GULF STATES UTILITIES COMPANY

May 15, 1985
RBG-21,018
File Nos. G9.5, G9.33.4,
G9.19.2

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

Dear Mr. Denton:

River Bend Station - Unit 1
Docket No. 50-458

The attached response provides additional information requested in a letter dated August 30, 1984 from Mr. A. Schwencer to Mr. W. J. Cahill, Jr. concerning the River Bend Station Safety Parameter Display System (SPDS). The SPDS hardware is currently installed. The SPDS is scheduled to be fully functional (i.e., tested, procedures approved and training completed) by February 1986.

This submittal supplements Gulf States Utilities Company's April 24, 1984 SPDS safety analysis letter to your office.

Sincerely,

J. E. Booker

J. E. Booker
Manager-Engineering
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/RJK/je

Attachment

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Attachment

1. HUMAN FACTORS PROGRAM

Provide a description of the display system, its human factored design, and the methods used and results from a human factors program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator.

RESPONSE

The safety parameter display systems (SPDS) is a function of the General Electric (GE) Emergency Response Information System (ERIS) as described in NEDE-30284-P dated November 1983.(1) GE's extensive engineering, training, operations, startup, test design and analysis and information systems components have all contributed to provide well engineered human-factored displays required by Supplement 1 to NUREG-0737. During the NRC's July 24-27, 1984 Design Verification Audit of the GE SPDS design, the staff evaluated a report titled "Human Factors and Performance Evaluations of the Emergency Response Information System", July 10, 1984, ANACAPA Sciences, Inc.(2) This in conjunction with supplementary information submitted by GE in references 4 and 8 establishes the adequacy of the SPDS design

2. DATA VALIDATION

Describe the specific methods used to validate data displayed in the SPDS. Also describe how invalid data is defined to the operator.

RESPONSE

The methods used for the RBS SPDS to validate a value of a parameter to display when the input is based on more than one data point are the same methods used to validate the GE ERIS/SPDS displayed data. The NRC's July 24-27, 1984 Design Verification Audit of the GE ERIS/SPDS design included a review to assure the data displayed are valid. The staff's draft SER on SPDS for GESSAR II addressed data validation and confirmed, "that means are provided in the SPDS design to assure that the data displayed are validated."(2) In addition, the GSU April 24, 1984 SPDS Safety Analysis Report addresses data validation.(5)

3. VERIFICATION AND VALIDATION PROGRAM

Define and discuss the Verification and Validation Program Plan which was used in the development of the SPDS. Also, describe results to date from the Verification and Validation Program, and the corrective actions taken to address identified design deficiencies.

RESPONSE

Verification and validation (V&V) of the SPDS consists of many activities which take place throughout the entire design and implementation process. This includes hardware hookup, software, data base and display corrections as well as interface and interaction needs. The RBS V&V Program is based on the GE ERIS/SPDS V&V Program which is patterned after NSAC-39.[6] The NRC's July 24-27, 1984 Design Verification Audit of the GE V&V Program concluded that the GE V&V Program," is similar to the one described in NSAC-39 and it is acceptable for the development of an SPDS."

GE Validation Test Procedures have been developed. The Validation Test for the GE ERIS/SPDS has been conducted and the test results provided to the NRC in NEDC-30885.[3] The GE submittal identifies the problems found during the tests and how they were resolved.

The calibration method (Re: G-CAL-016) used for all permanent plant ERIS signals during preoperational testing verifies the hardware hookup from the signal source through all intervening hardware, and verifies the software point id used by ERIS to identify the signal. Documentation is automatically provided by the as-installed database, which can provide complete connection details for any given point. In addition, all ERIS hardware drawings are field verified during this process and corrections are made such that permanent plant files documentation represents the "as-built" status of all hardware.

