appropriate parameters that have small deviations about a steady state value during normal operating conditions and that have distinctive large variations from the steady state value during abnormal conditions."

In other parts of the document, human engineering issues (such as readability, labeling, color-coding, and so forth) are discussed. An appendix discusses and gives examples of acceptable and unacceptable display patterns. The document does not attempt to address any non-human factors issues, such as signal isolation or the design basis for the choice of parameters.

Flexibility of Final Requirements

With regard to the choice of parameters, it was recognized throughout the history of the SPDS that the choice of parameters would be best left to the discretion of the industry. It was felt that flexibility was necessary to assure that any plant-specific or reactor-type-specific characteristics would be appropriately represented. The industry, especially the Electric Power Research Institute, the Nuclear Safety Analysis Center, and various owners' groups, has made significant progress in this area. The NRC staff has attempted to encourage innovation in the SPDS concept by stepping back a little from its normally stringent and proscriptive stance in other regulatory areas as well.

It is hoped that this flexibility will allow for both innovative designs and speedier implementation. The Commission considers the SPDS a feasible concept and one that can provide an important contribution to plant safety.

Current Requirements

On July 16, 1982, the Commission approved the issuance of a supplement to NUREG-0737, "Clarification of TMI Action Plan Requirements," on the subject of emergency response capabilities at nuclear power stations. This new supplement would apply to all operating nuclear power plants and to all applicants for operating licenses.

The Commission considered the staff recommendations and approved the requirements recommended in "NRC Staff Recommendations on the Requirements for Emergency Response Capability" as appropriately clarifying and providing greater detail with respect to the Safety Parameter Display System (SPDS). Included in the supplement would be a request, pursuant to 10 CFR 50.54(f), that all operating reactor licensees and holders of construction permits furnish, a proposed schedule for completing the basic requirements for the SPDS. In addition, licensees would be requested to submit a description of their plans for phased implementation and integration of the emergency response activities. The plans for integration will be reviewed as part of the staff's evaluation of the proposed schedule. After the staff completes its evaluation, it will take action, as necessary, to assure that such requirements and commitments are appropriately enforceable. NRC will make allowance for work already done by licensees in a good-faith effort to meet requirements as they understood them.

Basic SPDS Requirements

1. Each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the control room operators. This system will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events.
2. The control room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10 CFR 50) forms the basic safety components required for safe reactor operation under normal, transient, and accident conditions. The SPDS is used in addition to the basic components and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation (e.g., GDC 2, 3, 4 in Appendix A; 10 CFR Part 100; single-failure requirements). The SPDS need not meet requirements of the single-failure criterion and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. Procedures that describe the timely and correct safety status assessment when the SPDS is and is not available will be developed in parallel with the SPDS. Furthermore, operators should be trained to respond to accident conditions both with and without the SPDS available.
3. There is a wide range of useful information that can be provided by various systems. This information is reflected in staff documents such as NUREG-0696, NUREG-0835, and Regulatory Guide 1.97. Prompt implementation of an SPDS can provide an important contribution to plant safety. The selection of specific information that should be provided for a particular plant shall be based on engineering judgment of individual plant licensees, taking into account the importance of prompt implementation.
4. The SPDS display shall be designed to incorporate accepted human factors principles so that the displayed information can be readily perceived and comprehended by SPDS users.
5. The minimum information to be provided shall be sufficient to provide information to plant operators about:
 - a. Reactivity control
 - b. Reactor core cooling and heat removal from the primary system
 - c. Reactor coolant system integrity
 - d. Radioactivity control
 - e. Containment conditions

The specific parameters to be displayed shall be determined by the licensee.

