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SNUPPS
Standardized Nuclear Unit
Power Plant System

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Nicholas A. Petrick
Executive Director

April 15, 1983

SLNRC 83-0019 FILE: 0281.6/0671.1/0278

SUBJ: Generic Letter 82-33

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References:
1) SLNRC 81-042 dated 6/4/81
2) SLNRC 81-051 dated 6/26/81
3) SLNRC 81-065 dated 8/12/81
4) SLNRC 82-004 dated 1/19/82
5) SLNRC 82-016 dated 3/16/82
6) SLNRC 82-020 dated 4/12/82
7) KMLNRC 82-214 dated 6/29/82
8) SLNRC 82-031 dated 7/6/82

Dear Mr. Denton:

The SNUPPS Utilities (Kansas Gas and Electric Company and Union Electric Company) received NRC Generic Letter 82-33, Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability, in December, 1982. The Generic Letter requested that holders of Construction Permits furnish by April 15, 1983, the proposed schedule for completing each of the basic requirements of the supplement. This letter provides the requested information for the SNUPPS Utilities.

CURRENT STATUS

A003
ADD: W. Paulson

Due to the construction and licensing status at the time the TMI requirements were issued in NUREG-0737, the SNUPPS Utilities have already addressed those requirements in Chapter 18.0 of the FSARs and in letters to the NRC. Items which have been reviewed and approved by the NRC as well as those items still considered unresolved by the NRC are addressed in Section 22 of the Callaway and Wolf Creek Safety Evaluation Reports. In a few cases, these early efforts by the SNUPPS Utilities have resulted in some differences between our efforts and the NRC's requirements issued in the supplement. The SNUPPS Utilities will rely on the NRC commitment to make allowances for the work done in a good-faith effort.

The following paragraphs describe the current status of the major items addressed in Generic Letter 82-33:

(a) The SNUPPS Safety Parameter Display System (SPDS) is presently under development. The system was described in the SNUPPS response to Generic Letter 81-10 in June, 1981 (Reference 1). The Attachment presents the details of the current status of our developmental efforts.

(b) An initial assessment of the Callaway and Wolf Creek Control Room designs was conducted by the Essex Corporation for both the Callaway and Wolf Creek Control Rooms. The results of these reviews and of meetings with the NRC staff were provided to the NRC in several letters (References 2, 3, 4, 5, 6, and 7). Items remaining open from this initial assessment are addressed as a part of the remaining control room design review program described in the Attachment.

(c) A description of the compliance of the SNUPPS design to the recommendations of Regulatory Guide 1.97 was provided to the NRC in July 1982 (Reference 8). This comparison of the SNUPPS plants to 1.97 criteria is now contained in the SNUPPS FSAR as Appendix 7A.

(d) The Westinghouse Owners' Group has issued Emergency Response Guidelines (ERG) for use in preparing Emergency Operating Procedures (EOP). These ERGs are being reviewed and as we understand are close to approval by the NRC. The SNUPPS Utilities have each prepared Writers' Guides for use in developing EOPs and are well along in the effort to prepare plant specific procedures based upon the ERGs. The Attachment describes the ongoing activities for the SNUPPS EOP generation efforts.

(e) The Emergency Response Facilities (ERF) buildings at both sites, including the TSC, OSC, and EOF are completed. Descriptions of these facilities were first documented to the NRC in Reference 1 and are described in a portion of each plant's Emergency Plan. Current status of equipment installation and the schedule for having the facilities fully operational is given in the Attachment.

As described above and in the Attachment, the SNUPPS Utilities are well advanced in developing programs to address the emergency response enhancements described in the Generic Letter.

SCHEDULE

The SNUPPS Utilities are presently proceeding vigorously with the design, testing, and implementation of all the control room enhancement elements (SPDS, EOPs, Reg. Guide 1.97, CRDR) and ERFs. As shown in the following schedules, all hardware will be received, all structures will be completed, and EOP's will be implemented prior to fuel load for each plant.

The current schedules for fuel loading at the plants are as follows:

Callaway - April 1984

Wolf Creek - October 1984

The required computer systems, which are a major portion of the effort to meet NUREG-0737 Supplement I, are complex and require significant integration. It is anticipated that full operability, training, and integration of the computer systems with the emergency operating procedures, where applicable, will not be completed until the first refueling. It is currently projected that these refuelings will take place as follows:

Callaway - June, 1986*

Wolf Creek - December, 1986*

To summarize schedular commitments made in the Attachment, the listing below provides the SNUPPS Utilities plans for submitting other information requested in NUREG-0737, Supplement 1.