Database and calibration corrections will be made to points and display algorithms driving ERIS displays throughout startup testing. This process will be necessary since most of the display points require operation at power to test the functional response. Necessary changes will be made to include the adjustment of database plant-specific constants after testing, corrections to display-driving algorithms, and the assignment of point processing parameters. The process will allow for a more accurate determination of plant-specific constants and the correction of logical errors, and will ensure that the accuracies required by the ERIS software are provided. Final database constants and logical changes will be reviewed by GE and included under the V&V plan.

Operator training on the use of the SPDS will be provided by the Nuclear Training Department. Training will be included as part of the River Bend simulator training and will provide this training as an integral part of Licensed Operator initial and requalification training programs. Training on the use of the SPDS will also be provided to the Shift Technical Advisors. Additional or specialized training will be provided as needed by the Technical Staff Computer Systems Group.

4. UNREVIEWED SAFETY QUESTIONS

Provide conclusions regarding unreviewed safety questions or changes to technical specifications.

RESPONSE

The RBS Technical Specifications are currently in a proof-and-review/final copy stage and are scheduled to be finalized by fuel load. Any safety evaluation completed after issuance of the RBS operating license under the provision of 10CFR50.59 that identifies an unreviewed safety question will be reviewed by the Nuclear Review Board (NRB) as discussed in Section 6.5.3 of the RBS Technical Specifications. Station procedures are being developed at this time to require that all future design changes (i.e. post-O.L.) be reviewed for their impact on the design basis of the plant. From this review process unreviewed safety questions as well as proposed Technical Specification changes (10CFR50.59) would be identified and properly resolved. Resolution of 10CFR50.59 unreviewed safety questions and Technical Specification changes require NRC approval per 10CFR50.90.

5. PARAMETER SELECTION

a. The applicant has stated that General Electric (GE) is furnishing the River Bend Station SPDS. Identify the specific design (e.g., generic ERIS) being provided by GE.

b. Concerning reactivity control monitoring, the applicant does not include Source Range Monitors (SRMs) as part of the SPDS. While the staff agrees that the Average Power Range Monitors (APRMs) are of primary interest for flux monitors, the SRMs should also be available on SPDS to serve as a key reactivity control monitor for events at low power. Therefore, to provide a rapid, continuous reactivity status indication covering all conditions, the staff requests that the SRMs be specifically included as part of the SPDS display, or justification provided for not including this parameter.

c. Concerning radioactivity control monitoring, the applicant does not include Primary Containment Radiation as part of the SPDS. To provide a Radioactivity Control safety function monitor under conditions when the vent stack and other monitors are not available (e.g., containment isolation), the staff requests that a containment radiation monitor be included as part of the SPDS display, or justification provided for not including this parameter.

d. How is the status of containment isolation identified on the SPDS?

e. Provide a Listing or Table relating the SPDS safety parameters to the five Critical Safety Functions identified in 4.1(f) of Supplement 1 to NUREG-0737 (Generic Letter No. 82-33).

RESPONSE

a. The RBS SPDS design is consistent with the SPDS function of the GE ERIS as described in NEDE-30284-P dated November 1983.(1)

b. River Bend Station (RBS) considers its SPDS to be a useful operator aid as well as a diagnostic tool for use by the plant operators and technical support personnel during a transient or accident event. The database upon which the SPDS draws for displaying information was derived with the idea that it would be necessary and sufficient for allowing RBS to reconstruct and identify what the plant was doing at any moment in time. Therefore, not all plant parameters are monitored and displayed by the system. One such parameter is the count rate produced by the source range monitors (SRMs). The SRM count rate has significance after a SCRAM by indicating the trend for neutron flux. This is true whether the detectors are inserted or withdrawn from the core. The operator will monitor this parameter on the main control console 1H13*P680. Given this parameter's restricted usage, RBS opted not to monitor it as part of the SPDS instead choosing to monitor control rod positions, average power range monitor flux levels, and the position of the safety relief valves. The aforementioned parameters give a positive indication of reactivity control following a SCRAM. This fact and given the low cost/benefit ratio of monitoring the SRMs caused RBS not to monitor the subject variable as part of its SPDS.

