SECTION 18

HUMAN FACTORS ENGINEERING

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18.1 DETAILED CONTROL ROOM DESIGN REVIEW

Item I.D.1, "Control Room Design Reviews," of Task I.D, "Control Room Design," of the NRC Action Plan (NUREG-0660) developed as a result of the accident at Three Mile Island (TMI), Unit 2, states that licensees and applicants for operating licenses will be required to perform detailed control room design review (DCRDR) to identify and correct design discrepancies. The objective, as stated in NUREG-0660, is to improve the ability of nuclear power plant control room operators to prevent or cope with accidents if they occur by improving the information provided to them. Supplement 1 of NUREG-0737 confirmed and clarified the DCRDR requirement in NUREG-0660. As a result of Supplement 1 to NUREG-0737, each applicant or licensee is required to conduct a DCRDR.

NUREG-0700, "Guidelines For Control Room Design Reviews," describes four phases of the DCRDR to be performed by the applicant or licensee. The phases are

1. Planning
2. Review
3. Assessment and implementation
4. Reporting

Criteria for evaluating each phase are contained in Section 18.1 and Appendix A to Section 18.1 of the Standard Review Plan (NUREG-0800).

Supplement 1 to NUREG-0737 requires each applicant or licensee to submit a program plan that describes how the following elements of the DCRDR will be accomplished:

1. Establishment of a qualified multidisciplinary review team.

2. Function and task analysis to identify control room operator tasks and information and control requirements during emergency operations,

3. A comparison of display and control requirements with a control room inventory.

4. A control room survey to identify deviations from accepted human factors principles.

5. Assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected.

6. Selection of design improvements.

7. Verification that selected design improvements will provide the necessary correction.

8. Verification that improvements will not introduce new HEDs.

9. Coordination of control room improvements with changes from other programs such as the safety parameter display system, operator training, Regulatory Guide 1.97 instrumentation, and upgrade of emergency operating procedures.

Supplement 1 to NUREG-0737 also requires each applicant or licensee to submit a summary report at the end of the DCRDR. The report should describe the proposed control room changes and implementation schedules and provide justification for leaving safety significant HEDs uncorrected or partially corrected.

As required by Supplement 1 to NUREG-0737, PSE&G submitted the DCRDR program plan for Hope Creek (see Reference 18.1-1). PSE&G subsequently submitted a description of the DCRDR function and task analysis methodology (see Reference 18.1-2) and the DCRDR Summary

Report (see Reference 18.1-3).

Information regarding technical evaluations, conclusions, and recommendations to satisfy the requirements of Supplement 1 to NUREG-0737 for a DCRDR of the Hope Creek Generating System is included in References 18.1-4 and 18.1-5.

PSE&G submitted Control Room Design Review Supplemental Report I (see Reference 18.1-6) which includes a description of changes made to the DCRDR management and staffing; a description of the methodology used in the task analysis upgrade process and the verification of component characteristics; a discussion of the survey of the plant computer system; an amendment to the solutions proposed for two human engineering discrepancies (HEDs) (A69 and A140); identification of HEDs discovered since the DCRDR Summary Report was submitted by letter dated August 14, 1984. This supplemental report and a Hope Creek license condition to submit two additional supplemental reports and to provide temporary zone markings on safety-related instruments in the control room, satisfy the requirement of Supplement 1 to NUREG-0737 for a DCRDR of the Hope Creek Generating Station with the effort completed to date and in progress, and the operator machine interfaces of the control room are adequate to support safe operation.

Installation of temporary zone markings on safety-related instruments in the Hope Creek control room is complete (see Reference 18.1-7).

PSE&G submitted CRDR Supplemental Report II (see Reference 18.1-8) which includes (1) findings for operator interviews, (2) results of ambient noise survey, (3) heating, ventilation, and air conditioning survey results, (4) illumination survey results, (5) emergency equipment survey results, (6) communication survey results, (7) results of the review of back panel 10C604 (radiological monitoring system), and (8) results of the review of implemented design change package modifications. These items were identified in CRDR Supplemental Report I as requiring further attention.