NUREG-0737

Supplement 1

Submittal	Reference Section	Schedule
1. SPDS Safety Analysis	4.2a	December 30, 1983
2. CRDR Program Plan	5.2a	June 1, 1983
3. CRDR Summary Report	5.2b	120 days prior to Callaway fuel load date
4. RG-1.97 Report	6.2	Complete - Included in FSAR as Appendix 7A

*Based upon projections of energy requirements needed to supply the SNUPPS Utilities systems and typical first cycle nuclear power plant availability experience.

Submittal	Reference Section	Schedule
5. EOP Technical Guidelines	7.2a	Complete - Sent by WOG in September, 1982
6. EOP Procedures Generation Package	7.2b	Callaway - May, 1983 Wolf Creek - October, 1983

These schedules were discussed with the SNUPPS project managers for the NRC (Messrs. Edison and Holonich) in a meeting on April 12, 1983.

Per the requirements of 10CFR50.54(f), Kansas Gas and Electric and Union Electric are forwarding, under separate cover, letters which reference this letter and incorporate the information contained herein into their license applications.

Very truly yours,

N A Petrick

Nicholas A. Petrick

JOC/nld4a1
Attachment

cc: G. L. Koester KGE
D. T. McPhee KCPL
J. H. Neisler USNRC/CAL
H. Roberds/W. Schum USNRC/WC
D. F. Schnell UE

ATTACHMENT

I. SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

The Safety Parameter Display System (SPDS) conceptual design was previously described in Reference 1. The following discussion updates the design and provides the current status.

A. GENERAL CONSIDERATIONS

SPDS has been designed jointly by a group of Westinghouse NSSS utilities of which SNUPPS was a member. It provides a centralized, flexible, computer-based data and display system to assist control room personnel in evaluating the safety status of the plant. This assistance is accomplished by providing control room personnel with a high-level graphical display containing a minimum set of key plant parameters representative of the plant safety status. There are three displays for the SPDS; 1) Normal Operation, 2) Heatup and Cooldown, and 3) Cold Shutdown. All graphical displays are presented to control room personnel on a multiplecolor CRT.

All data displayed by the SPDS is validated by comparing redundant sensors, checking the value against reasonable limits, calculating rates of change, and/or checking temperature versus pressure curves.

All displays of the SPDS have been carefully designed by persons with plant operating experience and evaluated against human factors design criteria. The concepts used in the SPDS design have been verified using the Indian Point simulator. The intent of the SPDS is to present to control room personnel a few easily understandable displays which use color coding and pattern recognition techniques to indicate off-normal values. These displays are updated and validated on an essentially real-time basis.

The SPDS will be operable during normal and abnormal plant operating conditions and will operate during all modes of plant operation. The availability factor of the SPDS is dependent on plant operating conditions. SPDS maintenance and modifications will take place as required and will be scheduled when possible during the cold shutdown mode of plant operation. The normal operation mode will encompass all plant conditions at or above normal operating pressure and temperature. When the reactor coolant system is intentionally cooled below normal operating values, the operator will select the Heatup-Cooldown mode which alters the limit checking algorithm for the key parameters. An additional mode is also provided to address cold shutdown plant conditions.

B. DISPLAY HARDWARE LOCATIONS AND OPERATION

The SPDS will be implemented on a single CRT located adjacent to the main control console. This CRT contains the displays from which the overall safety status of the plant may be assessed. The operator can select the display appropriate for the mode of operation. The SPDS has been designed such that

control room personnel can utilize its features without requiring additional operations personnel. Current plans are to provide an additional CRT in the control room to provide flexibility. The SPDS displays are also planned to be available in the Technical Support Center (TSC).

C. DISPLAY CONTENTS

The SPDS displays consist of bar graphs of selected parameter values, digital status indicators for important safety system parameters and digital values. The parameters indicated by bar graphs and digital values include: RCS pressure, RCS cold leg temperature, pressurizer level, steam generator levels and steam generator pressures. Status indicators are provided for containment environment and secondary system radiation. Reactor vessel level, core exit temperature, amount of subcooling and containment radiation are indicated by digital values. In addition, there is a message area which will be used to indicate further information related to power level, core average temperature, date, time, and the time of certain engineering safety feature actuation signals.