The NRC Staff has stated in its draft SER evaluation of SPDS in GESSAR II that..."The Staff finds that the variables selected for the GE SPDS would be acceptable with the addition of radiation monitors to identify the status of the Radioactivity Control Safety Function." The GE SPDS described in GESSAR II is similar to RBS and does not monitor the SRM count rate. It is apparent that both RBS and the NRC Staff are in agreement with the omission of monitoring the SRMs owing primarily to the fact that the SPDS is not safety-related, serves as an operator aid only, and is presently furnished with an adequate database upon which to ascertain reactivity control.

c. The design of River Bend Station (RBS) includes the implementation of a sophisticated digital radiation monitoring system (DRMS) described in Chapter 7 of the RBS FSAR. The system furnishes Class 1E readouts in the main control room. Additionally, wherever an SPDS display console is located there is located in its immediate vicinity a DRMS display console. It was decided by RBS engineering and operations personnel that the DRMS and SPDS databases would be kept separate owing to cost/benefit ratio considerations. The DRMS performs a specialized function in regards to monitoring the Radioactivity Control Safety Function and as such is viewed as a separate display of the SPDS even though its data processing system is separate from the SPDS. Therefore, RBS has found no basis for

combining the two databases for display on the SPDS console. Operator usage of this design has been shown to be acceptable from a human factors standpoint given the use of the SPDS as an operator aid.

d. The status of containment isolation will be displayed by the ERIS SPDS in two (2) ways. The first is an 'event box' on selected displays that shows group isolation status (see attached Figures 1, 2 and 3). This box will indicate "NO ISLN" (no isolation), "ISLN CMD" (isolation command), or "ISOL" (isolation). The text within the event box always appears white. The border of the event box changes color to reflect current status. During normal operation, the border will be blue and the text will read "NO ISLN". When an isolation command is sensed, the text will read "ISLN CMD", and the box border will turn yellow. If a valve(s) does not isolate, the border of the box will turn red. The border will also turn red if a valve(s) fails to isolate by a pre-determined time. The pre-determined time constant represents the upper limit of the expected travel times for valves after an isolation signal. If isolation is successful, the border of the event box will turn green and the text will read "ISOL". The border will also change from red to green, and the text to "ISOL", after a slow valve has achieved it's isolation position. The valves that are designed to respond to an isolation command are used as input to the algorithm that drives this event box, so that the failure of any valve to isolate will alert the operators to a problem.

   The second method that will report isolation status is a display designed as a result of the Detailed Control Room Design Review. A display was designed for the SPDS which will identify any valve (by system) that fails to isolate. If all valves isolate, the display will merely display "ALL VALVES ISOLATED", and will not display any data for individual valves. The list of valves that will be used as input to this display are included as Attachment 1.

e. A table relating the SPDS safety parameters to the five Critical Safety Functions identified in 4.1(f) of Supplement 1 to NUREG-0737 (Generic Letter 82-33) is provided in Table 1, attached.

6. PARAMETER VALIDATION

Insufficient information is provided to evaluate the validation testing used for the SPDS system. Specifically, the identification of the transient and accident sequence test cases used for performance tests (useability) of the SPDS should be provided and justified. If a specific parameter is not testable in a fully simulated transient sequence, the source of the validation data should be identified.

RESPONSE

The validation testing for the GE generic SPDS system has been completed. A report, NEDC 30885 covering this testing, was submitted to the NRC in April 1985. NRC questions on this report have been responded to and the NRC is expected to issue a supplemental SER resolving the outstanding NRC concerns with the SPDS for GESSAR II.

NEDC-30885 reports on the test methods and results used to validate the performance of the SPDS system. The test used simulated reactor transients which had been prepared to simulate standard reactor transients for operator training on a simulator. In addition, a simple ramp transient was used to provide precise timing of system limit changes and range checks. This ramp transient allowed testing of those parameters which were not fully tested by the reactor transients.

The hardware and software used on the River Bend Station (RBS) ERIS/SPDS is the same as used in the GE ERIS/SPDS design. Therefore, the testing reported in NEDC-30885 applies to the RBS ERIS/SPDS system.