NRC Review Process

The design of the SPDS is to be integrated with other related control room activities, such as the development of symptom-oriented emergency procedures, and control room design review. As part of this integration, the licensee shall 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. The licensee's safety analysis for the proposed SPDS shall be reviewed in accordance with the Commission's Regulation 10 CFR 50.59, to determine whether the changes involve an unreviewed safety question or changes of technical specifications. If they do, they shall be processed in the normal fashion with prior NRC review. If the changes do not involve an unreviewed safety question or a change in the technical specifications, the licensee may implement such changes without prior approval by NRC, or the licensee may request a pre-implementation review and approval. A pre-implementation review should be initiated with an early submittal of the SPDS implementation plan and a request for review.

For implementation of the SPDS without prior NRC approval, the staff will conduct a post-implementation review. Licensees are to prepare a written safety analysis describing the basis on which the selected variables are sufficient for the operators to assess the safety status of the plant. The licensees will also prepare a specific implementation plan for the SPDS. The safety analysis and the implementation plan are submitted to NRC for review promptly on completion of review by the licensee's offsite safety review committee. Basically, the NRC plans to conduct a one-step review of the SPDS as a post-implementation verification review. Based on the results of NRC review, the Director of the Office of Inspection and Enforcement or the Director of the Office of Nuclear Reactor Regulation may request or direct the licensee to cease implementation if a serious safety question is posed by the licensee's proposed system or if the licensee's analysis is seriously inadequate.

Staff Review

The NRC review of the SPDS will be accomplished by an interdisciplinary team drawn from the staff of the Division of Human Factors Safety and the Division of Systems Integration. The Human Factors Engineering Branch will have lead responsibility. The review team will consist of members from the Human Factors Engineering Branch (HFEB), Instrumentation and Control Systems Branch (ICSB), and the Reactor Systems Branch (RSB), with support from the Core Performance Branch (CPB), Procedures and Test Review Branch (PTRB), and other functional disciplines as needed. In general, the functions are allocated to each discipline as follows:

- 1.) HFEB--operator/machine interface, including formatting and pattern recognition, and application software

- 2.) ICSB--hardware, including reliability and interfaces with protection systems
- 3.) RSB--parameter selection
- 4.) CPB--thermohydraulic algorithm and inventory monitor
- 5.) PTRB--integration of SPDS with Emergency Procedure Guidelines

Field meetings for generic activities will be coordinated with the Region IV Vendor Inspection Program. All in-plant field audits will be scheduled with and attended by representatives of the Commission's Regional Offices.

Pre-Implementation Reviews

It is essential that reactor operators be provided with accurate, reliable information from the SPDS and that the operators have confidence in the validity of the information provided. Staff review emphasis will be focused on the means employed by licensees to insure that information displayed to the operators accurately reflects the as-built operating plant. Furthermore, in order to gain operator confidence, it is important that the system design, fabrication, and installation be verified and validated (V&V).

The V&V program will help ensure that a quality SPDS system is implemented. NSAC/39, "Verification and Validation for Safety Parameter Display Systems," December 1981, describes a typical V&V program which has been developed by industry and is acceptable to the NRC staff.

A V&V program should consist of design reviews, development reviews, and testing conducted by qualified individuals other than the designers and developers of the equipment and software. The reviews and tests strive for a high quality product by identifying and correcting problems throughout design, fabrication, and installation. The V&V program addresses both hardware and software aspects of the equipment. Industry people rather than NRC staff will perform the V&V program.

The NRC staff pre-implementation review of the SPDS will be an evaluation of the licensee's proposed V&V program. The staff will evaluate the effectiveness of the proposed V&V program by assessing how problems will be identified, the types of problems to be identified, and the methods of developing and documenting problem resolution. This approach minimizes the burden on both the industry and the staff by limiting the staff's role to that of an auditor of the V&V program rather than involving the staff in the conduct of the V&V program.

The major steps of a staff review of the SPDS are:

1. Evaluate the program plan for the V&V of the SPDS design and confirm that Human Factors Engineering principles are to be followed.
2. Audit the results of the design verification task conducted by the licensee/designer for conformance to the V&V program plan.
3. Audit the as-built display for operational conformance to the design specifications.
4. Audit the control-room-installed system to verify that the correct relationship between sensor and displayed variable exists, and that interface requirements with Class 1E systems will be implemented.

