UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE SAFETY PARAMETER DISPLAY SYSTEM

PUBLIC SERVICE COMPANY OF COLORADO

FORT ST. VRAIN NUCLEAR GENERATING STATION

DOCKET NO. 50-267

1.0 INTRODUCTION

All holders of operating licenses issued by the Nuclear Regulatory Commission (i.e., licensees) and applicants for an operating license must provide a Safety Parameter Display System (SPDS) in the control room of their plant. The Commission-approved requirements for the SPDS are defined in Supplement 1 to NUREG-0737.

The purpose of the SPDS is to provide a concise display of critical plant variables to control room operators to aid them in rapidly and reliably determining the safety status of the plant. NUREG-0737, Supplement 1 requires licensees and applicants to prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Licensees and applicants shall also prepare an Implementation Plan for the SPDS, which contains schedules for design, development, installation, and full operation of the SPDS as well as a design Verification and Validation Plan. The Safety Analysis and the Implementation Plan are to be submitted to the NRC for staff review. The results from the staff's review are to be published in a Safety Evaluation (SE).

2.0 SUMMARY

The staff reviewed Public Service Company of Colorado's SPDS Safety Analysis Report, additional information on electrical isolators, and audited a prototype of the display system. From the results of the review and the commitments made by the licensee, the staff concludes that it is acceptable for the licensee to continue implementing its SPDS program.

3.0 EVALUATION

Public Service Company of Colorado submitted for staff review a Safety Analysis Report and a Program Plan (Ref. 1) on the Safety Parameter Display System for the Fort St. Vrain High Temperature Gas Reactor. In addition, the staff audited a prototype of the display system and documented the results in a report (Ref. 2). Our Safety Evaluation of the design is presented next.

A. Display Description

The Fort St. Vrain SPDS will be incorporated as a function within the existing plant computer system. A single, dedicated cathode-ray tube (CRT) will be used in the control room as the SPDS's interface with the operator. The SPDS's display formats will be distinguished by five labeled critical safety function alarm boxes in the lower part of each display page. These boxes appear on the primary display format and the master menu of critical safety functions, as well as on the secondary display formats. Each of the secondary display formats is dedicated to a critical safety function and contains current data on the process variables for the function.

The top-level-display page contains a master menu of secondary level displays. Operator selection from the menu is performed by using a track-ball to position a cursor to the desired item, then a keystroke to activate the desired display format. Operator access to the secondary display formats may also be achieved by positioning the cursor at the poke points located immediately above each alarm box at the bottom of the screen, then a keystroke to display the selection. A third method of accessing the display formats is provided by means of a "page forward" key and a "page backward" key.

B. Variable Selection

Section 4.1.(a) of 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. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective actions by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phases of an accident."

In addition, Section 4.1.(f) states:

"The minimum information to be provided shall be sufficient to provide information to the plant operator about:

(i) Reactivity control

(ii) Reactor core cooling and heat removal from the primary system

(iii) Reactor coolant system integrity

(iv) Radioactivity control

(v) Containment conditions."

For review purposes, these five functions have been designated as Critical Safety Functions.

The Critical Safety Functions defined for the SPDS by NUREG-0737, Supplement 1, apply to light water reactors. The Fort St. Vrain Nuclear Generating Station is a High Temperature Gas-Cooled Reactor (HTGR). The staff's review of the Fort St. Vrain SPDS Safety Analyis (Ref. 1) evaluated the Critical Safety Functions monitored by the display system to determine if they meet the intent of the Critical Safety Functions defined in NUREG-0737, Supplement 1. In addition, the staff's review of the Safety Analysis evaluated:

- the adequacy of the process variables monitored to assess the safety status of a Critical Safety Function,

- the adequacy of the display system's response time to plant transients of monitored process variables to aid control room operators to rapidly and reliably determine the safety status of the plant,

- the adequacy of the scope and use of the SPDS to monitor the Critical Safety Functions.

The results of our review are presented next.

In the licensee's Safety Analysis (Ref. 1), five Critical Safety Functions are identified for the Fort St. Vrain reactor. They are:

- Reactivity Control, which relates to heat generation,

- Primary Heat Removal, and

- Secondary Heat Removal, both of which relate to primary coolant circulation for heat removal from the core and heat transfer to the secondary coolant,

- Primary Coolant System Integrity, which relates to both core cooling and containment of fission products, and

- Radioactivity Control, which relates to fission product releases from the fuel or the Prestressed Concrete Reactor Vessel (PCRV).