REFERENCES

1. NEDE-30283-P, "Licensing Topical Report for the General Electric Company Proprietary Information," November 1983.

2. Letter, from Cecil O. Thomas, NRC, to Glen G. Sherwood, General Electric, Subject; "Draft SER on the Safety Parameter Display System for GESSAR II," December 18, 1984.

3. NEDC-30885, "Generic ERIS (Basic RTAD) Software Validation," April 1985.

4. Letter, from H. C. Pfefferlen, General Electric, to Leo Beltracchi, NRC; Subject, "Open Items from Draft SER on GESSAR II SPDS," December 20, 1984.

5. Letter, from J. E. Booker, Gulf States Utilities Company, to Harold R. Denton, NRC; Subject, "River Bend Station SPDS Safety Analysis," April 24, 1984.

6. NSAC-39, "Verification and Validation for Safety Parameter Display Systems," December 1981, Nuclear Safety Analysis Center, Electric Power Research Institute.

7. Letter, from J. E. Booker, Gulf States Utilities Company, to Harold R. Denton, NRC; Subject; "River Bend Station SPDS Safety Analysis Errata Sheets," May 2[illegible], 1984.

8. Letter from G.G. Sherwood, General Electric, to H. L. Thompson, NRC; Subject, "Emergency Response Information System (ERIS)", May 9, 1985.

ATTACHMENT 1

VALVES TO BE USED TO DRIVE CONTAINMENT ISOLATION STATUS DISPLAY

MSIVs
- 1B21*AOVF022A
- 1B21*AOVF028A
- 1B21*AOVF022B
- 1B21*AOVF028B
- 1B21*AOVF022C
- 1B21*AOVF028C
- 1B21*AOVF022D
- 1B21*AOVF028D

Turb. Plt.
- 1B21*MOVF067A
- 1B21*MOVF067B
- 1B21*MOVF067C
- 1B21*MOVF067D
- 1B21*MOVF016
- 1B21*MOVF019

RHR
- 1E12*MOVF053A

Floor Drains
- 1DFR*AOV102
- 1DFR*AOV101

Fire Prot.
- 1FWP*MOV121

Service Air
- 1SAS*MOV102

Instrument Air
- LIAS*MOV106

RPCCW
- 1CCP*MOV138
- 1CCP*MOV158
- 1CCP*MOV159

Service Water
- 1SWP*MOV5A
- 1SWP*MOV5B

Ventilation Chilled Water Return
- 1HVN*MOV102
- 1HVN*MOV128
- 1HVN*MOV127

Cond. Make-Up
- 1CNS*MOV125

RWCU
- 1WCS*MOV178
- 1WCS*MOV172
- 1G33*MOVF040
- 1G33*MOVF039
- 1G33*MOVF001
- 1G33*MOVF004

RCIC
- 1E51*MOVF077
- 1E51*MOVF078

FPC
- 1SPC*MOV119
- 1FSC*MOV120
- 1FSC*MOV122
- 1FSC*MOV139
- 1SFC*MOV121
- 1C11*MOV083

Containment and Drywell Purge and Containment Hydrogen Purge Out. Lines
- 1HVR*AOV165
- 1HVR*AOV123
- 1HVR*AOV128
- 1HVR*AOV166

Equipment Drains
- 1DER*AOV127
- 1DER*AOV126

TABLE 1

SPDS Monitored Variables
River Bend Station - Unit 1

CRITICAL SAFETY FUNCTION	MONITORED VARIABLE
Reactivity Control	reactor power level (APRM system) control rod positions SCRAM status (all rods in) SRV positions
Reactor Core Cooling and Heat Removal	RPV water level RPV water temperature MSIV positions SRV positions
Reactor Coolant System Integrity	RCPB isolation valve positions RPV pressure drywell pressure containment pressure drywell sump levels
Containment Integrity	containment temperature drywell temperature drywell pressure containment pressure suppression pool water level suppression pool water temperature containment isolation valve positions containment/drywell hydrogen concentration
Radioactivity Control	* ** area radiation levels * HVAC damper positions ** primary containmetn radiation levels ** containment effluent radioactivity containment isolation valve position ** airborne radioactivity release activities ** liquid effluent radioactivity levels

TABLE 1 (cont'd.)