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If a licensee requests a pre-implementation review, the staff desires to complete it in three meetings to accomplish the four steps listed above. NRC staff recommends that licensees and vendors define the contents of each meeting in terms of these steps. Licensees should also propose a schedule for the meetings. The staff will review the proposed schedules, plan its resources for the meetings, and propose schedule modifications if needed. It is anticipated that the bulk of the review will be accomplished at these meetings. The SPDS that are well under way, or completed, and with which the staff is already familiar, will receive consolidated reviews and may not require the three meeting review process.

Post-Implementation Reviews

The installation in the control room of an unreliable, or poorly designed and untested SPDS represents a potential unreviewed safety question. For example: inaccurate data may be misleading to the operator, or an installed system that requires extensive post installation troubleshooting and debugging will degrade ultimate operator confidence (can lead to operators ignoring the system when it is finally operational). Licensees electing post-implementation reviews should conduct V&V programs and human factors engineering programs prior to design, fabrication, and use of the SPDS. These programs will be reviewed by the NRC as part of its post-implementation review.

Licensees should also prepare a written description of the basis on which the variables selected for the SPDS are sufficient for the operator to assess the safety status of the plant. The description will be reviewed by the NRC staff in a design review meeting. The review will also include an audit of the installed system (same as item 4 of the pre-implementation review steps above). Post-implementation reviews will be results oriented, in contrast with pre-implementation reviews which are to be oriented to the V&V program and the human factors engineering program proposed for use by the licensee.

Schedule

The staff will prioritize its review resources among those licensees requesting pre-implementation reviews based upon:

1. Firm procurement commitments of licensees, which are conditional on NRC staff review.
2. Generic review of vendor and owner group efforts. Where there is sufficient commonality between generic and plant-specific designs and design methodology, plant-specific reviews may be minimal. Simple confirmation of participation in a generically approved design effort may suffice.
3. Plant-unique design features.

Staff review of licensees electing pre- or post-implementation review will be conducted on mutually agreed schedules.

Staff Documentation

The staff will write letters following each meeting between licensee and staff to advise the licensee of positive and negative findings. Negative findings may be resolved in supplementary meetings as mutually agreed upon by licensee and staff. Other than these meeting reports, the staff will not prepare a summary safety evaluation report until completion of its audit of the SPDS after it is installed and operational in the control room.

Review Status

As of October 20, 1982, the NRC has received one request for a pre-implementation review and approval of an SPDS design. At this time the staff has had no request for a post-implementation review. As for the pre-implementation review, the staff reviewed the proposed V&V program and found the basis of the program acceptable for the design and development of an SPDS. As stated earlier in this paper, an important function in the development, fabrication, and installation of an SPDS design is the V&V program. The staff believes that a good V&V program will ensure that a quality SPDS will be implemented. Based on the acceptance of the V&V program the NRC will proceed to set up an interdisciplinary review/audit team from within NRR. The objective of the review team is to audit the conformance of the design to the V&V program and resolve topics within three separate meetings. The following is an example of the type of technical topics that may be discussed during an audit.

1. Application software associated with data and information management; software psychology (ease of use, simplicity in learning, improved reliability, minimizes human error frequency and enhances user satisfaction), response to needs of user.
2. Data and information
 - (a) identify specific needs (parameters, variables)
 - (b) storage and recall capability
3. Presentation and displays
 - (a) location in control room
 - (b) operator interaction capability, readability, information density
 - (c) formats - what and how to present
 - (d) paging capability and access to data
4. The relationship and consistency of SPDS with normal and emergency procedures
5. Reliability and availability of computer system
6. Isolation of safety signals:
 - (a) The adequacy of and the basis of the parameters selected by the licensee including the need for recall and trending capability of specific parameters.
 - (b) Audit the design to ensure that information displayed to the operator is validated for each critical safety function.
 - (c) Audit the design for consistency of the SPDS with emergency procedure.

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