In evaluating these functions we note that the NUREG-0737, Supplement 1 Critical Safety Functions of Reactor Coolant System Integrity and Containment Conditions are addressed principally by Fort St. Vrain's Primary Coolant System Integrity function. The PCRV serves to contain the mass inventory of the primary coolant, the helium gas. Also, the PCRV serves as a barrier to the uncontrolled release of radioactive fission products within the primary coolant system. Fort St. Vrain's other Critical Safety Functions are directly related to the remaining functions stated in NUREG-0737, Supplement 1. Based on these facts, the staff confirms that the five Critical Safety Functions identified by the licensee for the Fort St. Vrain reactor respond to the intent of the five Critical Safety Functions identified in NUREG-0737, Supplement 1.

The licensee's Safety Analysis provides a detailed basis to justify the process variables used in the SPDS. The analysis states that those process variables, which indicate that fuel particle temperatures or PCRV pressure are unsafe or could become unsafe, are also variables that indicate abnormal conditions significant to safety. The plant's Technical Specifications include two safety limits: a reactor core safety limit and a reactor vessel safety limit. The licensee's analysis states that operation within these limits ensures that plant conditions do not result in an uncontrolled or unplanned release of radioactivity. The process variables associated with these safety limits are included in the SPDS.

In terms of Critical Safety Functions, the process variables selected by the licensee for the SPDS are identified in Table I. The data sample time and the data processing/alarm processing time for each process variable are also identified in Table I. The sum of these times defines the SPDS response time, which is also stated.

The licensee's Safety Analysis contains a detailed discussion on each process variable to justify its selection for the SPDS. The importance of the process variable to the Critical Safety Function is defined in terms of significant events and the goals of safe operation. This portion of the licensee's design process for the SPDS appears to be thorough and comprehensive.

The staff's review of the licensee's Safety Analysis confirms that the process variables selected for the SPDS will reflect the status of the Critical Safety Functions for a wide range of events and abnormal plant conditions. Also, our review of Reference 5 noted that the development of the SPDS was being coordinated with the development of the Emergency Operations Procedures, which is responsive to the recommendations made in NUREG-0737, Supplement 1.

The staff also evaluated the REACTIVITY display format and recommended (via a phone conference with the licensee on January 21, 1986) that the variables AVERAGE NEUTRON POWER, PRIMARY HEAT BALANCE POWER, and SECONDARY HEAT BALANCE POWER be grouped as a set within the display format. By grouping these related process variables, it facilitates control room operator use of them. Significant differences among these variables will result in a change in the temperature of the fuel. A rise in fuel temperature results in negative feedback of reactivity to the nuclear fission process. We also recommend that primary coolant moisture be presented as a trend graph to facilitate rapid detection of water leaks into the primary system.

In Reference 5, the licensee responded to the above staff recommendations. The licensee proposed to change the order of the variables on the REACTIVITY display to read: PRIMARY HEAT BALANCE, SECONDARY HEAT BALANCE, AVERAGE NEUTRON POWER, then NEUTRON FLUX RATE-OF-CHANGE. The proposed change will be reviewed by operators for their input and comments. Based on the information provided, the staff finds the licensee's commitment acceptable.

Also in Reference 5, the licensee states that a trend graph of primary coolant moisture cannot be implemented on their CDC 1974 computer. However, the licensee states that the numerical value of the rate-of-change in primary coolant moisture is displayed in the SPDS, which is acceptable to the staff.

Our review of data sample times did note a potential problem in the data sample time for maximum region outlet temperature mismatch. The total SPDS response time for this variable is stated as one minute and 25 seconds, which appears to be excessive if immediate operator actions are necessary. We discussed this issue with the licensee in our phone conference of January 21, 1986.

In Reference 5, the licensee responded to the above expressed concern. The licensee states that the Fort St. Vrain reactor core has a thermal response time in excess of 5 minutes to changes in reactor power level or region flow rate. This response time is due to the high heat capacity and large mass of the graphite moderator. Also, the Technical Specification LCO 4.1.7 imposed limits on the Maximum-to-Average Region Outlet Temperature Mismatch, which are set to provide a reasonable margin to maintain the core within the envelope of conditions assumed in developing core safety limits. Continuous operation to the LCO 4.1.7 mismatch limits will not result in fuel damage. This LCO allows for an orderly shutdown of the reactor (by control room operators) if the allowable maximum-to-Average Region Outlet Temperature Mismatch is exceeded by 100 degrees F. Based on this information, the staff finds the SPDS's response time for maximum region outlet temperature mismatch appropriate.