A third supplemental CRDR summary report will be submitted one year after fuel load to provide the results of a study to be performed on zone marking needs.

18.1.1 References

18.1-1 R.L. Mittl, PSE&G, to A. Schwencer, NRC, "Generic Letter 82-83; Control Room Design Review Program Plan for Hope Creek Generating Station," dated October 17, 1983. (Enclosure: Hope Creek Control Room Design Review Program Plan, dated October 14, 1983).

18.1-2 R.L. Mittl, PSE&G, to A. Schwencer, NRC, "Control Room Design Review Program Plan," dated April 10, 1984.

18.1-3 R.L. Mittl, PSE&G, to A. Schwencer, NRC, "Control Room Design Review Summary Report," dated August 14, 1984.

18.1-4 R.L. Mittl, PSE&G, to A. Schwencer, NRC, "CRDR Pre-Implementation Audit Commitments," dated December 6, 1984.

18.1-5 NUREG-1048, HCGS SSER No. 1, Appendix I, "Technical Evaluation Report of the Detailed Control Room Design Review for PSE&G Hope Creek Generating Station," March 1985.

18.1-6 C. McNeill, PSE&G, to E. Adensam, NRC, "Control Room Design Review Supplemental Report I," dated January 9, 1986.

18.1-7 C. McNeill, PSE&G, to E. Adensam, NRC, "Detailed Control Room Design Review," dated July 28, 1986.

18.1-8 C. McNeill, PSE&G, to E. Adensam, NRC, "Control Room Design Review Supplemental Report II," dated November 12, 1986.

18.2 SAFETY PARAMETER DISPLAY SYSTEM

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 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.

Documentation regarding PSE&G's compliance with NUREG-0737, Supplement 1, will be located in the CRIDS Critical Software Package.

18.2.1 Description

The Hope Creek SPDS provides data acquisition and display as part of the CRIDS system. Operator access to displays is provided at SPDS terminals located in the main control room and the Shift Supervisor's office. Terminals also are provided in the Technical Support Center (TSC), the Hope Creek Control Point, the Salem Control Point, the Salem TSC, the Hope Creek OSC, the Salem OPS Ready Room, and the Emergency Operations Facility.

The capability for continuous monitoring of plant safety status is provided in the form of a control function parameter matrix (CFPM) displayed at the top of each SPDS display page (except the overview page which already includes the actual values of the parameters in the CFPM).

18.2.2 Parameter Selection

Section 4.1(f) of Supplement 1 to NUREG-0737 states:

> The minimum information to be provided shall be sufficient to provide information to plant operators about:
>
> 1. Reactivity control
> 2. Reactor core cooling and heat removal from the primary system
> 3. Reactor coolant system integrity
> 4. Radioactivity control
> 5. Containment conditions.

The Hope Creek SPDS is based on the safety analysis provided in the letter dated April 10, 1985 (R. L. Mittl, PSE&G, to A. Schwencer, NRC) with consideration given to the BWR generic emergency procedure guidelines. The variables and their relationship to the critical safety functions are summarized below.

1. reactivity control

   a. reactor power (average power range monitor (APRM))
   b. reactor pressure vessel (RPV) pressure
   c. RPV water level

2. reactor core cooling and heat removal from primary system

   a. RPV water level
   b. RPV pressure
   c. reactor power (APRM)

3. reactor coolant system integrity

   a. drywell pressure
   b. drywell temperature
   c. suppression pool water temperature
   d. suppression pool water level
   e. suppression chamber air temperature
   f. suppression chamber pressure
   g. RPV water level
   h. RPV pressure

4. radioactivity control

   a. reactor building area radiation
   b. offsite radioactivity release rate

5. containment conditions

   a. drywell pressure
   b. drywell temperature
   c. reactor building area temperatures
   d. reactor building radiation levels
   e. suppression pool water level
   f. suppression pool water temperature
   g. secondary containment area water level
   h. secondary containment sump water level

In addition, PSE&G will add the following four parameters to the Hope Creek SPDS and have them operational by July 1988. See Reference 18.2-4.