Each of the bar graphs indicates wide-range values. If a parameter's value is outside the normal range, the bar color will turn red. Arrows next to the bar will indicate the trend direction (increasing or decreasing) based on data smoothing algorithms.

An exception to the above display definition is the SPDS display for cold shutdown. This display consists of bar graphs, digital values, and trend graphs of the important safety parameters. The important safety parameters are: RCS pressure, core exit temperature, RCS vessel inventory, Residual Heat Removal (RHR) flow rate, RHR heat exchanger inlet and outlet temperature, and two source range channels counts per second.

D. HUMAN FACTORS CONSIDERATIONS

Human factors engineering and industrial design techniques have been effectively combined to establish man-machine interface design requirements, maximize system effectiveness, reduce training and skill demands, and minimize operator error.

The CRT color graphic formats and functional keyboard designs have been developed through an interdisciplinary team of senior operational, human factors, industrial design and computer interface personnel.

Minimum use of color combined with a simplified format throughout the CRT presentation are the key design features which provide both normal and off-normal pattern recognition. Control room personnel who are the end users, have been directly involved from the conception to insure that man-machine interface goals of SPDS have been satisfied. Human factor engineering standards and testing verification have been used which are consistent with accepted practices.

E. VALIDATION AND VERIFICATION

The SPDS is implemented on a digital computer system which includes a peripheral display generator computer for color graphic displays. The software that controls the sensor data validation, key parameter construction, and display formats has been developed under strict verification and validation procedures. The original development of the SPDS software began with a functional specification that was developed over a period of 18 months by a technical committee comprised of members from a number of utilities and consultants. These functional specifications are transformed into a design specification. Reviews of the design specification assure conformance of the SPDS to those functions discussed in NUREG-0737 Supplement I.

During the course of software development, a set of static test cases were developed which test the key features of each software module. Furthermore, static system test cases were developed and used to verify the correct operability of the total system. A set of dynamic test cases were generated by recording nuclear plant simulator data on magnetic tape from a number of different plant transients which test the dynamic behavior of the system under "real" conditions. A design review that compares these test results to the original functional and design specifications was performed. A selected number of the static test cases were "frozen" such that they could be used to verify future changes to the software. In summary, verification and validation was addressed and designed into the SPDS software from the beginning to provide a highly reliable product and a mechanism for identifying and controlling future changes. The generic SPDS program will be updated to implement necessary site specific modifications. Changes to the SPDS software to effect site specific modifications will undergo a rigorous verification and validation program to assure the SPDS functions are as originally designed.

F. STATUS

The SPDS software is currently being tailored to meet site specific requirements. Hardware for displaying the SPDS has been purchased. Linking the SPDS to the BOP computer system and Emergency Response Facility Information System is scheduled to occur prior to fuel load. However, due to the complexity of this linking process the SPDS will not be fully completed until the startup after the first refueling of each plant. This fully completed target date of the first refueling includes full operability, completion of operator training, and integration of the SPDS into the emergency operating procedures, where applicable.

G. SAFETY ANALYSIS PROVIDING BASIS FOR SPDS PARAMETER SELECTION

The basis for the parameter selection used in the SPDS utilized several sources of input. Principal sources were an NSAC probabilistic study of accident sequences and an independent review of the accident sequences by the group of Westinghouse utilities sponsoring the SPDS development. These analyses are scheduled for submittal to the NRC by December 30, 1983 and will address any site specific changes that have been included.

H. PRE IMPLEMENTATION REVIEW

The SPDS as designed by the group of Westinghouse NSSS utilities has been the subject of several review meetings with the NRC in 1981 and 1982. Thus, it is the SNUPPS position that the SPDS utilized on the SNUPPS plants has already had a preimplementation review by the NRC and has been found to be acceptable.

II. DETAILED CONTROL ROOM DESIGN REVIEW (DCRDR)

A. Status of DCRDR

The SNUPPS DCRDR (Detailed Control Room Design Review) consists of five interrelated programs, the scope and status of which are as follows:

1. Emergency Response Guidelines (ERGs). These guidelines are being developed by Westinghouse and the Westinghouse Owners Group (WOG) for guidance in writing plant-specific Emergency Operating Procedures for all Westinghouse plants. The ERGs describe the functions to be performed by plant operators in response to a comprehensive set of emergency situations. The completeness of the ERGs has been ensured by (a) performance of probabilistic analyses of accident sequences and provision of guidelines for all but extremely improbable sequences, (b) inclusion of instructions for maintaining critical safety functions during emergency situations that have not been diagnosed, and (c) a series of reviews of the ERGs by the NRC. Revision 0 of the ERGs, applicable to plants (such as SNUPPS) with high pressure safety injection systems, was completed in September 1982. Revision 1 of these ERGs is to be completed in June 1983.