The staff also evaluated the Safety Analysis for the scope and use of the SPDS. The data (from Ref. 1) in Table II define the operational use of the process variables as a function of reactor operating mode. Our review of these data concludes that each Critical Safety Function is monitored by at least one or more process variables during the modes of operation stated, and we find this acceptable.

C. Data Validation

The staff reviewed the licensee's design to determine that means are provided to assure that the data displayed are valid. The licensee's Program Plan (Ref. 1) for the SPDS states that hardware rejects, errors detected by transducer range limit checking, and errors detected during engineering unit conversion will cause the data to be flagged as invalid. Conservative alarm and warning limits are to be selected and implemented for the SPDS process variables so that the operator will be alerted of approaching abnormal plant conditions prior to reaching the setpoints of the safety system. An inverse video blinking magenta colored "V" next to the displayed value of a process variable is used to identify invalid data to the operator.

Based on the information provided by the licensee, the staff confirms that means are available in the SPDS design to assure that the data displayed are valid.

D. Human Factors Program

The staff's review of this Program consisted of a pre-implementation audit of the licensee's prototype SPDS and an evaluation of the licensee's Program Plan. The results of our pre-implementation audit (Ref. 2) concluded that the prototype display system was user-friendly, uncluttered, and easy to read and comprehend. The results from our evaluation of the licensee's Program Plan (Ref. 1) are presented next.

The licensee's Program Plan defines the functional and operational requirements, which include the human factors requirements for the operator's interface. The Program Plan contained descriptions of design features, such as details of the keyboard and access to related display formats. The staff's review of this material concluded that it is consistent with our audit observations of the design, and in some cases, improvements have been added. The licensee's Program Plan also contained copies of the display formats in the SPDS, and these were labeled as prototypes. The staff compared these display formats with the display formats that existed at the time of our audit (Ref. 2). Our review concluded that the display formats in the Program Plan were cluttered with respect to the display formats evaluated during our audit. Specifically, our review of the display format titled PRIMARY SYSTEM noted several human engineering deficiencies, such as:

- Inconsistent use of text, MAX and MAXIMUM,

- a confusing display of the title, PRIMARY DISPLAY, and the apparent subtitle, CORE INLET ORIFICE VALVES DATA 0317, which is the page-up data,

- the numerical value of the outlet temperature mismatch appears to be 399 degrees below zero.

Our review also noted that many features of these display formats do not conform to the design directives contained in DD-SLS-1, DESIGN DIRECTIVE FOR SCREEN LAYOUT AND STRUCTURING. Our audit of this design directive (Ref. 2) concluded that the design guidelines therein appeared appropriate and should prove useful to designers and design verifiers in the development of the SPDS. We stated the above identified concerns in our phone conference with the licensee on January 21, 1986.

In Reference 5, the licensee agreed that the prototype displays do not in all cases conform to DD-SLS-1, Design Directive for Screen Layout and Structuring. Furthermore, the licensee states that the SPDS developers and the verification and validation team will review the screens in the final format to ensure that they conform with DD-SLS-1. The staff finds this commitment by the licensee acceptable.

E. Electrical and Electronic Isolation

In order to satisfy the NRC requirements concerning the Safety Parameter Display System (SPDS), Public Service Company of Colorado submitted an SPDS Program Plan and Safety Analysis by letter dated January 20, 1984. This material did not address the requirement that the SPDS must be suitably isolated from equipment and sensors that are used in safety systems to prevent electrical and electronic interference. A request for additional information, which included specific questions on these isolators, was sent to the licensee on September 14, 1984. The requested information was received in letters dated January 31, 1985, March 8, 1985, August 26, 1985, and February 19, 1986. Several telephone conferences were held with the licensee to clarify the information submitted on the Energy Incorporated's isolation amplifiers and to discuss the analysis used in arriving at the values of the maximum credible fault. The licensee submitted a letter, dated April 29, 1986, in which the clarifications discussed in the telephone conferences were documented.

The Fort St. Vrain SPDS is isolated from the plant's safety-related systems by electrical isolation devices manufactured by Energy Incorporated. These isolators use a Burr-Brown optical isolator for the Class 1E (input) to non-Class 1E (output) isolation. The isolators were subjected to a surge withstand capability test, a thermal drift test, a hi-pot test, and a design basis fault test. The design basis fault test applied the maximum credible fault (MCF) voltage/current to the output terminals of the isolator in the transverse mode. The MCF voltage/current used in the test was more severe than that analyzed for the plant.