SPDS Monitored Variables
River Bend Station - Unit 1

* Input data furnished from auxiliary, fuel, and reactor building locations
** Information displayed on the DRMS console in the main control room

NOTE

The abbreviations used herein are defined as follows:

APRM	average power range monitor
SRV	safety/relief valve
SPV	reactor pressure vessel
MSIV	main steam isolation valve
RCPB	reactor coolant pressure boundary
HVAC	heating, ventilation, and air conditioning

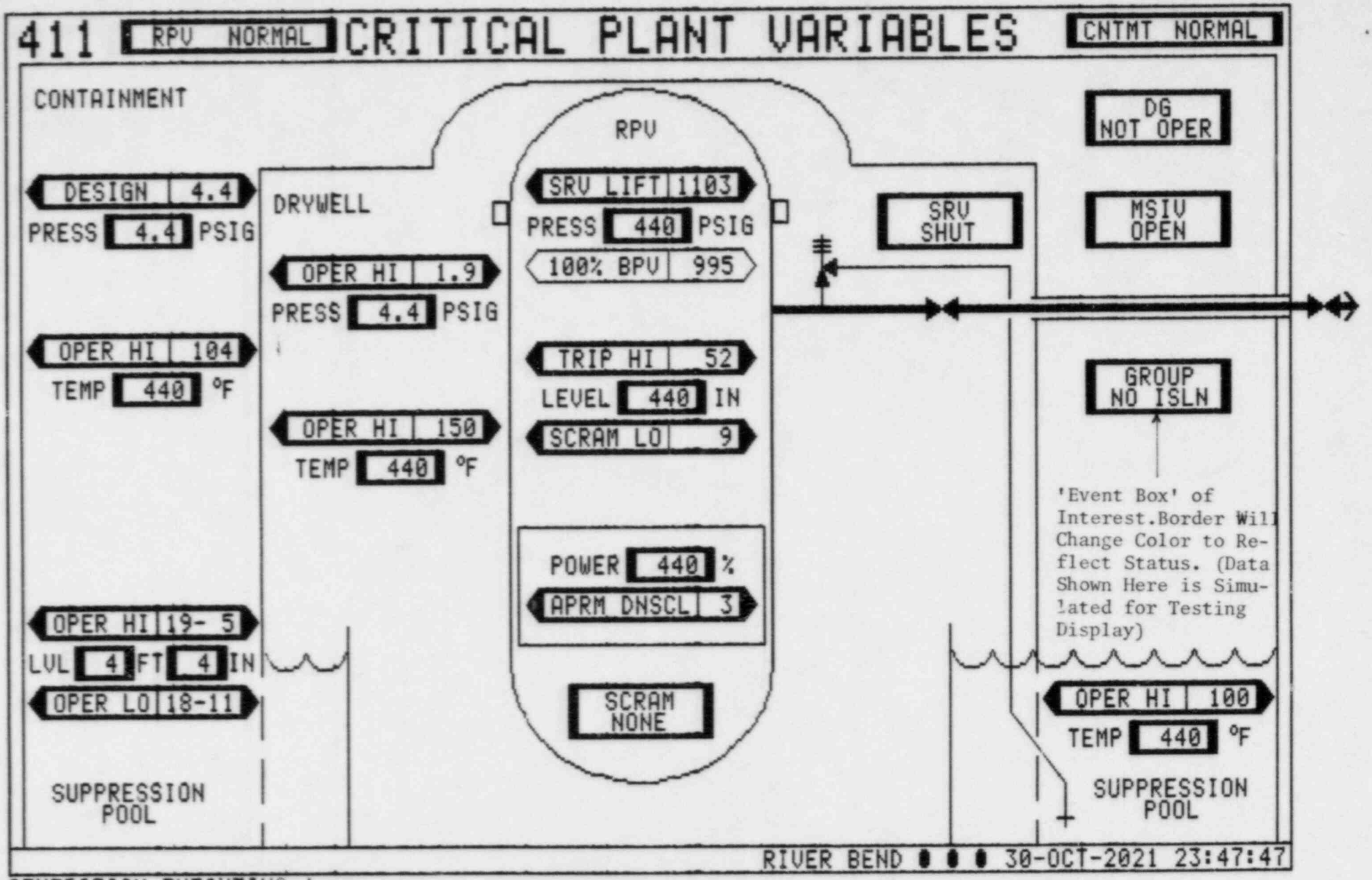
411
RPV NORMAL
CRITICAL PLANT VARIABLES
CNTMT NORMAL
CONTAINMENT
DRYWELL
RPV
DG NOT OPER
SRV SHUT
MSIV OPEN
DESIGN 4.4
PRESS 4.4 PSIG
OPER HI 1.9
PRESS 4.4 PSIG
OPER HI 104
TEMP 440 °F
OPER HI 150
TEMP 440 °F
SRV LIFT 1103
PRESS 440 PSIG
100% BPV 995
TRIP HI 52
LEVEL 440 IN
SCRAM LO 9
POWER 440 %
APRM DNSCL 3
SCRAM NONE
GROUP NO ISLN
'Event Box' of Interest.Border Will Change Color to Reflect Status. (Data Shown Here is Simulated for Testing Display)
OPER HI 19- 5
LVL 4 FT 4 IN
OPER LO 18-11
SUPPRESSION POOL
OPER HI 100
TEMP 440 °F
SUPPRESSION POOL
RIVER BEND 30-OCT-2021 23:47:47
GENERATION EXECUTING !