2. System Review and Task Analysis. This program has been performed by Westinghouse for the WOG and includes identification of (a) all tasks to be performed by control room operators, (b) all instrumentation, and (c) all controls needed to implement the ERG. This program is complete except for a final report.

3. Control Room Survey. This task, which is being performed by SNUPPS, consists of confirming that the installed instruments and controls agree with the instruments and controls required to implement the ERGs. This task is scheduled for completion in September 1983.

4. Human Engineering Review of Control Room. SNUPPS contracted with Essex Corporation in September 1980 for a human engineering review of the control room. The review was initiated at that time in order to provide as much lead time as possible to make any changes determined to be necessary. The NRC (Human Factors Engineering Branch) followed up on the Essex review by performing a review of its own in July 1981. Actions taken by SNUPPS in response to these reviews have been documented in a series of letters, dated from June 1981 to April 1982. Because the control rooms at Callaway and Wolf Creek were not complete at the time of the Essex and NRC reviews, there are several aspects of the design (e.g., lighting, noise, communication) that have not yet been evaluated. SNUPPS plans to perform a self-evaluation of those aspects of the design, utilizing the guidelines of NUREG-0700, when the control rooms are substantially completed. It is expected that the NRC will also complete its independent review at that time. Tentatively, these supplementary reviews are scheduled to be completed 120 days before fuel load.

5. Verification and Validation. A verification and validation program directed to the ERG was performed at the Callaway simulator (which is a duplicate of the Callaway control room) in June 1982. The program consisted of "walking-through" forty-one of the ERGs (substantially the entire set) and evaluating the guidelines. The operators for this program were operating personnel from the Callaway and Wolf Creek plants. Westinghouse engineering and training personnel observed the program. An NRC representative was present for part of the time. Though the verification and validation program was specifically focused on the ERGs, the program also verified that all necessary instruments and controls are available and conveniently located in the SNUPPS control room. Videotapes of the program are available for audit.

B. Submittal of Program Plan

A formal program plan will be submitted to the NRC by June 1, 1983.

C. Submittal of Summary Report

A summary report of the DCRDR program, which will reference reports on the five subprograms described above, will be submitted to the NRC 120 days prior to scheduled fuel load at the first SNUPPS plant. The limiting element of this report, as regards schedule, is completion of the human engineering review of control room environmental conditions, as discussed under item 4, above.

III. REGULATORY GUIDE 1.97, REVISION 2

A comprehensive comparison of the SNUPPS design to the guidance of Regulatory Guide 1.97, Rev. 2 is contained in Appendix 7A to the SNUPPS FSAR.

With a few exceptions, all of the instrumentation and design features described in Appendix 7A of the SNUPPS FSAR will be installed and operable prior to fuel load of the Callaway and Wolf Creek plants. Instrumentation described in Appendix 7A that may not be operable at the time of fuel load is as follows:

1. Source range instrumentation, qualified to post-accident environmental conditions (Table 7A-3, Data Sheet 1.1)

2. Reactor vessel water level instrumentation system (Table 7A-3, Data Sheet 1.4)

3. Subcooling monitor (Table 7A-3, Data Sheet 1.5)

4. Radiation monitors for releases from steam generator safety/ relief valves or atmospheric dump valves (Table 7A-3, Data Sheet 12.3)

5. Auxiliary feedwater pump turbine exhaust monitor (Table 7A-3, Data Sheet 12.4)

The SNUPPS Utilities are proceeding on a best efforts basis to procure, install, and test these instruments. However, it is anticipated that some of these instruments will not be fully operational until the startup after the first refueling outage at Callaway and at Wolf Creek. Factors mitigating against an earlier commitment to operational status of these instruments are that (a) the need for the above instruments was not identified until design and construction were well advanced, (b) some of these instruments are developmental in nature (e.g., items 2 and 4 above), (c) in some cases design and environmental qualification programs have not yet been completed (items 1, 2, and 3 above).