The MCF voltage and current have been determined to be 120 VAC at a current of 4 amps. The low value of the amperage is the result of power distribution system modifications and isolator rack modifications. The 120 VAC power distribution panels were modified to accept an AMP-TRAP 4 amp fuse in series with a 15 amp circuit breaker. This limits the available current to the isolator cabinet at 4 amps.

The isolator cabinets were also modified by removing all sources of AC power going to the cabinets with exception of the source coming from the power distribution panel via the AMP-TRAP fuses, which are 4 amp fuses. This now being the only source of AC power within the isolator cabinets, the MCF voltage/current can be set at 120 VAC at a current of 4 amps.

The specific isolator unit tested contained four channels. All input channels were terminated with their normal input resistors. The MCF was applied to the non-Class 1E output of Channel 2. The actual test used 480 VAC (60 Hz) at 12 amps driving current as the MCF voltage and current. The acceptance criteria stated that upon the application of the MCF, no more that 0.25 mv of the 60 Hz signal shall appear across any of the input resistors of the isolator. Upon the application of the MCF, Channel 2 suffered output circuit component damage, and Channel 1 suffered a little smoke damage. All four Class 1E input channels of the isolator passed the acceptance criteria and were undamaged.

In addition to the tests previously mentioned, the isolators were qualified to IEEE-381-1977, "Standard Criteria for Type Tests of Class 1E Modules Used in Nuclear Power Generating Stations." The isolators are located in a mild environment and, therefore, they do not come under the requirements of 10 CFR 50.49.

Based on the staff's review of the information submitted by the licensee with respect to the Energy Incorporated isolation devices, the staff concludes that these devices are qualified isolators and are acceptable for interfacing the SPDS with Class 1E safety-related systems. The staff also concludes that these devices meet the Commission's requirements in NUREG-0737, Supplement 1 as stated in Section 4, Safety Parameter Display System.

F. Verification and Validation Program (V&V)

The staff evaluated this Program during our pre-implementation audit of the SPDS. Our review concluded (Ref. 2) that the licensee's V&V Program was similar to the one described in NSAC-39, "Verification and Validation For Safety Parameter Display Systems." Further, we concluded that if the Program was properly implemented, it should serve to minimize errors in the development of the SPDS.

The licensee provided additional details on the V&V Program in the Program Plan (Ref. 1). The staff's review of this program confirms our previous audit results.

G. Operator Training and Procedures

The licensee's SPDS Program Plan (Ref. 1) commits to an operator training program for use of the SPDS. The operator training program described will contain information and provide guidance for the resolution of invalid data defined by the SPDS. Further, operating procedures will be developed that will allow the operator to rapidly and correctly assess the safety status of the plant when the SPDS is not available. Finally, an SPDS user's manual will be available in the control room for the operators' use and reference. The staff finds these commitments by the licensee acceptable.

4.0 CONCLUSIONS

The NRC staff reviewed a Safety Analysis and audited a prototype of Fort St. Vrain's SPDS for conformance to the NUREG-0737, Supplement 1 requirements. As a result of that review, the staff concludes:

- the Critical Safety Functions identified by the licensee for the Fort St. Vrain reactor respond to the intent of the five Critical Safety Functions identified in NUREG-0737, Supplement 1,
- the process variables selected for each Critical Safety Function reflect the status of the safety function for a wide range of events,
- a minimum of one process variable for each safety function is monitored and displayed by the licensee's SPDS for most modes of plant operation,
- means are available in the SPDS design to assure that the data displayed are valid,
- the licensee's Verification and Validation Program for the design and development of the SPDS should serve to minimize errors in the display system,
- with the licensee's commitments and our review of the design, we confirm that a Human Factors Program exists in the development of the SPDS,
- qualified isolators are being used to interface the SPDS with Class 1E Safety-related systems.

Based on its review to date, the staff concludes that no serious safety questions are posed by the proposed SPDS and, therefore, implementation of the SPDS by the licensee may continue. The conclusion that the SPDS implementation may continue does not imply that the SPDS meets or will meet the requirements of Supplement 1 to NUREG-0737. Such confirmation can be made only after a post-implementation audit or when sufficient information is available for the staff to make such a determination.

5.0 REFERENCES

1. Letter from D.W. Warembourg, Public Service Company of Colorado, to Regional Administrator, Region IV, NRC, Subject: Safety Parameter Display System Safety Analysis Report, dated November 13, 1984, with attachments.