FIGURE 1

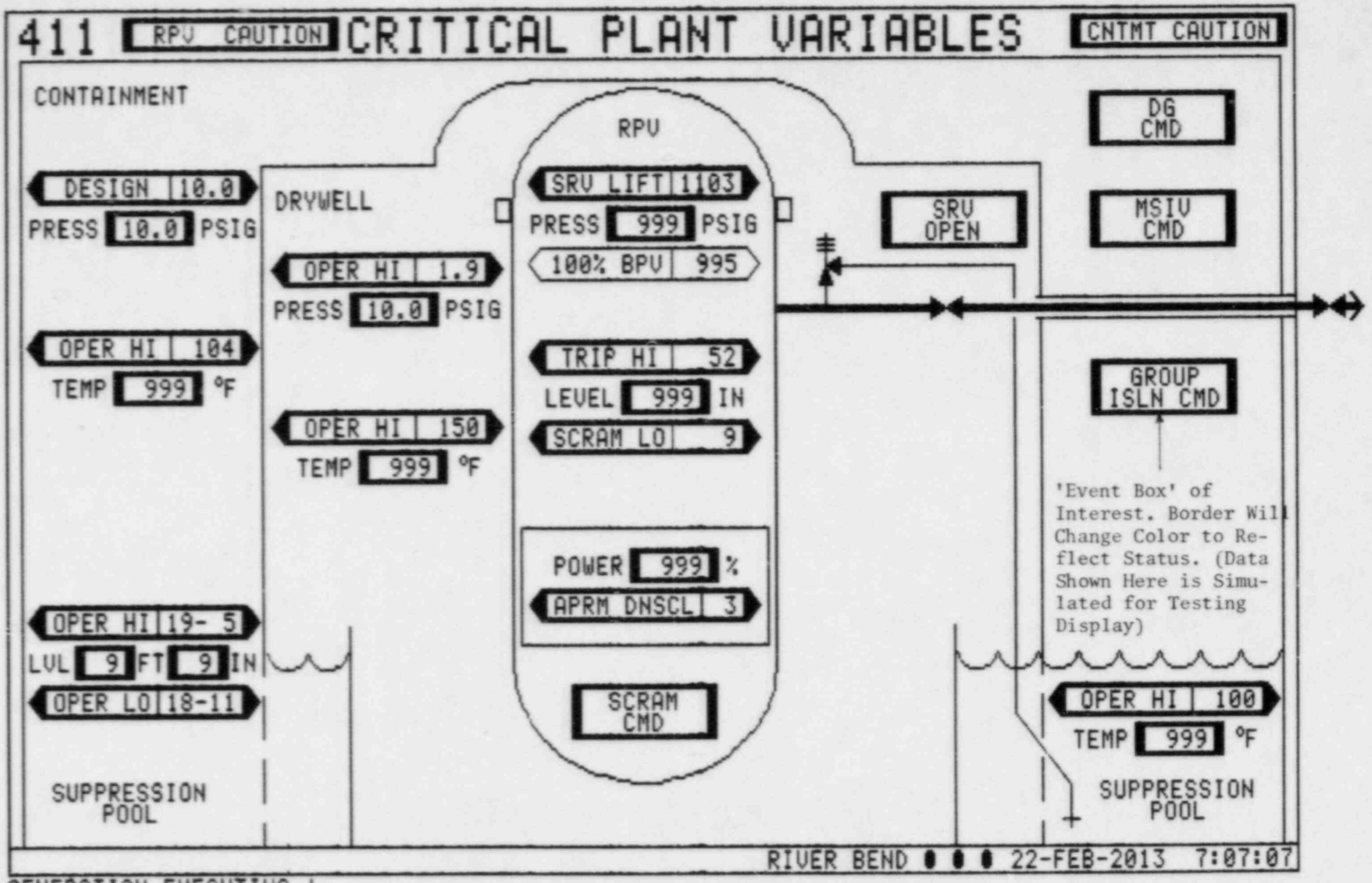
411
RPV CAUTION
CRITICAL PLANT VARIABLES
CNTMT CAUTION
CONTAINMENT
DRYWELL
RPV
DESIGN 10.0
PRESS 10.0 PSIG
OPER HI 1.9
PRESS 10.0 PSIG
OPER HI 104
TEMP 999 °F
OPER HI 150
TEMP 999 °F
SRV LIFT 1103
PRESS 999 PSIG
100% BPV 995
TRIP HI 52
LEVEL 999 IN
SCRAM LO 9
POWER 999 %
APRM DNSCL 3
SCRAM CMD
DG CMD
SRV OPEN
MSIV CMD
GROUP ISLN CMD
'Event Box' of Interest. Border Will Change Color to Reflect Status. (Data Shown Here is Simulated for Testing Display)
OPER HI 19- 5
LVL 9 FT 9 IN
OPER LO 18-11
SUPPRESSION POOL
OPER HI 100
TEMP 999 °F
SUPPRESSION POOL
RIVER BEND 22-FEB-2013 7:07:07
GENERATION EXECUTING !