1. primary containment radiation
2. primary containment isolation status
3. combustible gas concentration in the primary containment
4. source range neutron flux

The system performance validation for Hope Creek includes a simulator evaluations using transients selected to exercise the emergency operating procedures in coordination with the SPDS displays. Since the simulator is generally limited to design basis events, PSE&G has made provisions to cover beyond design basis conditions (e.g., primary containment venting at elevated pressures, liquid level measurements below the top of the active fuel, and conditions of high radiation in the Reactor Building).

18.2.3 Display Data Validation

The method of data validation currently used in the Hope Creek SPDS is range/status checking, that is, checking that the indicated value is within instrument range and has not failed. Where data are available from more than one sensor, the on-scale values are averaged. Invalid data are displayed with a "BAD" quality indication. Reactor pressure vessel (RPV) level is further validated by having the fuel zone instrument readings included in the average only when both recirculation pumps are off.

18.2.4 Human Factors Program

PSE&G has incorporated good human engineering principles into the Hope Creek SPDS design at several points in the design process. Initially, the design was conceptualized on the basis of a comprehensive task analysis. Design features needed to support operator decisions and actions were specifically documented in PSE&G Design Memoranda CDM 483 and 484. In addition, some human factors input was derived from coordination with the detailed control room design review (DCRDR) and through the independent review of a human factors consultant.

18.2.5 Electrical and Electronic Isolation

The SPDS must be isolated from equipment and sensors that are used in safety systems to prevent an SPDS fault from degrading safety systems. PSE&G has provided information and test data on all of the different types of isolators used at Hope Creek. See Reference 18.2-5. The devices are qualified isolators and are acceptable for interfacing the SPDS with safety systems and therefore meet the requirements in NUREG-0737, Supplement 1.

18.2.6 References

18.2-1 R.L. Mittl, PSE&G, to A. Schwencer, NRC, "Safety Parameter Display System," dated April 10, 1985, and R.L. Mittl, PSE&G, to W. Butler, NRC, "Safety Parameter Display System," dated July 31, and October 14, 1985.

18.2-2 E. Adensam, NRC, to C. McNeill, PSE&G, "Safety Parameter Display System," dated March 18, 1986.

18.2-3 NUREG-1048, HCGS SSER No. 5, Section 18.2, "NRC Safety Evaluation Report for the Safety Parameter Display System," April 1986.

18.2-4 C.A. McNeill, PSE&G, to E. Adensam, NRC, "Four Addition Parameters to the SPDS," dated March 7, 1986.

18.2-5 R.L. Mittl, PSE&G, to W. Butler, NRC, "SPDS Isolation Devices," dated July 31 and November 22, 1985.

18.3 TASK ANALYSIS

A formal task analysis based on existing guidelines will be conducted. The results of such an analysis will be provided in an amendment to the FSAR. The arrangements of systems and controls on operator interface panels C650 and C651 were made as a result of an operability analysis in the early design stages of the Hope Creek Generating Station (HCGS) control room and panels.

18.4 THE MAIN CONTROL ROOM

A formal control room design review will be conducted. Section 7.5 includes a description of the main control room basic design and the visual displays important to safety. Complete documentation will be provided in an amendment in context with the control room design review.

18.5 CONTROL CENTERS OUTSIDE OF THE MAIN CONTROL ROOM

See Section 7.4, which provides a description of the remote shutdown facility (RSF).

A complete description of the human factors evaluations of the facilities listed below will be provided in an amendment to the FSAR.

1. The RSF

2. The technical support center (TSC)

3. The emergency operations facility (EOF)