2. Letter from E.H. Johnson, NRC, to O.R. Lee, Public Service Company of Colorado, Subject: SPDS Pre-Implementation Audit Results for Fort St. Vrain, dated September 14, 1984.

3. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability," US NRC Report NUREG-0737, Supplement No. 1, dated January 1983.

4. Letter from J.W. Gahm, Public Service Company of Colorado, to Regional Administrator, Region IV, NRC, Subject: Fort St. Vrain Procedure Generation Package, Summary Report, dated October 30, 1985.

5. Letter from D.W. Warembourg, Public Service Company of Colorado, to H.N. Berkow, NRC, Subject: Safety Parameter Display System, dated February 19, 1986.

NRC Reviewers: L. Beltracchi, DPWRL-B
R. Eckenrode, DPWRL-A

TABLE I

FORT ST. VRAIN SPDS

FUNCTIONS, VARIABLES, AND RESPONSE TIMES

CRITICAL SAFETY FUNCTION	PROCESS VARIABLE	DATA SAMPLE TIME	DATA PROCESSING ALARM PROCESSING TIME	SPDS RESPONSE TIME
Reactivity Control	1. Average Neutron Power	5 sec.	15 sec.	20 sec.
	2. Neutron Flux Rate-of-Change	5 sec.	15 sec.	20 sec.
	3. Primary Heat Balance Power	1 min.	no alarm	1 min. 5 sec.
	4. Secondary Heat Balance Power	5 min.	no alarm	5 min. 5 sec.
Primary Heat Removal	5. Power-to-Flow Ratio	5 sec.	15 sec.	20 sec.
		5 sec.	1 sec.	5 sec
	6. Primary Helium Flow	1 min.	15 sec.	1 min. 25 sec.
	7. Core Average Outlet Temperature	1 min.	no alarm	1 min. 25 sec.
	8. Maximum Region Outlet Temperature Mismatch	1 min.	15 sec.	1 min. 25 sec.
	9. Average Circulator Inlet Temperature	1 min.	15 sec.	1 min. 25 sec.
Secondary Heat Removal	10. Feedwater Flow	5 sec.	15 sec.	20 sec.
	11. Main Steam Temperature	5 sec.	15 sec.	20 sec.
	12. Main Steam Pressure	5 sec.	15 sec.	20 sec.
	13. Hot Reheat Steam Temperature	5 sec.	15 sec.	20 sec.
	14. Hot Reheat Steam Pressure	5 sec.	15 sec.	20 sec.
	15. Steam Jet Air Ejector Activity	1 min.	15 sec.	1 min. 25 sec.
Reactor Coolant System Integrity	16. Primary Coolant Pressure	5 sec.	15 sec.	20 sec.
	17. Primary Coolant Moisture	1 min.	15 sec.	1 min. 25 sec.
	18. Circ. & Steam Generator Penetration Pressure	5 sec.	1 sec.	5 sec.
Radioactivity Control	19. Primary Coolant Activity	1 min.	15 sec.	1 min. 25 sec.
	20 Stack Activity	1 min.	15 sec.	1 min. 25 sec.

TABLE II

FORT ST. VRAIN SPDS

PROCESS VARIABLES FOR SPECIFIC OPERATING MODES AND POWER LEVELS

REACTOR MODE SWITCH					
PROCESS VARIABLE	FUEL LOADING	OFF	RUN INTERLOCK SEQUENCE SWITCH STARTUP	LOW POWER	POWER
1. Average Neutron Power			x	x	x
2. Neutron Flux Rate-of-Change	x	x	x	x	x
3. Primary Heat Balance Power			x	x	x
4. Secondary Heat Balance Power					x
5. Power-to-Flow Ratio				x	x
6. Primary Helium Flow	x	x	x	x	x
7. Core Average Outlet Temp	x	x	x	x	x
8. Region Outlet Temp Mismatch			x	x	x
9. Average Circ Inlet Temp	x	x	x	x	x
10. Feedwater Flow				x	x
11. Main Steam Temp					x
12. Main Steam Pressure					x
13. Hot Reheat Steam Temp			x	x	x
14. Hot Reheat Steam Pressure			x	x	x
15. Steam Jet Air Ejector Activity	x	x	x	x	x
16. Primary Coolant Pressure	x	x	x	x	x
17. Primary Coolant Moisture		x	x	x	x
18. Circ. and Stm. Gen. Penet. Pressure	x	x	x	x	x
19. Primary Coolant Activity	x	x	x	x	x
20. Stack Activity	x	x	x	x	x

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