FIGURE 2

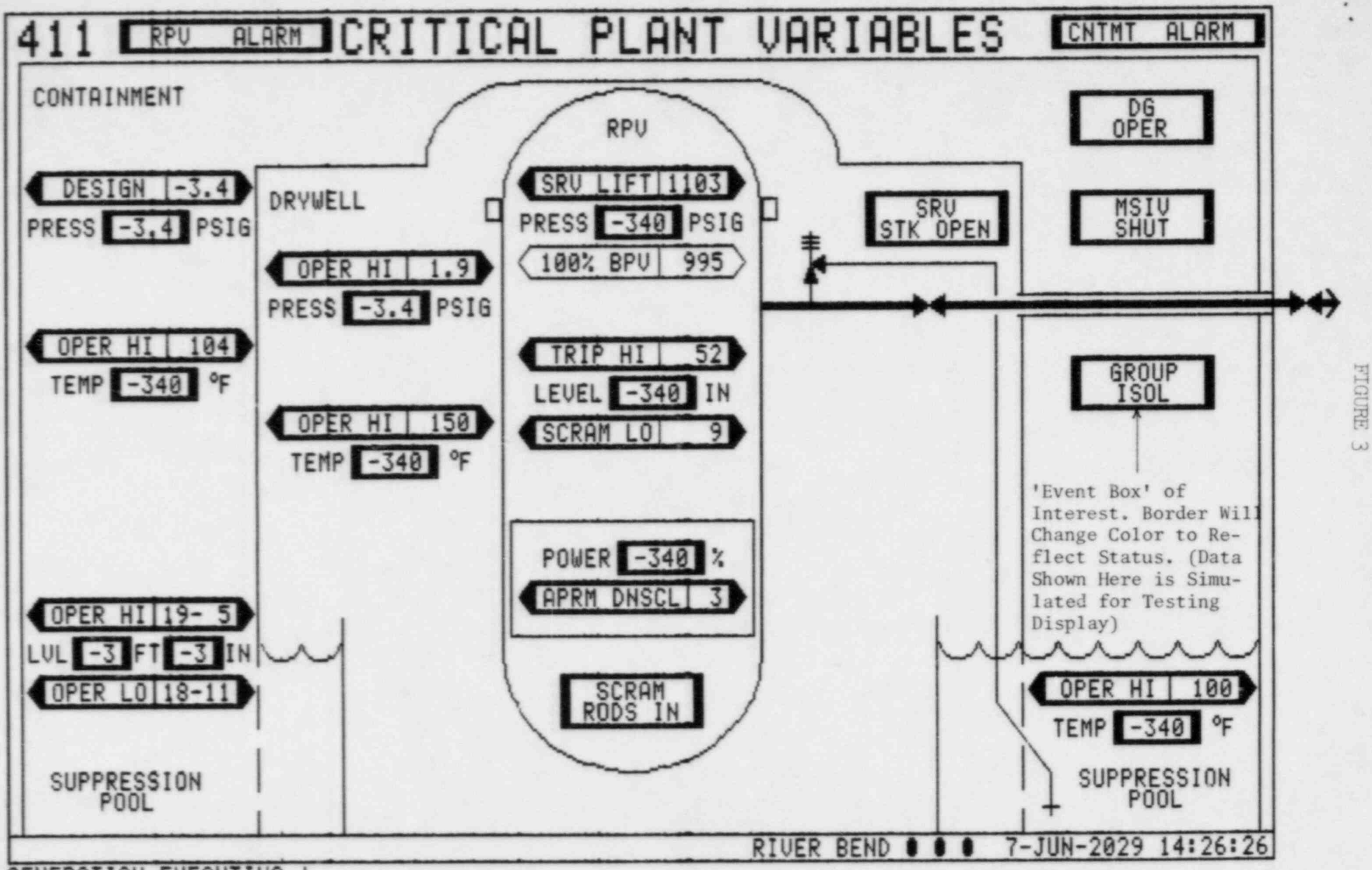
411
RPV ALARM
CRITICAL PLANT VARIABLES
CNTMT ALARM
CONTAINMENT
DESIGN -3.4
PRESS -3.4 PSIG
OPER HI 104
TEMP -340 °F
OPER HI 19- 5
LVL -3 FT -3 IN
OPER LO 18-11
SUPPRESSION POOL
DRYWELL
OPER HI 1.9
PRESS -3.4 PSIG
OPER HI 150
TEMP -340 °F
RPV
SRV LIFT 1103
PRESS -340 PSIG
100% BPV 995
TRIP HI 52
LEVEL -340 IN
SCRAM LO 9
POWER -340 %
APRM DNSCL 3
SCRAM RODS IN
DG OPER
SRV STK OPEN
MSIV SHUT
GROUP ISOL
'Event Box' of Interest. Border Will Change Color to Reflect Status. (Data Shown Here is Simulated for Testing Display)
OPER HI 100
TEMP -340 °F
SUPPRESSION POOL
RIVER BEND
7-JUN-2029 14:26:26
GENERATION EXECUTING !

FIGURE 3