Read-outs of RG 1.97 instruments in the Technical Support Center and Emergency Operations Facility are through the ERFIS (Emergency Response Facilities Information System) computer system. The schedule for completion of these facilities is given in Section IV of this attachment.

IV. EMERGENCY RESPONSE FACILITIES

The Emergency Response Facilities for the Callaway and Wolf Creek plants are described in each plant's FSAR Site Addendum Section 18.3.2.2 and in Reference 1. The following table summarizes the status and schedule for these facilities:

	Callaway	Wolf Creek
Technical Support Center (TSC)		
Structure	complete	complete
HVAC	4/83	9/83
Standby Diesel Generator	8/83	9/83
Operations Support Center (OSC)		
Structure	complete	complete
Emergency Operations Facility (EOF)		
Structure	complete	complete
HVAC	complete	complete
Standby Diesel Generator	complete	4/83

The backup EOF for Callaway is located in the SEMA offices in Jefferson City, Mo. Jefferson City is located approximately 25 miles southwest of the plant site. The backup EOF for Wolf Creek is located at Beto Junction, Kansas, approximately 13 miles from the plant.

Communications equipment will be available on the following schedule:

	Callaway	Wolf Creek
Telephones	8/83	6/84
Radios	10/83	6/84
Meteorological Data Telemetering	6/83	1/84

The Emergency Response Facilities Information System (ERFIS) for both Callaway and Wolf Creek consists of the following subsystems; 1) ERFIS Computer, 2) Radiololgical Release Information System (RRIS), 3) Safety Parameters Display System (SPDS), 4) Post Accident Sampling System (PASS) and, 5) Radiation Monitoring System (RMS). The status of the SPDS has been discussed earlier in this report. The following table gives the status of these subsystems.

	Callaway	Wolf Creek
Emergency Response Facilities Information System (ERFIS)		
Hardware	received	8/83
Hardware testing	9/83	9/83
Software	9/83	10/83

It is expected that ERFIS computer will be available for limited operation at Callaway and Wolf Creek in November, 1983.

Radiological Release Information System (RRIS)		
Complete system integration	7/83	10/83
Factory acceptance test	7/83	10/83
Site Delivery	8/83	11/83
RRIS Startup	11/83	1/84
Post Accident Sampling System (PASS)		
System design and integration	complete	complete
Factory acceptance test	5/83	7/83
Site Delivery	6/83	8/83
PASS Startup	7/83	2/84
Radiation Monitoring System (RMS)		
Site Delivery	complete	complete
Startup	7/83	1/84

These systems will also be tied into the Balance of Plant Computer. Because of the complex interfacing with other computer systems, it is probable that the computer systems will require at least 12 additional months to be fully tested and validated. The development of procedures and training of personnel will require an additional 4 to 6 months. These systems will be functional by the first refueling.

Callaway Emergency Plan Implementing Procedures are currently 60% complete. These will be completed by October, 1983. Emergency Plan Training will be done in two phases. Classroom training will be complete by July, 1983 with walkthroughs and drills complete by October, 1983.

The Wolf Creek draft Emergency Plan Implementing Procedures will be issued June, 1983 and will be completed by December, 1983. Classroom training will be complete by April, 1984 with walkthroughs and drills complete by June, 1984.

The Callaway Public Relations Emergency Response Program will be completed by October, 1983. The Wolf Creek Public Relations Program will be completed by December, 1983.

V EMERGENCY OPERATING PROCEDURES

The Westinghouse Owners Group Generic Technical Guidelines, Revision [illegible] have been submitted to the Nuclear Regulatory Commission and have been used to develop plant specific Emergency Operating Procedures (EOP) for Callaway and Wolf Creek. The Generic Guidelines were submitted to the Commission in September, 1982.

The method for developing plant specific EOP's from the generic guidelines was started in December 1981 and completed in June of 1982. The plant specific EOP's were used during a verification and validation program using the Callaway simulator in June, 1982. Westinghouse Owners Group, INPO and NRC representatives were present during this program. The program was also put on video tape. The results of the verification and validation program indicated some minor changes were required to the generic guidelines. These will be corrected in a revision to the guidelines.

The EOP Procedures Generation Package is scheduled for submittal to the NRC in May, 1983 at Callaway and October, 1983 at Wolf Creek. Operators are presently using the plant specific EOP's, developed for the validation program, in their training programs. The implementing date for the EOP's will be at fuel load for Callaway and Wolf Creek respectively.

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